Draft Submittal

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EXAM NOS. 05000335/2004301
AND 05000389/2004301

AUGUST 9 - 20, 2004

1. Senior Reactor Operator Written Exam

Nuclear Regulatory Commission Senior Reactor Operator Licensing Examination

St Lucie Nuclear Plant

ANSWER KEY

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Bank Question: 353.3 ***Answer: C***

1 Pt(s) A male worker needs to repack a valve in an area that has the following radiological characteristics:

- General area dose rate = 30 mrem/hr
- Airborne contamination concentration = 20 DAC
- The protection factor for a full-face respirator = 100

The worker's present exposure is 830 mrem for the year.

The job will take 4 hours with a mechanic wearing a full-face respirator. It will only take 2 hours if the mechanic does NOT wear the respirator.

Which of the following choices for completing this job would maintain the workers exposure within the Station ALARA requirements?

References Provided: ADM-05.01 Appendix C

A. The worker should wear the respirator because the ALARA program requires respirator protection for all work in contaminated areas.

B. The worker should NOT wear the respirator because the dose received will exceed neither NRC nor site annual personnel dose limits.

C. The worker should wear the respirator because the total TEDE dose received will be less than if he does not wear one.

D. The worker should NOT wear the respirator because the total TEDE dose received will be greater than if he wears one.

Distracter Analysis:

Radiation exposure comparison:
Without respirator:
DDE = 30 mrem/hr x 2 hr = 60 mrem
From airborne contamination:
CEDE = 20 DAC x 2 hr x 2.5 mrem/DAC-hr = 100 mrem
TEDE = 60 + 100 = 160 mrem from job
Total exposure for year = 830 + 160 = 990 mrem

With respirator

DDE = 30 mrem/hr x 4 hr = 120 mrem
CEDE = 0 (2 mrem if appendix C is used)
TEDE = 120 mrem
Total exposure for year = 120 + 830 = 950 mrem

(With respirator) (Without respirator)
TEDE = 120 mrem < 160 mrem = use a respirator

IF THE CANDIDATE USES THE TEDE ALARA EVALUATION FORM, THE DOSE W/ RESPIRATOR IS 73 mrem, BUT THE ANSWER IS THE SAME.

A. **Incorrect**: The ALARA program provides for a respirator assessment.
Plausible: This minimizes internal exposure, and was the policy in prior years.

B. **Incorrect**: While the dose will not exceed any limits, this is not the criterion for maintaining exposure ALARA
Plausible: This is a legal approach to the problem

C. **Correct answer**

D. **Incorrect**: The exposure will be greater if you DO NOT wear the respirator
Plausible: If the candidate incorrectly computes the exposure

Level: SRO Exam

KA: G2.3.2 (2.9)

Lesson Plan Objective: none

Source: Bank PSHA #353

Level of knowledge: comprehension

References:
1. ADM-05.01 pages 13-14, 29
2. HPP-63 pages 1-12

K/A G2.3.2: Knowledge of the facility ALARA program. (CFR 41.12 / 43.4 / 45.9 / 45.10)

Objective: none

Bank Question: 1093CE ***Answer: A***

1 Pt(s) Unit 2 was operating at 100% power when ASI approached the Tech Spec limit. Given the following events and conditions:

- Group 5 was inserted to 132"
- One CEA in group 5 slips to 121"
- The CEA Secondary Position Display is selected to slipped CEA

Which one of the following indications will be present for the slipped CEA?

	On RTGB 204 ADS	On RTGB 204 CEA Secondary Position Display	DCS
A.	**121"**	**121"**	**132"**
B.	**121"**	**132"**	**132"**
C.	**132"**	**121"**	**121"**
D.	**132"**	**132"**	**121"**

Distracter Analysis: Note: the training material has not been updated to show that the new "Digital Control System" (DCS) has replaced the old DDPS for Unit 2.

A. **Correct**: The ADS will detect the CEA slippage – and the Digital Backup system will also show the individual rod reed switch position. The DCS will show only group position.

B. **Incorrect**: The digital backup system will detect the slipped rod.
Plausible: If the candidate thinks that the digital backup system does not see individual reed switch position for each CEA.

C. **Incorrect**: The ADS will detect the slipped rod and DCS (formerly DDPS) will not.
Plausible: If the candidate transposes the ADS and DCS display capabilities.

D. **Incorrect**: The displays are reversed.
Plausible: If the candidate transposes the rod positions (i.e. group vs slipped CEA) or reverses the logic of the question.

Level: SRO Exam

KA: SYS 014G2.4.4(4.0/4.3)

Lesson Plan Objective: 0702405-04

Source: Bank #558

Level of knowledge: comprehension

References:
1. 0711405 pages 37-40, 73, 96

SYS 014G2.4.4(4.0/4.3) Rod Position Indication - Ability to recognize abnormal indications for system operating parameters, which are entry-level conditions for emergency and abnormal operating procedures. (CFR 41.10 / 43.2 / 45.6) IMPORTANCE RO 4.0 SRO 4.3

Objective: 0702405-04 Describe the physical interface between the Control Element Drive Systems and the Analogue Display System (Unit 2) and the CEA Position Display System (Unit 1) Digital Data Processing System (DDPS) [now changed to Digital Control System (DCS).

Bank Question: 1100CE ***Answer: A***

1 Pt(s) Unit 2 was starting up after a refueling outage. Given the following events and conditions:

- The letdown system is being placed in service in accordance with NOP-02.02 (*Charging and Letdown*)
- An operator incorrectly positions the isolation valve to the purification filter "B" during the initial valve lineup. The required valve position was "*open*" but the valve is actually *closed*. This error is unknown to the control room.
- The operators align CVCS valves in an attempt to establish letdown flow.

What actions will occur and what procedure are the control room operators required to enter?

A. The letdown system relief valves V-2354 and/or V-2345 open to protect the low pressure letdown piping. Immediately isolate letdown in ONP-02.02 (*Charging and Letdown*)

B. V-2516 (CONTAINMENT ISOL VALVE - IC) automatically closes on high D/P across the regenerative HX to protect the low pressure letdown piping. Correct the valve misalignment and restore letdown in accordance with NOP-02.02.

C. V-2520 (ION EXCHANGER BYPASS VALVE) automatically opens on high pressure to divert letdown around the ion exchangers and coolant purification filter into the VCT. Correct the valve misalignment and restore letdown in accordance with NOP-02.02.

D. PCV-2201Q (PRESSURE CONTROL VALVE) automatically closes on high pressure to isolate the low pressure letdown piping. Immediately isolate letdown in ONP-02.03.

Distracter Analysis: Note: on Unit 2, coolant purification filter "A" is normally isolated and bypassed. "A" is upstream of the ion exchangers. Coolant purification filter "B" is on line and downstream of the ion exchangers. These filters are in a parallel, not series alignment.

A. **Correct**: Both V-2354 and V-2345 are sized to permit 100% letdown flow to protect the downstream piping.

B. **Incorrect**: There will not be a high D/P condition across the letdown HX. Hi D/P only occurs when there is high flow.
Plausible: Partially correct - If there was a high D/P across the letdown HX -- V-2516 would close.

C. **Incorrect**: V-2520 automatically diverts flow around the ion exchangers but not around the letdown purification filter "B".
Plausible: If the candidate does not know the system alignment.

D. **Incorrect**: PCV-2201Q has a delimiter stop that prevents the valve from closing all the way. The valve may be closed manually – but if this occurred, relief valve V-2345 would open to protect the letdown HX.
Plausible: If the candidate confuses the location of the letdown pressure control valves (PCV-2201P/Q) and the letdown level control valves (LCV-2110P/Q) which are located upstream of relief valve V2345.

Level: SRO Exam

KA: SYS 004A2.24(2.8/2.8)

Lesson Plan Objective: none

Source: New

Level of knowledge: comprehension

References:
1. 0711205 pages 8-16, 45-46, 71, 75
2. 2-NOP-02.02 page 7
3. 2-ONP-02.03 page 4

SYS 004A2.24(2.8/2.8) Chemical Volume Control - Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.24 Isolation of both letdown filters at one time: downstream relief lifts 2.8 2.8 (CFR: 41.5 / 43/5 / 45/3 / 45/5)

Objective 0702205-9(?): Evaluate the effects of CVCS operations on the RCS and other plant parameters to the following:

A. Inadvertent boration
B. Inadvertent dilution
C. Increase or decrease of letdown temperature (e.g. - loss of CCW to letdown heat exchanger)
D. Shifting ion exchanger while divert valve lined up to VCT
E. Loss of Charging and Letdown
F. RCS crud burst

G. Failed fuel element
F. Inadvertent start of a charging pump while the RCS is solid

Bank Question: 1101CE ***Answer: C***

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1 Pt(s) Unit 1 is at 100% power. Given the following events and conditions:

- PIC-1100Y (PRESSURE) is selected for control
- Pressurizer backup heaters B-1 and B-5 are energized
- PIC-1100Y fails to 100% output

Which one of the following statements correctly describes the:
1. Expected plant response, and
2. Appropriate operator action?

A. **1. Pressurizer spray valves closed, all backup heaters on**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure increase.

B. **1. Pressurizer spray valves closed, proportional heater output to maximum**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure increase.

C. **1. Pressurizer spray valves open, proportional heater output to minimum**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure decrease.

D. **1. Pressurizer spray valves open, all backup heaters on;**
2. Enter 1-EOP-01 (*Standard Post Trip Actions*) and recover the plant.

Distracter Analysis: The failure of PIC-1100Y causes the spray valves to fully open and proportional heaters to go to minimum output. Backup heaters will also turn on as actual pressure decreases. The operators should use ONP-10120035 to initially respond to the problem. Selecting manual on HIC-1100 or selecting PCI-1100X will stop the pressure decrease.

A. **Incorrect**: Spray valves open -- not close. Back up heaters do not energize.
Plausible: Partially correct – entering ONP-0120035 is the proper response. If the candidate thinks that the PIC-1100Y failure closes the spray valves – causes a low pressure condition. This thought process partially reverses the failure logic.

B. **Incorrect**: Spray valves open – not close. Proportional heaters energize not deenergize in response to pressure drop for this failure.
Plausible: If the candidate reverses the failure logic, the spray valves would not open to control the pressure increase. Following this logic, pressure would increase and the operators would enter 1-EOP-01 if pressure increased to the hi pressure trip set point.

C. **Correct**:

D. **Incorrect**: Wrong procedure – no need to trip.
Plausible: Partially correct – spray valves open. Backup heaters would trip on. Would enter 1-EOP-01 if unable to close the spray valves. This would be the correct answer if HIC-1100 had failed.

Level: SRO Exam

KA: SYS 010A2.02(3.9/3.9)

Lesson Plan Objective: 0702206-11

Source: NRC exam 2002 #2101

Level of knowledge: comprehension

References:
1. 0711206 pages 21, 38-44, 94, 96-97, 114-118
2. ONOP 1-0120035 pages 4-6

SYS 010A2.02(3.9/3.9) Pressurizer Pressure Control - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Spray valve failures 3.9 3.9 (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Objective 0702206-11: Predict the effects on the PPLCS of a Pressurizer level or pressure control channel failing high or low.

Bank Question: 1103CE ***Answer: A***

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1 Pt(s) Unit 2 was operating at 100% power when a problem occurred with the AFW system. Given the following sequence of events and conditions on 8/21/2004:

- 0100 AFW pump 2C was declared to be out of service due to a governor problem. Repairs are scheduled to be completed in 24 hours.
- 0200 AFW pump 2B motor caught fire when it was started for a surveillance test. Repairs are estimated to take as long as one month.
- 0300 The operators commenced reducing power in preparation to shutdown for repairs.
- 0400 The isolation valve for AFW pump 2A is jammed shut. Repair time is uncertain.
- 0400 Power level = 75%.

What action(s) is required to comply with Tech Specs while the AFW pumps are being repaired?

A. **Stop the shutdown and continue power operation while immediately initiating corrective action to restore at least one AFW pump to operable status. Do not shutdown until one AFW pump has been restored to operability.**

B. **Stop the shutdown for up to 72 hours while repairing the AFW pumps. If repairs to at least one pump are NOT completed within 72 hours, then shutdown to HOT STANDBY within the next six hours and be in HOT SHUTDOWN within the next six hours.**

C. **Continue the shutdown to HOT STANDBY to arrive no later than 0900. Have one AFW pump repaired no later than 0100 on 8/23 and proceed to COLD SHUTDOWN no later than 1300 on 8/23.**

D. **Continue the shutdown to be HOT STANDBY no later than 0900, HOT SHUTDOWN no later than 1500, and COLD SHUTDOWN no later than 1500 on 8/22/2004.**

Distracter Analysis[BCH1]**:** Unit 1 Tech Spec 3.7.1.2 is different than the Unit 2 version of 3.7.1.2. Unit 2 has action statement c that requires remaining at power and not shutting down until repairs are made to at least one AFW pump. Unit 1 Tech Spec 3.7.1.2 requires going to Tech Spec 3.0.3 if all 3 AFW pumps are out of service.

A. **Correct**: Tech Spec LCO 3.7.1.2 Action Statement "c" requires remaining at power and not shutting down until repairs are made to at least one AFW pump.

B. **Incorrect**: Must not shutdown until at least one AFW is operable. **Plausible**: This action is required for one inoperable AFW pump – action statement "a".

C. **Incorrect**: Must not shutdown until at least one AFW is operable. **Plausible**: This action is required for loss of 2 AFW pumps – action statement "b".

D. **Incorrect**: Must not shutdown until at least one AFW is operable. **Plausible**: If the candidates think that Tech Spec 3.0.3 applies. This would be the correct action for Unit 1.

Level: SRO Exam

KA: SYS 061G2.1.2(3.0/4.0)

Lesson Plan Objective: 0702412-13

Source: New

Level of knowledge: comprehension

References:
1. Unit 2 Tech Specs page 3/4 7-4 LCO 3.7.1.2
2. Unit 2 Tech Specs page 3/4 0-1 LCO 3.0.3

SYS 061G2.1.2(3.0/4.0) Auxiliary/Emergency Feedwater - Knowledge of operator responsibilities during all modes of plant operation. (CFR: 41.10 / 45.13) IMPORTANCE RO 3.0 SRO 4.0

Objective 0702412-13: Given a set of plant conditions, identify if the AFAS Technical Specifications LCO requirements are being challenged.

Bank Question: 1104CE **Answer: C**

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1 Pt(s) Unit 1 was operating at 100% power when instrument air header pressure dropped to 85 psig.

Given the following indications on the Unit 1 instrument air dryer:

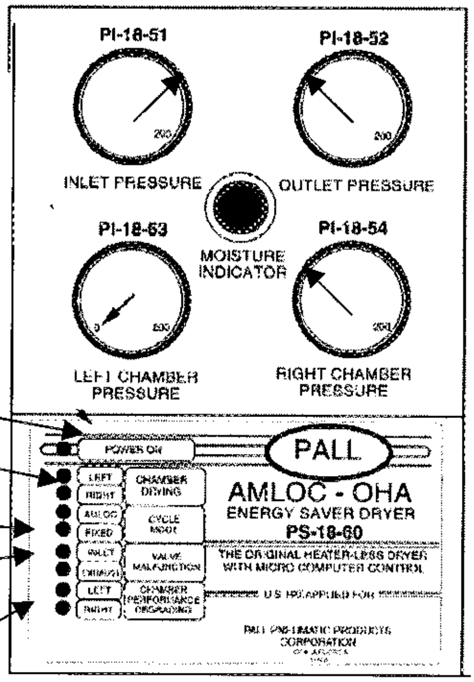
Inlet press = 120 psig
Outlet press = 85 psig
Left chamber press = 0 psig
Right chamber press = 85 psig

Power on – lit
Chamber drying:
- Left – not lit
- Right – lit

Cycle mode:
- AMLOC – not lit
- Fixed – lit

Valve malfunction
- Inlet – lit
- Exhaust – not lit

Chamber performance degrading
- Left – lit
- Right – lit



Which one of the following statements correctly describes the cause of the problem and the correct method to temporarily maintain instrument air pressure until the air dryer can be repaired?

A. **The inlet valve on the right dryer chamber has failed closed. Shift to the left dryer chamber and repressurize the air header.**

B. **The exhaust valve on the right dryer chamber has failed closed. Shift to the left dryer chamber and repressurize the air header.**

C. **The inlet valve on the right dryer chamber has failed closed. Bypass around the air dryer and repressurize the air header.**

D. **The exhaust valve on the right dryer chamber has failed closed. Bypass around the air dryer and repressurize the air header.**

Distracter Analysis:

A. **Incorrect**: The proper operator action is to bypass the dryer unit per 1-ONP-1010030. The dryer unit is computer controlled and there is no way to manually shift to the left chamber. In addition, the left chamber performance degrading indication is lit and the chamber is completely depressurized -- which indicates a leak or failure.
Plausible: Partially correct. A failed closed inlet valve would cause chamber air pressure to drop to header pressure and all the other indications.

B. **Incorrect**: If the exhaust valve failed closed, the "exhaust" annunciator would be lit. Also, the dryer pressure would be at 120 psig inlet pressure.
Plausible: If the candidate reverses the logic between the inlet and outlet valve failure.

C. **Correct**:

D. **Incorrect**: A failed exhaust valve would show chamber pressure at 120 psig.
Plausible: Partially correct – bypassing dryer unit is the correct response. If the candidate does not understand the panel display and mixes up the unit 1 and unit 2 system alignments.

Level: SRO Exam

KA: SYS 078A2.01(2.4/2.9)

Lesson Plan Objective: 0702413-2, 3

Source: New

Level of knowledge: comprehension

References:
1. 0711413 pages 23-24, 29-34, 40-46, 48, 96-97, 99-100, 104
2. ONP 1-1010030 pages 7-8

SYS 078A2.01(2.4/2.9) Instrument Air - Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.01 Air dryer and filter malfunctions 2.4 2.9

Objective 0702413-2, 3:
2: Illustrate the flowpaths of the Plant Air system by:
A. Drawing a one line diagram of the Unit 1 or Unit 2 Instrument Air System. Include only the major components, piping, valves, and instrumentation associated with the supply of Instrument Air. Specifically, include the:
1) Air Compressors

2) Air receivers
3) Air dryers
4) Instrument Air After Filters

3: Describe the instrumentation, available in the control room, used to evaluate the Plant Air System status under normal, off-normal and emergency conditions.

Bank Question: 1107CE **Answer: C**

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1 Pt(s)

Unit 1 was operating at 100% power. Given the following events and conditions:

- Reactor trip occurred 10 minutes ago
- RCS hot and cold leg temperatures are stable and at normal values
- Pressurizer pressure is 1900 psia and lowering
- Pressurizer level is 0%
- Containment pressure is 0.5 psia and rising slowly
- Charging and letdown are responding as expected

Which one of the following break locations would result in these symptoms?

A. **Pressurizer PORV open with a 0.5-inch equivalent break size.**

B. **Reactor vessel vent line between the head vent and the reactor vessel**

C. **Pressurizer spray line between the RCS loop and the spray valve**

D. **Pressurizer level transmitter condensing pot**

Distracter Analysis: This K/A is testing whether the loss of pressurizer level is imminent in a small break LOCA. Pressurizer level will be lost if the break flow exceeds injection flow for a long enough period of time to drain the pressurizer. Nominally break size this is approximately a 1 inch or less.

A. **Incorrect**: A 0.5-inch break in the pressurizer steam space would not cause pressurizer level to drop to 0.
Plausible: If the candidate does not understand the difference between a break in the pressurizer steam space – which would cause PZR level to increase – and a 3 inch break on the pressurizer spray line.

B. **Incorrect**: Reactor vessel head break would not cause the pressurizer to empty. A 7/32" flow-restricting orifice is located close to the reactor vessel to limit leak rate to 44 gpm – within the capacity of a charging pump.
Plausible: A 1-inch break in this location would cause the PZR to empty.

C. **Correct**: Pressurizer spray line break would cause the pressurizer to empty. The pressurizer spray valve would isolate on low pressurizer pressure. The spray line is a 3-inch line.

D. **Incorrect**: The instrument line to the condensing pot has a flow-restricting orifice to limit the break size to within the capability of one charging pump.
Plausible: A break in the pressurizer steam spay would cause the pressurizer level to remain above 0 and perhaps even increase as voids form in the reactor vessel. This was the answer the bank question which was modified.

Level: SRO Exam

KA: EPE 009EA2.06(3.8/4.3)

Lesson Plan Objective: 0702824-5

Source: Mod #2129

Level of knowledge: comprehension

References:
1. 0711824 pages 5-7, 9-12, 52
2. 0711201 pages 24-25, 27, 34, 38-39

EPE 009EA2.06(3.8/4.3) Small Break LOCA - Ability to determine or interpret the following as they apply to a small break LOCA: EA2.06 Whether PZR water inventory loss is imminent 3.8/ 4.3 (CFR: 43.5 / 45.13)

Objective 0702824-5, 8:
5: Describe the effects that the following factors would have on a LOCA
A. RCS and Containment Pressure
B. Break Location

8: Explain the expected trends for the following key parameters during a small break LOCA inside the Containment
D. Pressurizer Level

Objective 0702201-2: Illustrate the flow paths of the RCS by:
A. Drawing a one-line diagram of the RCS. Include labeling of the major components, instrumentation and penetrations to other systems.
B. Drawing a one-line diagram of the Reactor Coolant Gas Vent System. Include labeling of the major components, instrumentation and penetrations to other systems.

Bank Question: 1108CE ***Answer: B***

1 Pt(s) Unit 2 was operating at 100% power. Given the following events and conditions:

- CCW flow to RCP 1A1 motor air cooler has degraded due to flow blockage at the cooler outlet flow-restricting orifice.

Which one of the following conditions will be the FIRST indication of the problem, assuming no operator action?

A. **Increased RCP motor stator temperature**

B. **Increased containment air temperature**

C. **Elevated RCP seal temperature**

D. **Elevated lower RCP motor bearing oil temperature**

Distracter Analysis:

A. **Incorrect**: The motor air coolers cool the air leaving the RCP motor to reduce containment heat loads.
Plausible: If the candidate thinks that the CCW motor coolers cool the air entering the RCP motors.

B. **Correct**: The RCP motor air coolers cool the air leaving the RCP motors to reduce containment air temperature.

C. **Incorrect**: Flow blockage to the motor air cooler does not affect the CCW flow path to the seal coolers.
Plausible: If the candidate does not understand the cooling flow path between the seal cooler and the motor air coolers – thinks they are in series not parallel.

D. **Incorrect**: The CCW motor coolers cool the air leaving the motor -- not entering the motor. A blockage in the CCW supply would not restrict airflow through the cooler but would increase exit air temperature to containment.
Plausible: If the candidate thinks that the lower bearing oil coolers are in series with the motor air coolers for CCW flow,

Level: SRO Exam

KA: APE 015/17 AA2.02(2.8/3.0)

Lesson Plan Objective: none

Source: New

Level of knowledge: comprehension

References:
1. 0711202 pages 22, 27, 55-57, 82

APE 015/17 AA2.02 (2.8/3.0) RCP Malfunction - AA2.02 Abnormalities in RCP air vent flow paths and/or oil cooling system 2.8 3.0 (CFR: 43.5 / 45.13)

Objective: none

Bank Question: 1109CE **Answer: B**

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1 Pt(s) Unit 2 is operating at 100% power steady state with only the 2B charging pump running. Given the following events and conditions:

- Annunciator M-31 (2B CHARGING PUMP TROUBLE) alarms
- The 2B Charging pump has tripped.

Which one of the following statements correctly describes
- the cause of the Charging pump trip, and
- the required operator actions?

A. The Charging pump has tripped on low oil level. Immediately isolate letdown.

B. The Charging pump has tripped on low oil pressure. Immediately isolate letdown.

C. The Charging pump has tripped on low suction pressure. Immediately restart the 2B charging pump.

D. The Charging pump has tripped on low seal tank level. Immediately verify that the 2A (standby) charging pump started.

Distracter Analysis:

A. **Incorrect**: Low oil level will not trip a charging pump.
Plausible: Partially correct – isolate letdown is the proper operator action for a loss of charging pump.

B. **Correct**: Low oil pressure trips the pump at 4 psig on Unit 2 – this trip is not used in unit 1.

C. **Incorrect**: Restarting the 2B charging pump is not a proper operator action – must immediately isolate letdown.
Plausible: Partially correct – the low suction pressure trips the charging pump.

D. **Incorrect**: Low seal tank level does not cause a charging pump to trip. The standby charging pump will not auto start on loss of charging – only on pressurizer level deviation.
Plausible: Most standby pumps start on loss of the running pump.

Level: SRO Exam

KA: APE 022G2.4.49(4.0/4.0)

Lesson Plan Objective: none

Source: Mod #2171

Level of knowledge: memory

References:
1. 0711205 pages 26-27, 50
2. 2-ARP-01-M31
3. 2-ONP-02.03 page 6

APE 022G2.4.49(4.0/4.0) Loss of Reactor Coolant Makeup - Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. RO 4.0 SRO 4.0 (CFR: 41.10 / 43.2 / 45.6)

Objective: None

Bank Question: 1110CE **Answer: B**

1 Pt(s) Unit 2 was heating up in Mode 4 following a refueling outage preparing to come off shutdown cooling. Given the following events and conditions:

- The shutdown cooling full flow relief valves (V-3666, V-3667) are reported to have been set to lift at 375 psig during the outage.
- RCS pressure = 270 psig
- RCS temperature = 145 °F
- PORVs are selected to LTOP
- PC-1106 (PRESSURE LOW RANGE) fails low

The Operations Manager asks the SRO if the heatup can continue and, if so, what is the least limiting requirement? Which one of the following statements provides the correct answer to his question?

(Note: For the purposes of answering this question, the selections are ranked in order of limitation -- A is less limiting than B – which is less limiting than C – which is less limiting than D.)

References Provided: Unit 2 Tech Spec 3.4.9.3
Distracter - Unit 1 Tech Spec 3.4.13

A. **There are no restrictions on heating up to Mode 3 provided PC-1106 channel is placed in a bypassed condition within one hour.**

B. **The heatup can continue provided we can reach raise RCS temperature above 165°F within the next 8 hours. You will then have 7 days to heat up above 247 °F where there are no further restrictions from this event.**

C. **The heatup can continue provided we can reach raise RCS temperature above 247°F within the in 8 hours. You then can heat up to mode 3 with no further restrictions from this event.**

D. **The heat up cannot continue. We must immediately depressurize to atmospheric pressure and vent the RCS.**

Distracter Analysis: The PORVs operate on 2/2 logic. If one pressure channel input fails low, there is no way to get 2 channels actuation. PORV V-1475 is inoperable with PC-1106 failed low. In addition, the SDCRVs are both inoperable because their lifting setpoint has been adjusted above their required setpoint (350 psig). At the present time, there are NO operable low-pressure protection valves

for the RCS. Heating up above 165 °F will restore operability to one PORV.

A. **Incorrect:** The RCS must be heated up to comply with Tech Spec 3.4.9.3. Alternatively, the RCS could be depressurized and vented but this is not the least limiting requirement.
Plausible: If the candidate does not recognize that one PORVs is inoperable and the SDCRVs are both inoperable. Bypassing the failed pressure channel will not satisfy the PORV operability requirements.

B. **Correct**: With SDCRVs inoperable, the RCS must be heated up above 247 °F or depressurized within 8 hours (action b2). But raising temperature above 165 °F will restore operability to one PORV and thus action statement a2 will then apply.

C. **Incorrect**: Not the least limiting answer. When the RCS exceeds 165 °F, action statement A2 will apply and there will be an additional 7 days AOT with one operable LTOP protective device.
Plausible: If the candidate does not recognize that the PORV will be operable when RCS temperature exceeds 165 °F.

D. **Incorrect**: Although depressurizing and venting the RCS will comply with Tech Specs, it is NOT the correct answer to the OPS Manager's question. The heat up CAN continue.
Plausible: If the candidate does not interpret Tech Specs correctly or elects to provide an unrealistically conservative answer.

Level: SRO Exam

KA: APE 025AA2.06(3.2*/3.4*)

Lesson Plan Objective: 0702206-15

Source: New

Level of knowledge: analysis

References:
1. Unit 2 Tech Spec 3.4.9.3 pages 3/4 4-35 to 37a - PROVIDED
2. Unit 2 Tech Spec 3.4.13 page 3/4 4-59 - PROVIDED
3. 0711206 pages 24-26, 28-29
4. 0711207 page 58
5. Tech Spec 3.3.2 & Table 3.4-3

APE 025AA2.06(3.2*/3.4*) Loss of RHR System - Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: AA2.06 Existence of proper RHR overpressure protection 3.2* 3.4* (CFR: 43.5 / 45.13)

Objective 0702206-15: Given a set of plant conditions, identify if the PPLCS related Tech Spec LCO requirements are being challenged.

Bank Question: 1111CE **Answer: D**

1 Pt(s) Unit 1 was operating at 100% power. Given the following events and conditions:

- V-1402 (PORV) is in NORMAL
- V-1403 (BLOCK VALVE) is OPEN
- V-1404 (PORV) is in NORMAL
- V-1405 (BLOCK VALVE) is CLOSED
- V-1403 failed on thermal overload. V-1403 is full open. The estimated time to repair the MOV is 90 minutes.

Which one of the following statements correctly describes the required Tech Spec actions?

A. **Open V-1405 – remain at 100% power**

B. **Close V-1402 and remove power from the valve – remain at 100% power**

C. **Open V-1405 and remove power from V-1403 – remain at 100% power**

D. **Commence shutdown to hot standby**

Distracter Analysis: Unit 1 Tech Spec 3.4.12 action "a" requires any inoperable PORV block valves to be closed with power removed within one hour. If this cannot be achieved, then shutdown must be commenced. This is not required by Unit 2 Tech Specs -- which allows 2 PORVs to be aligned for operation at the same time.

A. **Incorrect**: Insufficient action -- does not comply with action a.
Plausible: satisfies action b and provides a PORV aligned to an operable block valve.

B. **Incorrect**: does not satisfy action a.
Plausible: Closes off one PORV relief path and provides only one PORV aligned to an open block valve. If candidate thinks that you can swap a PORV for a block valve.

C. **Incorrect**: Insufficient action – does not comply with action a.
Plausible: Puts plant in an alignment that has one PORV block valve open and the other PORV block valve closed. This is close to the requirement – a reversal of the power removal sequence.

D. **Correct**: Must commence shutdown if unable to repair V-1403 within one hour.

Level: SRO Exam

KA: APE 027G2.1.30(3.4/4.1)

Lesson Plan Objective: 0702206-15

Source: New

Level of knowledge: memory

References:
1. Unit 1 Tech Spec LCO 3.4.3
2. 0711206 pages 30-31
3. Unit 2 Tech Spec LCO 3.4.4
4. Unit 1 Tech Spec LCO 3.4.13

APE 027G2.1.30(3.4/4.1) Pressurizer Pressure Control System Malfunction - Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) IMPORTANCE RO 3.4 SRO 4.1

Objective 0702206-15: Given a set of plant conditions, identify if the PPLCS related Tech Spec LCO requirements are being challenged.

Bank Question: 1112CE **Answer: B**

1 Pt(s) Unit 1 was operating at 100% power when a reactor trip occurred. Given the following events and conditions.

0200 - The turbine tripped for an unknown reason. The turbine trip failed to cause an automatic reactor trip.
0201 - The manual reactor trip caused the CEAs to drop into the core. During the period of time when an ATWS was occurring, the pressurizer PORV lifted and reseated repeatedly.
0202 Three CEAs remained stuck out of the core.
0205 – Safety injection was actuated.
0208 – The SRO first assesses the emergency classification.

What is the correct emergency classification for the above events at 0208 when the SRO determines the classification?

References Provided: EPIP-01 (Classification of Emergency)

A. **No classification is required – the events have already been terminated.**

B. **Alert – the Emergency Coordinator can immediately terminate the event.**

C. **Site Area Emergency – the Emergency Coordinator can immediately terminate the event.**

D. **Site Area Emergency**

Distracter Analysis: The key issue is how to treat an emergency that has already ended for the initial classification. EPIP-01 states:

2 If an EAL was met and the condition completely cleared prior to an emergency classification being declared,
Then:

A. Classify the event in accordance with Attachment 1.
B. Termination of the event
 1. An event classified as an Unusual Event or Alert may be terminated at the time of declaration by the EC.
 2. An event classified as a Site Area Emergency or General Emergency may only be downgraded and/or terminated by the Recovery Manager (RM).

A. **Incorrect**: The event should be classified as an ALERT because of the ATWS.
Plausible: If the candidate misreads the step that allows for immediate termination.

B. **Correct**: The ATWS require classification at ALERT level.

C. **Incorrect**: The criterion for SAE is not met because the operators brought the reactor to a subcritical condition. Also – the Emergency Coordinator does not have the authority to immediately terminate an SAE – only the Recovery Manager can terminate an SAE.
Plausible: If the candidates interpret the ATWS EAL for SAE incorrectly. The SAE EAL for ATWS reads: Failure of the RPS to bring the reactor subcritical when needed and operator actions fail to bring the reactor subcritical. Operator actions have made the reactor subcritical. Also – if the candidates think that a PORV cycling during a pressure transient is a LOCA.

D. **Incorrect**: The criterion for SAE is not met because the operators brought the reactor to a subcritical condition.
Plausible: If the candidates interpret the ATWS EAL for SAE incorrectly. If they determine an SAE is correct – then this distracter allows for the decision that it cannot be terminated. Also – if the candidates think that a PORV cycling during a pressure transient is a LOCA.

Level: SRO Exam

KA: EPE 029G2.4.38(2.2/4.0)

Lesson Plan Objective: none

Source: New

Level of knowledge: comprehension

References:
1. EPIP-01

EPE 029G2.4.38(2.2/4.0) Anticipated Transient w/o Scram - 2.4.38 Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator. (CFR: 43.5 / 45.11) IMPORTANCE RO 2.2 SRO 4.0

Objective 902701-16: Determine who can authorize downgrading classification from all emergency action levels. (EPIPs-02, 06)

Objective 0902701-5: Correctly FILL OUT the appropriate check lists for a given E-Plan and event.

Objective 0702833-3: Identify the plant parameters or plant conditions, which have a significant impact on the classifications of emergencies.

Bank Question: 1113CE **Answer: A**

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1 Pt(s) Unit 1 was at 100% power, responding to a ground on the 1A DC bus. Given the following conditions and events:

- DC bus 1AB is being supplied from DC bus1A.
- Operators are executing NOP-50.01 (*DETERMINATION OF 125V DC BUSES 1A AND 1AB GROUND LOCATION Appendix A*).
- A caution at step 1 of this procedure states:

  CAUTION

  Failure of Battery Charger 1AB amps to rise may indicate the battery charger has not accepted load on 125V DC Bus 1AB. Separation of 125V DC Buses 1A and 1AB may result in loss of Bus 1AB loads.
- The operators fail to heed this caution when opening the 1A to 1AB tie breaker for ground isolation.
- The 1AB DC bus is deenergized when the 1AB battery charger supply breaker trips immediately after opening the 1A-1AB tie breaker.
- The grounds on DC bus 1A have now disappeared.

If battery charger breaker 1AB cannot be re-closed:

1. What is the operability status of the DC buses 1A?
2. What has to be done to promptly reenergize bus 1AB and restore the DC bus alignment to comply with Tech Specs?

A. **1. Bus 1A is operable - only bus loads from 1AB are inoperable**
2. Close the 1A to 1AB bus tie breaker to reenergize the bus.

B. **1. Bus 1A is operable – only bus loads from 1AB are inoperable**
2. Close the 1C to 1AB bus tie breaker

C. **1. Bus 1A is inoperable without bus 1AB being energized**
2. Close the 1A to 1AB bus tie breaker to reenergize the bus

D. **1. Bus 1A is inoperable without bus 1AB being energized**
2. Close the 1C to 1AB bus tie breaker

Distracter Analysis: The bus 1A-to-1AB tie breaker has an under-voltage interlock which prevents reenergizing bus 1A from bus 1AB without first having voltage on bus 1A. The DC bus grounds are clearly isolated to loads on DC bus 1AB.

A. **Correct**: Bus 1A is energized by battery charger 1A and DC battery. The loss of 1AB loads does not affect the operability of bus 1A.

B. **Incorrect**: While this would restore power to bus 1AB, DC bus 1C is not a safety related bus -- loads on bus 1AB are safety related loads. Bus 1AB loads must be powered from bus 1A or 1B.
Plausible: If the candidate thinks that the 1A to 1AB bus tie breaker interlock operates in reverse -- i.e. you must first have power on bus 1AB before you can close the tie breaker.

C. **Incorrect**: DC bus 1A operability is not affected by the loss of DC bus 1AB.
Plausible: Partially correct – the sequence to restore power is correct. The candidate may think that the safety loads on bus 1AB are considered part of bus 1A when these busses are tied together.

D. **Incorrect**: DC bus 1A operability is not affected by the loss of DC bus 1AB. While this would restore power to bus 1AB, DC bus 1C is not a safety related bus – loads on bus 1AB are safety related loads. Bus 1AB loads must be powered from bus 1A or 1B
Plausible: The candidate may think that the safety loads on bus 1AB are considered part of bus 1A when these busses are tied together. The candidate may think that the 1A to 1AB bus tie breaker interlock operates in reverse – i.e. you must first have power on bus 1AB before you can close the tie breaker.

Level: SRO Exam

KA: APE 058G2.1.32(3.4/3.8)

Lesson Plan Objective: 0702503-4, 7, 8.

Source: New

Level of knowledge: comprehension

References:
1. 1-ONP-50.01 Appendix A pages 8-10
2. 0711503 pages 14, 18-19, 58
3. Tech Spec 3.8.2.3

APE 058G2.1.32 Loss of DC Power - Ability to explain and apply all system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) RO 3.4 SRO 3.8

Objectives 0702503-4, 7, 8:

4: Describe the interlocks associated with each of the following:
- A. 125 VDC system.
- B. 120 Volt Instrument AC system.

7. Explain the operation of the 125 Volt DC and 120 Volt Instrument AC Systems operation during all modes of plant operation by:

> A. Describing the sequence of steps involved in transferring power supplies for the AB 125 Volt DC Bus to the alternate DC bus on both Units.
> B. Describing the sequence of steps necessary to place an instrument bus on the maintenance bypass buss on both units.

8: Given a set of plant conditions, identify if the 120 Volt AC and 125 VDC Systems related Tech Spec LCO requirements are being challenged.

Bank Question: 1114CE **Answer: C**

1 Pt(s) Unit 2 is performing a natural circulation cooldown. Given the following events and conditions:

- The unit has been cooling down at 30 °F/hour from hot standby conditions for 5.5 hours.
- Highest CET = 532 °F
- Rep CET = 514 °F
- T_{hot} = 492 °F
- T_{cold} = 467 °F

What is the current saturation pressure of the upper head?

References provided: ONP 2-0120039 Figure 3 and Steam Tables

A. **500 psia**

B. **630 psia**

C. **775 psia**

D. **900 psia**

Distracter Analysis:

A. **Incorrect**: The correct pressure is 775 psia
Plausible: If the candidate uses T_{cold} to determine saturation conditions.

B. **Incorrect**: The correct pressure is 775 psia.
Plausible: If the candidate uses T_{hot} or uses the 50 °F/hr cooldown line.

C. **Correct**: The correct pressure is 775 psia – can be determined using Figure 3 of ONP 2-0120039 or steam tables.

D. **Incorrect**: The correct pressure is 775 psia
Plausible: If he candidate uses the highest CET or misreads the curve – this is the pressure for 4.4 hours.

Level: SRO Exam

KA: APE CE/A13 AA2.2(2.9/3.8)

Lesson Plan Objective: 0702858-9

Source: NRC exam 2001 #1775

Level of knowledge: memory

References:
1. ONP-2-0120039 Figure 3 (Unit 2)
2. ONP-1-0120039 Figure 3 (Unit 1)

CE/A13 AA2.2(2.9/3.8) Natural Circ - Ability to determine and interpret the following as they apply to the (Natural Circulation Operations) AA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. IMPORTANCE RO 2.9 SRO 3.8 (CFR: 43.5 / 45.13)

Objective 0702858-9: RCS Related ONOPs - Given an Off-Normal system condition; use appendices, tables, figures or other ONP attachments to answer system related questions.

Bank Question: 1115CE ***Answer: B***

1 Pt(s) Unit 2 is in Mode 5 on SDC preparing to heatup the RCS. Given the following events and conditions:

- Both Personnel airlock doors are open
- A loss of shutdown cooling occurs and the unit inadvertently heats up to an RCS temperature of 205 °F.

Which one of the following statements correctly describes the status of:
1. containment integrity, and
2. the bases for this requirement

A. **1. Both airlock doors must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR20 during accident conditions.

B. **1. At least one airlock doors must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR100 during accident conditions.

C. **1. At least one airlock door must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR20 during refueling conditions.

D. **1. Airlock doors may be open provided a means for closure is available within the calculated mean time to core boiling.**
2. To limit site boundary radiation dose rates to the limits of 10CFR100 during refueling conditions.

Distracter Analysis:

A. **Incorrect**: The bases for containment integrity is 10CFR100.
Plausible: Partially correct -- Although Tech Spec 3.6.1.1 requires only one air lock door must be closed to establish containment integrity, closing both doors would also meet requirements for containment integrity.

B. **Correct**:

C. **Incorrect**: The bases for this requirement is 10CFR100 not 10CFR20.

Plausible: Partially correct – only one door has to be closed for containment integrity to be established.

D. **Incorrect**: The plant has inadvertently transitioned from Mode 5 to Mode 4. At least one air lock door must be closed in Mode 4.
Plausible: If the candidate does not recognize that the plant is in Mode 4. Partially correct – the requirement is 10CFR100.

Level: SRO Exam

KA: APE 069G2.2.25(2.5/3.7)

Lesson Plan Objective: 0702861-08

Source: Mod #1825

Level of knowledge: memory

References:
1. 0711602 page 50
2. Tech Spec page 1-2
3. Tech Spec LCO 3.6.1.1
4. Tech Spec LCO 3.6.1.3
5. Tech Spec Bases 3/4.6.1.1 pages 3-4
6. Tech Spec Table 1.2
7. ONP 2-1300030 page 3-4

APE 069 Loss of CTMT Integrity - Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2) IMPORTANCE RO 2.5 SRO 3.7

Objective 0702861-08: Given an Off-Normal system condition; describe the basic content and sequence of procedural steps applicable to the Off-Normal Condition by stating:
* Off-normal events addressed in procedure
* Overall strategy to restore system to normal plant conditions
* Basis behind selected Cautions and Notes

Bank Question: 1117CE **Answer: C**

1 Pt(s) Unit 1 is conducting a reactor startup. Given the following events and conditions:

- Criticality has been achieved.
- Group 7 control element assemblies (CEAs) were being withdrawn in 4 step increments to reach the point of adding heat.
- The RCO inadvertently withdraws group 7 CEAs 8 steps.
- All wide range nuclear instrument (WR NI) channel start-up-rate (SUR) pretrips were LIT.
- WR NI channel B SUR peaked at 2.5 decades per minute (DPM) on the recorder. At the same time, the other channels peaked at SUR = +1.5 DPM

Given the following current data:

	% Power	DPM
WR NI Ch A	$5x10^{-4}$	0
WR NI Ch B	$5x10^{-2}$	1
WR NI Ch C	$4.8x10^{-4}$	0.1
WR NI Ch D	$5.2x10^{-4}$	0

Which of the following actions/justifications are required?

A. **Manually trip the reactor because there was a valid SUR greater than 2.0 with no automatic action.**

B. **Continue the startup because wide range SUR protection is not yet in service.**

C. **Place WR NI channel B SUR in bypass because the channel is inoperable.**

D. **Insert the CEAs to the –500 PCM position until the problem with the WR NI channel B instrument is resolved.**

Distracter Analysis:

A. **Incorrect**: One channel failing to trip is not a valid trip signal failure.
Plausible: This would be the right action if multiple channels had failed.

B. **Incorrect**: The SUR protection comes into service at 1x10-4 %, and should have been effective.
Plausible: This could be the answer if the transient occurred earlier and if the data did not support a problem with Channel B.

C. **Correct**:

D. **Incorrect**: The start-up can continue with 3 of 4 channels operable.
Plausible: This would be a prudent but unnecessary action.

Level: SRO Exam

KA: APE032AA2.05 (2.9*/3.2*)

Lesson Plan Objective: 0702403-08

Source: New

Level of knowledge: comprehension

References:
1. 1-ONP-99.01 page 7
2. 1-GOP 302 page 44

K/A 032AA2.05: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: AA2.05 Nature of abnormality, from rapid survey of control room data 2.9* 3.2* (CFR: 43.5 / 45.13)

Objective 0702403-08: Describe the operation of the Unit 1 and 2 Nuclear Instrumentation System (including operating bands or setpoints) during normal, off-normal and emergency conditions.

A. Given a set of plant indications, diagnose a NI channel failure.

Bank Question: 1118CE **_Answer: B_**

1 Pt(s) Unit 2 was in Mode 6. Given the following events and conditions:

- The refueling contractors dropped a new fuel bundle during an inspection

Which one of the following actions is **NOT** a required immediate action on Unit 2?

A. Notify personnel to evacuate the new fuel area.

B. Start shield building ventilation to filter fuel building exhaust.

C. Notify personnel to evacuate the spent fuel pool area.

D. Ensure the fuel bundle is in a safe condition.

Distracter Analysis:

A. **Incorrect**: Evacuation of the new fuel area is an immediate action.
Plausible: Accidents in containment and the SFP only evacuate the primary area. Candidates have to know that multiple areas have to be evacuated for the new fuel accident.

B. **Correct**: A dropped new fuel bundle will not release fission product gasses as with spent fuel. There is no requirement to operate the shield building ventilation. Note that starting shield building ventilation is a subsequent operator action on Unit 1,

C. **Incorrect**: Evacuation of the spent fuel pool area is an immediate action.
Plausible: Accidents in containment and the SFP only evacuate the primary area. Candidates have to know that multiple areas have to be evacuated for the new fuel accident.

D. **Incorrect**: This is an immediate action for the new fuel accident.
Plausible: This is not an immediate action for accidents in the containment or SFP.

Level: SRO Exam

KA: APE 036G2.1.14 (3.3)

Lesson Plan Objective: 0702208-09

Source: New

Level of knowledge: memory

References:
1. ONP 2-0160030 page 5
2. ONP 1-1600030 pages 5, 8-9

K/A APE036G2.1.14: Fuel Handling Accident - Knowledge of system status criteria, which require the notification of plant personnel. (CFR: 43.5 / 45.12) IMPORTANCE RO 2.5 SRO 3.3

Objective 0702208-09: State the immediate operator actions for a damaged fuel element as specified in ONP 1[2]-1600030, "Accidents Involving New or Spent Fuel".

Bank Question: 1119CE **Answer: D**

1 Pt(s) Unit 1 is Mode 3 preparing to cooldown for an outage. Given the following events and conditions:

- Steam generator (S/G) 1B has a small tube leak and has been isolated.
- RCS T_{hot} is <510 °F controlled by the steam bypass system
- A loss of offsite power occurs and Emergency Diesel Generator (EDG) 1A failed due to a seized fuel pump.
- Off-site power is restored 15 minutes after the loss first occurred.

What are the earliest actions required by these conditions and the plant license?

References Provided: *Unit 1 Tech Spec's 3.7.1.2 & 3.8.1.1*

A. AFW is inoperable; be in Mode 4 within 6 hours.

B. AFW is inoperable; be in Mode 4 within 12 hours.

C. AC Power Sources are inoperable; fix EDG 1A within 72 hours.

D. AC Power Sources are inoperable; fix EDG 1A within 14 days.

Distracter Analysis:

A. **Incorrect**: AFW is operable -- capable of fulfilling its functions. The 1A AFW pump has power been restored. The 1B AFW pump has the safety bus powered from offsite - the EDG being OOS does not effect operability (it is considered degraded). The 1C AFW pump is still operable - the tube leak does not affect its ability to pump water into the A S/G.
Plausible: If the candidate forgets that power has been restored – or considers the C AFW pump and one other AFW pump inoperable (B AFW pump due to the B EDG being OOS) then they would go to Tech Spec 3.0.3 and this would be the answer.

B. **Incorrect**: AFW is fully operable (degraded only).
Plausible: Tech Spec 3.7.1.2 specifies 12 hours to Mode 4 when an AFW pump is inoperable for 72 hours. This would be the proper action assuming AFW 1C is inoperable by loss of its S/G 1B supply and AFW 1A is inoperable due to loss of emergency power supply.

C. **Incorrect**: EDG 1A can be out of service for 14 days.
Plausible: If the candidate thinks AFW 1A is inoperable due to EDG failure, Tech Spec 3.7.1.2 has a 72 action statement for AFW

restoration. Also – the change to 14 days for an single EDG out of service is very recent – used to be a 72 hour tech spec AOT.

D. Correct:

Level: SRO Exam

KA: G2.1.33 (3.9)

Lesson Plan Objective: 0702501-20

Source: New

Level of knowledge: comprehension

References:
1. Unit 1 Tech Spec 3.0
2. Unit 1 Tech Spec LCOs 3.7.1.2, & 3.8.1.1-PROVIDED

K/A G2.1.33: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)

Objective 0702501-20: Given a set of plant conditions, identify if the DG related Tech Spec LCO requirements are being challenged.

Bank Question: 1120CE ***Answer: B***

1 Pt(s) In order to increase electric output from the site, the feed system engineer proposes to add a steam driven main feedwater pump to substitute for the motor driven pumps at high power levels.

Because this new pump is not described in the Final Safety Analysis Report (FSAR), which of the following activities must FIRST be completed prior to implementing the modification?

A. **A temporary system alteration to address the drawing discrepancies created by the design change.**

B. **A safety evaluation to determine whether the change creates an unreviewed safety question.**

C. **A Technical Specification change to incorporate the new pump requirements.**

D. **An FSAR change to notify the NRC of the impending design change.**

Distracter Analysis:

A. **Incorrect**: A TSA is not appropriate since this is a permanent modification.
Plausible: TSA's do provide drawing control for temporary modifications to the plant.

B. **Correct**:

C. **Incorrect**: The feedwater pumps are not covered by Tech Specs.
Plausible: Some design changes require Tech Spec changes as a result of the safety evaluation results.

D. **Incorrect**: A safety evaluation is required prior to the design change. FSAR changes are not used to notify the NRC of impending design changes.
Plausible: An FSAR change will have to be submitted after the modification has been implemented to change the FSAR to reflect as-built conditions.

Level: SRO Exam

KA: G2.2.5 (2.7)

Lesson Plan Objective: none

Source: New

Level of knowledge: memory

References:
1. ADM 17-11 pages 15-17

K/A G2.2.5: Knowledge of the process for making changes in the facility as described in the safety analysis report. (CFR: 43.3 / 45.13)

Objective: none

Bank Question: 1121CE ***Answer: C***

1 Pt(s) Unit 1 is performing the initial start-up following core reload. Given the following conditions:

- The reactor has been restarted.
- Criticality occurred at 85 inches on CEA group 7, and subsequently the CEA height was adjusted to 130 inches by boron addition.
- Preparations are in progress to load the main turbine.
- Reactor Engineering has recommended that group 7 CEAs be inserted to 105 inches while loading the turbine continues.

Which response below best describes the basis for reactor engineering's recommendation?

A. **To ensure the new fuel assemblies do not exceed their design peak linear heat rate during the initial power ascension.**

B. **To limit the potential for axial shape index anomalies caused by burn-off of the Gadolinium loaded in the fuel rods at beginning-of-life.**

C. **To assure enough positive reactivity is available to offset a potential temperature decrease with a positive moderator temperature coefficient.**

D. **To limit the power peaking in the core in the event a CEA drops during turbine loading.**

Distracter Analysis:

A. **Incorrect**: The design linear heat flux cannot be approached at low power.
Plausible: Reactor engineering is concerned about various fuel conditioning limits during power ascension.

B. **Incorrect**: Gadolinium effects would be worse with rods inserted.
Plausible: There are ASI effects from uneven burn-off of Gadolinium.

C. **Correct**: At BOL, the MTC can become positive. As the turbine load picks up, Tave will drop and negative reactivity will be added to the core (opposite of + MTC response). The rods must be withdrawn to compensate to prevent the reactor from becoming subcritical.

D. **Incorrect**: The core is designed to withstand the power peaking effects of a dropped CEA.
Plausible: A lower CEA height would lessen the peaking effects of a dropped group 7 CEA.

Level: SRO Exam

KA: G2.2.33 (2.5/2.9)

Lesson Plan Objectives: 0702204-08 & 0702106-06

Source: Bank #206

Level of knowledge: comprehension

References:
1. LP 0711204 pages 24-25

K/A G2.2.33 (2.5/2.9): Knowledge of control rod programming. (CFR: 43.6)

Objectives 0702204-06, 08:

8: Describe how the Technical Specification limits on MTC change as a function of core life and power level.

6: Draw and explain a differential rod worth curve for one group of control rods for reactivity vs. % insertion.

Bank Question: 1122CE ***Answer: D***

1 Pt(s) A General Emergency has been declared at Unit 1 due to a steam generator tube rupture event with fuel damage and large offsite releases. Site evacuation is in progress. The Operations Support Center (OSC) is preparing a work team to gag two main steam safety valves. The three available maintenance technicians have the following radiation exposure histories:

	Current Annual Dose	Lifetime Dose	Age
Worker #1	600 mrem	6 Rem	24
Worker #2	200 mrem	25 Rem	40
Worker #3	50 mrem	45 Rem	40

The OSC supervisor requests the Emergency Coordinator's (EC) permission to authorize the workers to incur 4500 mrem while performing this job.

Which one of the following statements correctly describes the EC's authority to approve dispatching this work team?

REFERENCES PROVIDED: *EPIP-02 & HPP-30 Appendixes 1, 6, & 11*

A. The EC cannot dispatch the team without site vice president approval to exceed FPL exposure guidelines. Reentry requirements are not applicable at this stage of the emergency.

B. The EC can dispatch the team because no NRC annual exposure limits will be exceeded. The OSC Coordinator can authorize reentry.

C. The EC is required to authorize emergency dose exposure limits before dispatching this team. Only the EC can authorize reentry.

D. The OSC supervisor can dispatch the team without EC approval of emergency limits because emergency exposure is controlled separately from the workers' annual exposure limits. The OSC coordinator can authorize reentry.

Distracter Analysis:

A. **Incorrect**: The OSC supervisor can dispatch the team because the emergency dose is less then 5 Rem. The OSC Coordinator controls reentry.
Plausible: All workers will exceed 4500 mrem exposure, which normally requires SVP approval under non-emergency conditions.

B. **Incorrect**: Worker #1 will exceed the 5 Rem NRC annual exposure limit. The OSC Coordinator controls reentry.
Plausible: If the candidate miscalculates or misunderstands the annual exposure limits.

C. **Incorrect**: The supervisor can dispatch the team because the emergency dose is less then 5 Rem.
Plausible: The emergency Coordinator can authorize up to 25 Rem emergency exposure.

D. **Correct**: Annual existing dose shall not be included in ERP exposure guidelines.

Level: SRO Exam

KA: G2.3.4 (2.5/3.1)

Lesson Plan Objective: none

Source: New

Level of knowledge: comprehension

References:
1. HPP-30 page 48 - PROVIDED
2. EPIP-02 pages 36-37

K/A G2.3.4 (2.5/3.1): Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)

Objective: none

Bank Question: 1123CE ***Answer: A***

1 Pt(s) Unit 2 was operating at 100% when an inadvertent ESFAS caused by a human error during I&C testing resulted in both MSIVs closing. Given the following conditions:

- The main steam isolation valves closed.
- The reactor and turbine tripped.
- There were numerous failures of primary and secondary safeties.
- Reactor coolant temperature and pressure peaked at 600°F and 2755 psig, respectively.
- The plant has been stabilized at normal operating temperature and pressure, with no other failures.

What is the earliest report of this incident required to be transmitted to the Nuclear Regulatory Commission?

REFERENCES PROVIDED: *AP 0010721*

A. An immediate telephone report – not to exceed 1 hour.

B. A 4-hour telephone report.

C. An 8-hour telephone report.

D. A 60-day written report.

Distracter Analysis:

A. **Correct**: As required by Tech Spec 2.1.

B. **Incorrect**: An immediate report of safety limit violation is required. **Plausible**: A 4-hour is possible based **§ 50.72(b)(2)(iv)(A)** Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

C. **Incorrect**: An immediate report of safety limit violation is required. **Plausible**: An 8-hour report of safety system actuation is possible. § **50.72(b)(3)(ii)** Any event or condition that results in **(A)** The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;

D. **Incorrect**: An immediate report of safety limit violation is required.

Plausible: A 60-day licensee event report (LER) of Tech Spec violation is also required.

Level: SRO Exam

KA: G2.4.30 (2.2/3.4)

Lesson Plan Objective: 702842-4

Source: New

Level of knowledge: comprehension

References:
1. AP 0010721 page 20 - PROVIDED
2. Tech Spec 2.1.2
3. Tech Spec 6.7.1

K/A G2.4.30: Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR: 43.5 / 45.11)

Objective 0702842-4: STATE the values and required actions for Safety Limits at Unit 1 and Unit 2.

Bank Question: 1124CE ***Answer: C***

1 Pt(s) Unit 1 was operating at 100% power when a LOCA occurred. A plant General Emergency has been declared. Given the following events and conditions:

- A LOCA is in progress with an impaired containment
- CET temperature is 714°
- Wind direction is 123°
- Wind speed is 11 mph

Dose (mrem) at:	**TEDE (Whole Body)**	**CDE (Thyroid)**
1 mile	4700	23000
2 miles	2300	13000
5 miles	950	5100
10 miles	500	1500

Analyze the data below and choose the correct protective action recommendations for the 5-10 mile zone.

REFERENCE PROVIDED: *EPIP – 08 Attachments 1A & 2*

A.

Miles	**No Action**	**Evacuate**	**Shelter**
0-2:	**none**	**All**	**none**
2-5:	**none**	**none**	**NPQ**
5-10:	**none**	**none**	**none**

B.

Miles	**No Action**	**Evacuate**	**Shelter**
0-2:	**none**	**All**	**none**
2-5:	**none**	**NPQR**	**remaining**
5-10:	**none**	**none**	**All**

C.

Miles	**No Action**	**Evacuate**	**Shelter**
0-2:	**none**	**All**	**none**
2-5:	**none**	**NPQR**	**remaining**
5-10:	**none**	**NPQR**	**remaining**

D.

Miles	**No Action**	**Evacuate**	**Shelter**
0-2:	**none**	**All**	**none**
2-5:	**none**	**NPQ**	**none**
5-10:	**none**	**NPQ**	**remaining**

Distracter Analysis: PARs for plant conditions are evacuate the 2-mile radius and 5 mile downwind sectors – shelter all remaining sectors

PARs based on radiological release data are to evacuate the 2-mile radius and the 5 and 10 mile downwind sectors (NPQR), and to shelter all remaining sectors. Sector R is added because the wind direction is on a sector boundary.

A. **Incorrect**: PARs are based on radiological release data - evacuate the 2-mile radius and the 5 and 10 mile downwind sectors (NPQR), and shelter all remaining sectors.
Plausible: These are the PARs based on plant conditions without imminent core damage -- i.e. "default" or minimum PARs

B. **Incorrect**: PARs are based on radiological release data - evacuate the 2-mile radius and the 5 and 10 mile downwind sectors (NPQR), and shelter all remaining sectors.
Plausible: These are the PARs based on plant conditions with imminent core damage. They are also correct if the candidate does not note that the CDE dose at the 5 mile radius exceeds CDE limit and requires evacuation of the 5-10 mile downwind sectors.

C. **Correct**:

D. **Incorrect**: Does not shelter the 2-5 mile remaining sectors. Sector R must be included because wind direction is on the edge of a sector.
Plausible: If the candidate picks non-conservative actions because he/she does not note that the 2-5 mile non-downwind sectors are omitted from being sheltered and does not see that wind direction is at a boundary.

Level: SRO Exam

KA: G2.4.40 (2.3/4.0)

Lesson Plan Objective: 0902701-21

Source: Mod #1227

Level of knowledge: analysis

References:
1. EPIP-08 Attachments 1A & 2 - PROVIDED

K/A G2.4.40 (2.3/4.0): Knowledge of the SRO's responsibilities in emergency plan implementation. (CFR: 45.11)

Objective 0902701-21: Given the results from an offsite dose calculation make the protective action recommendations.

Bank Question: 1125CE ***Answer: B***

1 Pt(s) Unit 1 is operating at full power. Given the following events and conditions:

- A radioactive liquid release is in progress from the 1B waste monitor storage tank.
- Liquid Release Permit # 04-365 was issued to authorize this release.
- After 30 minutes, liquid radwaste discharge radiation monitor channel R-6627 alarms, the monitor indicates off-scale high.
- The Desk RCO reports that all the actions of ONP 1-0510030, *Uncontrolled Release of Radioactive Liquids*, have been completed.
- I&C reports that Channel R-6627 has failed high and will be out of service for at least 60 days.

Which one of the following statements correctly describes the required actions to properly complete the discharge from the 1B Waste Monitor Storage Tank?

A. **Restart the release using permit #04-365 with periodic grab samples in lieu of an OPERABLE radiation monitor.**

B. **Issue a new release permit with independent sample and lineup verifications.**

C. **Restart the release using permit #04-365, after independently verifying the release rate calculations.**

D. **Issue a new release permit using periodic grab samples in lieu of an OPERABLE radiation monitor.**

Distracter Analysis:

A. **Incorrect**: A new release permit with independent verifications is required.
Plausible: Grab samples are acceptable for other release sources.

B. **Correct**:

C. **Incorrect**: A new release permit with is required.
Plausible: If the candidate does not recognize that ONP 1-0510030 completes the old release permit.

D. **Incorrect**: Independent verifications are required.
Plausible: Grab samples are acceptable for other release sources

Level: SRO Exam

KA: APE059G2.1.33(3.4/4.0)

Lesson Plan Objective: 0702401-11

Source: New

Level of knowledge: memory

References:
1. ONP 1-0510030 page 3
2. COP C-200 pages 18-21
3. 1-NOP-06.01 pages 4, 6, 15

K/A APE059G2.1.33(3.4/4.0): Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) IMPORTANCE RO 3.4 SRO 4.0

Objective 0702401-11: Given a set of plant conditions, and a copy of Unit 1 Tech. Spec. and ODCM, identify if the following Unit 1 Radiation Monitoring System related requirements are being challenged:

A. Tech Spec LCO
B. ODCM Controls.

Bank Question: 1131CE ***Answer: D***

1 Pt(s) Unit 2 was conducting a turbine load increase to full power following an outage where the high-pressure turbine had been replaced. Given the following events and conditions:

- Turbine load was 200 MW – increasing slowly at 0.25 MW/min
- Annunciator D-15 (TURBINE VIBRATION ABNORMAL) alarmed
- The turbine supervisory indicated 15 mils turbine vibration and was confirmed to be valid
- Annunciator L-29 (LOSS OF LOAD/LCL PWR DENS CHANNEL TRIP BYPASSED) is dark

Which one of the following statements correctly describes the operator action(s) that are required to respond to this condition?

A. Open the main generator breakers to determine if the main generator causes the vibration – if the vibrations continue, trip the turbine.

B. Raise turbine load away from the destructive resonance point and request an engineering evaluation.

C. Immediately trip the turbine.

D. Trip the reactor and verify turbine trip.

Distracter Analysis: Per 2-ONP-22.02, if the reactor power is > 15% (135 MW) the reactor must be tripped before the turbine is tripped.

A. **Incorrect**: Not the correct immediate action. Also -- must trip the reactor prior to tripping the turbine if L-29 is cleared.
Plausible: If the candidate thinks that a vibration in the main generator could cause a vibration alarm in the turbine.

B. **Incorrect**: Not the correct immediate action.
Plausible: St Lucie has experienced destructive resonance vibration problems when a new turbine rotor has been accelerated past the resonance point – at low speeds. This action may be appropriate IF the vibration occurs during the initial turbine start up per a special engineering procedure.

C. **Incorrect**: Must trip the reactor before tripping the turbine above 15% power.
Plausible: If the candidate does not recognize that power is > 15%.

D. **Correct**:

Level: SRO Exam

KA: SYS 045G2.4.5(4.0/4.0)

Lesson Plan Objective: none

Source: New

Level of knowledge: memory

References:
1. 2-ONP-22.02 pages 5, 8
2. 2-ARP-01-L29
3. 0711303 pages 47-49, 66-67

K/A SYS 045 G2.4.5 (4.0/4.0) Main Turbine Generator - Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6) IMPORTANCE RO 4.0 SRO 4.0

Objective: 0702303-9 Explain the administrative requirements, which affect the turbine operation during all modes of plant operation including any required actions for each of the following.

A. State the administrative limit on Turbine vibration.
B. State the maximum turbine speed allowed prior to breaking vacuum.
C. State when the turbine drain valves must be opened.
D. State the minimum vacuum requirements during turbine operation.

Nuclear Regulatory Commission Senior Reactor Operator Licensing Examination

St Lucie Nuclear Plant

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Date of examination 8/20/2004

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1 Pt(s) A male worker needs to repack a valve in an area that has the following radiological characteristics:

- General area dose rate = 30 mrem/hr
- Airborne contamination concentration = 20 DAC
- The protection factor for a full-face respirator = 100

The worker's present exposure is 830 mrem for the year.

The job will take 4 hours with a mechanic wearing a full-face respirator. It will only take 2 hours if the mechanic does NOT wear the respirator.

Which of the following choices for completing this job would maintain the workers exposure within the Station ALARA requirements?

References Provided:

A. The worker should wear the respirator because the ALARA program requires respirator protection for all work in contaminated areas.

B. The worker should NOT wear the respirator because the dose received will exceed neither NRC nor site annual personnel dose limits.

C. The worker should wear the respirator because the total TEDE dose received will be less than if he does not wear one.

D. The worker should NOT wear the respirator because the total TEDE dose received will be greater than if he wears one.

---

1 Pt(s) Unit 2 was operating at 100% power when ASI approached the Tech Spec limit. Given the following events and conditions:

- Group 5 was inserted to 132"
- One CEA in group 5 slips to 121"
- The CEA Secondary Position Display is selected to slipped CEA

Which one of the following indications will be present for the slipped CEA?

	On RTGB 204 ADS	*On RTGB 204* CEA Secondary Position Display	DCS
A.	**121"**	**121"**	**132"**
B.	**121"**	**132"**	**132"**
C.	**132"**	**121"**	**121"**
D.	**132"**	**132"**	**121"**

---

1 Pt(s) Unit 2 was starting up after a refueling outage. Given the following events and conditions:

- The letdown system is being placed in service in accordance with NOP-02.02 (*Charging and Letdown*)
- An operator incorrectly positions the isolation valve to the purification filter "B" during the initial valve lineup. The required valve position was "*open*" but the valve is actually *closed*. This error is unknown to the control room.
- The operators align CVCS valves in an attempt to establish letdown flow.

What actions will occur and what procedure are the control room operators required to enter?

A. **The letdown system relief valves V-2354 and/or V-2345 open to protect the low pressure letdown piping. Immediately isolate letdown in ONP-02.02 (*Charging and Letdown*)**

B. **V-2516 (CONTAINMENT ISOL VALVE - IC) automatically closes on high D/P across the regenerative HX to protect the low pressure letdown piping. Correct the valve misalignment and restore letdown in accordance with NOP-02.02.**

C. **V-2520 (ION EXCHANGER BYPASS VALVE) automatically opens on high pressure to divert letdown around the ion exchangers and coolant purification filter into the VCT. Correct the valve misalignment and restore letdown in accordance with NOP-02.02.**

D. **PCV-2201Q (PRESSURE CONTROL VALVE) automatically closes on high pressure to isolate the low pressure letdown piping. Immediately isolate letdown in ONP-02.03.**

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1 Pt(s) Unit 1 is at 100% power. Given the following events and conditions:

- PIC-1100Y (PRESSURE) is selected for control
- Pressurizer backup heaters B-1 and B-5 are energized
- PIC-1100Y fails to 100% output

Which one of the following statements correctly describes the:
1. Expected plant response, and
2. Appropriate operator action?

A. **1. Pressurizer spray valves closed, all backup heaters on**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure increase.

B. **1. Pressurizer spray valves closed, proportional heater output to maximum**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure increase.

C. **1. Pressurizer spray valves open, proportional heater output to minimum**
2. Enter ONP-0120035 (*Pressurizer Pressure and Level*) and stop the pressure decrease.

D. **1. Pressurizer spray valves open, all backup heaters on;**
2. Enter 1-EOP-01 (*Standard Post Trip Actions*) and recover the plant.

---

1 Pt(s) Unit 2 was operating at 100% power when a problem occurred with the AFW system. Given the following sequence of events and conditions on 8/21/2004:

- 0100 AFW pump 2C was declared to be out of service due to a governor problem. Repairs are scheduled to be completed in 24 hours.
- 0200 AFW pump 2B motor caught fire when it was started for a surveillance test. Repairs are estimated to take as long as one month.
- 0300 The operators commenced reducing power in preparation to shutdown for repairs.
- 0400 The isolation valve for AFW pump 2A is jammed shut. Repair time is uncertain.
- 0400 Power level = 75%.

What action(s) is required to comply with Tech Specs while the AFW pumps are being repaired?

A. **Stop the shutdown and continue power operation while immediately initiating corrective action to restore at least one AFW pump to operable status. Do not shutdown until one AFW pump has been restored to operability.**

B. **Stop the shutdown for up to 72 hours while repairing the AFW pumps. If repairs to at least one pump are NOT completed within 72 hours, then shutdown to HOT STANDBY within the next six hours and be in HOT SHUTDOWN within the next six hours.**

C. **Continue the shutdown to HOT STANDBY to arrive no later than 0900. Have one AFW pump repaired no later than 0100 on 8/23 and proceed to COLD SHUTDOWN no later than 1300 on 8/23.**

D. **Continue the shutdown to be HOT STANDBY no later than 0900, HOT SHUTDOWN no later than 1500, and COLD SHUTDOWN no later than 1500 on 8/22/2004.**

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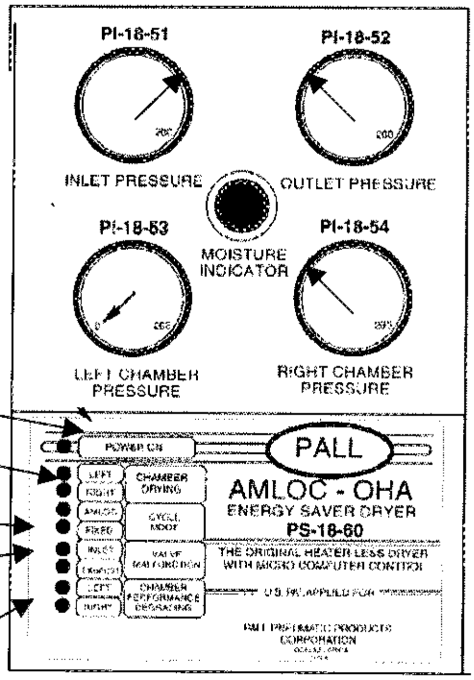
1 Pt(s) Unit 1 was operating at 100% power when instrument air header pressure dropped to 85 psig.

Given the following indications on the Unit 1 instrument air dryer:

Inlet press = 120 psig
Outlet press = 85 psig
Left chamber press = 0 psig
Right chamber press = 85 psig



Power on – lit
Chamber drying:
  Left – not lit
  Right – lit
Cycle mode:
  AMLOC – not lit
  Fixed – lit
Valve malfunction
  Inlet – lit
  Exhaust – not lit
Chamber performance degrading
  Left – lit
  Right – lit

Which one of the following statements correctly describes the cause of the problem and the correct method to temporarily maintain instrument air pressure until the air dryer can be repaired?

A. **The inlet valve on the right dryer chamber has failed closed. Shift to the left dryer chamber and repressurize the air header.**

B. **The exhaust valve on the right dryer chamber has failed closed. Shift to the left dryer chamber and repressurize the air header.**

C. **The inlet valve on the right dryer chamber has failed closed. Bypass around the air dryer and repressurize the air header.**

D. **The exhaust valve on the right dryer chamber has failed closed. Bypass around the air dryer and repressurize the air header.**

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1 Pt(s) Unit 1 was operating at 100% power. Given the following events and conditions:

- Reactor trip occurred 10 minutes ago
- RCS hot and cold leg temperatures are stable and at normal values
- Pressurizer pressure is 1900 psia and lowering
- Pressurizer level is 0%
- Containment pressure is 0.5 psia and rising slowly
- Charging and letdown are responding as expected

Which one of the following break locations would result in these symptoms?

A. **Pressurizer PORV open with a 0.5-inch equivalent break size.**

B. **Reactor vessel vent line between the head vent and the reactor vessel**

C. **Pressurizer spray line between the RCS loop and the spray valve**

D. **Pressurizer level transmitter condensing pot**

---

1 Pt(s) Unit 2 was operating at 100% power. Given the following events and conditions:

- CCW flow to RCP 1A1 motor air cooler has degraded due to flow blockage at the cooler outlet flow-restricting orifice.

Which one of the following conditions will be the FIRST indication of the problem, assuming no operator action?

A. **Increased RCP motor stator temperature**

B. **Increased containment air temperature**

C. **Elevated RCP seal temperature**

D. **Elevated lower RCP motor bearing oil temperature**

---

1 Pt(s) Unit 2 is operating at 100% power steady state with only the 2B charging pump running. Given the following events and conditions:

- Annunciator M-31 (2B CHARGING PUMP TROUBLE) alarms
- The 2B Charging pump has tripped.

Which one of the following statements correctly describes
- the cause of the Charging pump trip, and
- the required operator actions?

**A. The Charging pump has tripped on low oil level.
Immediately isolate letdown.**

**B. The Charging pump has tripped on low oil pressure.
Immediately isolate letdown.**

**C. The Charging pump has tripped on low suction pressure.
Immediately restart the 2B charging pump.**

**D. The Charging pump has tripped on low seal tank level.
Immediately verify that the 2A (standby) charging pump started.**

---

1 Pt(s) Unit 2 was heating up in Mode 4 following a refueling outage preparing to come off shutdown cooling. Given the following events and conditions:

- The shutdown cooling full flow relief valves (V-3666, V-3667) are reported to have been set to lift at 375 psig during the outage.
- RCS pressure = 270 psig
- RCS temperature = 145 °F
- PORVs are selected to LTOP
- PC-1106 (PRESSURE LOW RANGE) fails low

The Operations Manager asks the SRO if the heatup can continue and, if so, what is the least limiting requirement? Which one of the following statements provides the correct answer to his question?

(Note: For the purposes of answering this question, the selections are ranked in order of limitation – A is less limiting than B – which is less limiting than C – which is less limiting than D.)

References Provided:

A. **There are no restrictions on heating up to Mode 3 provided PC-1106 channel is placed in a bypassed condition within one hour.**

B. **The heatup can continue provided we can reach raise RCS temperature above 165°F within the next 8 hours. You will then have 7 days to heat up above 247 °F where there are no further restrictions from this event.**

C. **The heatup can continue provided we can reach raise RCS temperature above 247°F within the in 8 hours. You then can heat up to mode 3 with no further restrictions from this event.**

D. **The heat up cannot continue. We must immediately depressurize to atmospheric pressure and vent the RCS.**

1 Pt(s) Unit 1 was operating at 100% power. Given the following events and conditions:

- V-1402 (PORV) is in NORMAL
- V-1403 (BLOCK VALVE) is OPEN
- V-1404 (PORV) is in NORMAL
- V-1405 (BLOCK VALVE) is CLOSED
- V-1403 failed on thermal overload. V-1403 is full open. The estimated time to repair the MOV is 90 minutes.

Which one of the following statements correctly describes the required Tech Spec actions?

A. **Open V-1405 – remain at 100% power**

B. **Close V-1402 and remove power from the valve – remain at 100% power**

C. **Open V-1405 and remove power from V-1403 – remain at 100% power**

D. **Commence shutdown to hot standby**

1 Pt(s) Unit 1 was operating at 100% power when a reactor trip occurred. Given the following events and conditions.

0200 - The turbine tripped for an unknown reason. The turbine trip failed to cause an automatic reactor trip.
0201 - The manual reactor trip caused the CEAs to drop into the core. During the period of time when an ATWS was occurring, the pressurizer PORV lifted and reseated repeatedly.
0202 – Three CEAs remained stuck out of the core.
0205 – Safety injection was actuated.
0208 – The SRO first assesses the emergency classification.

What is the correct emergency classification for the above events at 0208 when the SRO determines the classification?

References Provided:

A. No classification is required – the events have already been terminated.

B. Alert – the Emergency Coordinator can immediately terminate the event.

C. Site Area Emergency – the Emergency Coordinator can immediately terminate the event.

D. Site Area Emergency

---

1 Pt(s) Unit 1 was at 100% power, responding to a ground on the 1A DC bus. Given the following conditions and events:

- DC bus 1AB is being supplied from DC bus1A.
- Operators are executing NOP-50.01 (*DETERMINATION OF 125V DC BUSES 1A AND 1AB GROUND LOCATION Appendix A*).
- A caution at step 1 of this procedure states:

  CAUTION

  Failure of Battery Charger 1AB amps to rise may indicate the battery charger has not accepted load on 125V DC Bus 1AB. Separation of 125V DC Buses 1A and 1AB may result in loss of Bus 1AB loads.
- The operators fail to heed this caution when opening the 1A to 1AB tie breaker for ground isolation.
- The 1AB DC bus is deenergized when the 1AB battery charger supply breaker trips immediately after opening the 1A-1AB tie breaker.
- The grounds on DC bus 1A have now disappeared.

If battery charger breaker 1AB cannot be re-closed:
1. What is the operability status of the DC buses 1A?
2. What has to be done to promptly reenergize bus 1AB and restore the DC bus alignment to comply with Tech Specs?

A. **1. Bus 1A is operable - only bus loads from 1AB are inoperable**
2. Close the 1A to 1AB bus tie breaker to reenergize the bus.

B. **1. Bus 1A is operable – only bus loads from 1AB are inoperable**
2. Close the 1C to 1AB bus tie breaker

C. **1. Bus 1A is inoperable without bus 1AB being energized**
2. Close the 1A to 1AB bus tie breaker to reenergize the bus

D. **1. Bus 1A is inoperable without bus 1AB being energized**
2. Close the 1C to 1AB bus tie breaker

---

1 Pt(s) Unit 2 is performing a natural circulation cooldown. Given the following events and conditions:

- The unit has been cooling down at 30 °F/hour from hot standby conditions for 5.5 hours.
- Highest CET = 532 °F
- Rep CET = 514 °F
- T_{hot} = 492 °F
- T_{cold} = 467 °F

What is the current saturation pressure of the upper head?

References provided:

A. **500 psia**

B. **630 psia**

C. **775 psia**

D. **900 psia**

---

1 Pt(s) Unit 2 is in Mode 5 on SDC preparing to heatup the RCS. Given the following events and conditions:

- Both Personnel airlock doors are open
- A loss of shutdown cooling occurs and the unit inadvertently heats up to an RCS temperature of 205 °F.

Which one of the following statements correctly describes the status of:
1. containment integrity, and
2. the bases for this requirement

A. **1. Both airlock doors must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR20 during accident conditions.

B. **1. At least one airlock doors must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR100 during accident conditions.

C. **1. At least one airlock door must be maintained closed to reestablish containment integrity.**
2. To limit site boundary radiation dose rates to the limits of 10CFR20 during refueling conditions.

D. **1. Airlock doors may be open provided a means for closure is available within the calculated mean time to core boiling.**
2. To limit site boundary radiation dose rates to the limits of 10CFR100 during refueling conditions.

---

1 Pt(s) Unit 1 is conducting a reactor startup. Given the following events and conditions:

- Criticality has been achieved.
- Group 7 control element assemblies (CEAs) were being withdrawn in 4 step increments to reach the point of adding heat.
- The RCO inadvertently withdraws group 7 CEAs 8 steps.
- All wide range nuclear instrument (WR NI) channel start-up-rate (SUR) pretrips were LIT.
- WR NI channel B SUR peaked at 2.5 decades per minute (DPM) on the recorder. At the same time, the other channels peaked at SUR = +1.5 DPM

Given the following current data:

	% Power	DPM
WR NI Ch A	$5x10^{-4}$	0
WR NI Ch B	$5x10^{-2}$	1
WR NI Ch C	$4.8x10^{-4}$	0.1
WR NI Ch D	$5.2x10^{-4}$	0

Which of the following actions/justifications are required?

A. **Manually trip the reactor because there was a valid SUR greater than 2.0 with no automatic action.**

B. **Continue the startup because wide range SUR protection is not yet in service.**

C. **Place WR NI channel B SUR in bypass because the channel is inoperable.**

D. **Insert the CEAs to the –500 PCM position until the problem with the WR NI channel B instrument is resolved.**

1 Pt(s) Unit 2 was in Mode 6. Given the following events and conditions:

- The refueling contractors dropped a new fuel bundle during an inspection

Which one of the following actions is **NOT** a required immediate action on Unit 2?

A. Notify personnel to evacuate the new fuel area.

B. Start shield building ventilation to filter fuel building exhaust.

C. Notify personnel to evacuate the spent fuel pool area.

D. Ensure the fuel bundle is in a safe condition.

---

1 Pt(s) Unit 1 is Mode 3 preparing to cooldown for an outage. Given the following events and conditions:

- Steam generator (S/G) 1B has a small tube leak and has been isolated.
- RCS T_{-hot} is <510 °F controlled by the steam bypass system
- A loss of offsite power occurs and Emergency Diesel Generator (EDG) 1A failed due to a seized fuel pump.
- Off-site power is restored 15 minutes after the loss first occurred.

What are the <u>earliest</u> actions required by these conditions and the plant license?

References Provided:

A. AFW is inoperable; be in Mode 4 within 6 hours.

B. AFW is inoperable; be in Mode 4 within 12 hours.

C. AC Power Sources are inoperable; fix EDG 1A within 72 hours.

D. AC Power Sources are inoperable; fix EDG 1A within 14 days.

---

1 Pt(s) In order to increase electric output from the site, the feed system engineer proposes to add a steam driven main feedwater pump to substitute for the motor driven pumps at high power levels.

Because this new pump is not described in the Final Safety Analysis Report (FSAR), which of the following activities must FIRST be completed prior to implementing the modification?

A. **A temporary system alteration to address the drawing discrepancies created by the design change.**

B. **A safety evaluation to determine whether the change creates an unreviewed safety question.**

C. **A Technical Specification change to incorporate the new pump requirements.**

D. **An FSAR change to notify the NRC of the impending design change.**

---

1 Pt(s) Unit 1 is performing the initial start-up following core reload. Given the following conditions:

- The reactor has been restarted.
- Criticality occurred at 85 inches on CEA group 7, and subsequently the CEA height was adjusted to 130 inches by boron addition.
- Preparations are in progress to load the main turbine.
- Reactor Engineering has recommended that group 7 CEAs be inserted to 105 inches while loading the turbine continues.

Which response below best describes the basis for reactor engineering's recommendation?

A. To ensure the new fuel assemblies do not exceed their design peak linear heat rate during the initial power ascension.

B. To limit the potential for axial shape index anomalies caused by burn-off of the Gadolinium loaded in the fuel rods at beginning-of-life.

C. To assure enough positive reactivity is available to offset a potential temperature decrease with a positive moderator temperature coefficient.

D. To limit the power peaking in the core in the event a CEA drops during turbine loading.

---

1 Pt(s) A General Emergency has been declared at Unit 1 due to a steam generator tube rupture event with fuel damage and large offsite releases. Site evacuation is in progress. The Operations Support Center (OSC) is preparing a work team to gag two main steam safety valves. The three available maintenance technicians have the following radiation exposure histories:

	Current Annual Dose	Lifetime Dose	Age
Worker #1	600 mrem	6 Rem	24
Worker #2	200 mrem	25 Rem	40
Worker #3	50 mrem	45 Rem	40

The OSC supervisor requests the Emergency Coordinator's (EC) permission to authorize the workers to incur 4500 mrem while performing this job.

Which one of the following statements correctly describes the EC's authority to approve dispatching this work team?

REFERENCES PROVIDED:

A. The EC cannot dispatch the team without site vice president approval to exceed FPL exposure guidelines. Reentry requirements are not applicable at this stage of the emergency.

B. The EC can dispatch the team because no NRC annual exposure limits will be exceeded. The OSC Coordinator can authorize reentry.

C. The EC is required to authorize emergency dose exposure limits before dispatching this team. Only the EC can authorize reentry.

D. The OSC supervisor can dispatch the team without EC approval of emergency limits because emergency exposure is controlled separately from the workers' annual exposure limits. The OSC coordinator can authorize reentry.

---

1 Pt(s) Unit 2 was operating at 100% when an inadvertent ESFAS caused by a human error during I&C testing resulted in both MSIVs closing. Given the following conditions:

- The main steam isolation valves closed.
- The reactor and turbine tripped.
- There were numerous failures of primary and secondary safeties.
- Reactor coolant temperature and pressure peaked at 600°F and 2755 psig, respectively.
- The plant has been stabilized at normal operating temperature and pressure, with no other failures.

What is the earliest report of this incident required to be transmitted to the Nuclear Regulatory Commission?

REFERENCES PROVIDED:

A. An immediate telephone report – not to exceed 1 hour.

B. A 4-hour telephone report.

C. An 8-hour telephone report.

D. A 60-day written report.

---

1 Pt(s) Unit 1 was operating at 100% power when a LOCA occurred. A plant General Emergency has been declared. Given the following events and conditions:

- A LOCA is in progress with an impaired containment
- CET temperature is 714°
- Wind direction is 123°
- Wind speed is 11 mph

Dose (mrem) at:	TEDE (Whole Body)	CDE (Thyroid)
1 mile	4700	23000
2 miles	2300	13000
5 miles	950	5100
10 miles	500	1500

Analyze the data below and choose the correct protective action recommendations for the 5-10 mile zone.

REFERENCE PROVIDED:

A.

Miles	No Action	Evacuate	Shelter
0-2:	none	All	none
2-5:	none	none	NPQ
5-10:	none	none	none

B.

Miles	No Action	Evacuate	Shelter
0-2:	none	All	none
2-5:	none	NPQR	remaining
5-10:	none	none	All

C.

Miles	No Action	Evacuate	Shelter
0-2:	none	All	none
2-5:	none	NPQR	remaining
5-10:	none	NPQR	remaining

D.

Miles	No Action	Evacuate	Shelter
0-2:	none	All	none
2-5:	none	NPQ	none
5-10:	none	NPQ	remaining

---

1 Pt(s) Unit 1 is operating at full power. Given the following events and conditions:

- A radioactive liquid release is in progress from the 1B waste monitor storage tank.
- Liquid Release Permit # 04-365 was issued to authorize this release.
- After 30 minutes, liquid radwaste discharge radiation monitor channel R-6627 alarms, the monitor indicates off-scale high.
- The Desk RCO reports that all the actions of ONP 1-0510030, *Uncontrolled Release of Radioactive Liquids*, have been completed.
- I&C reports that Channel R-6627 has failed high and will be out of service for at least 60 days.

Which one of the following statements correctly describes the required actions to properly complete the discharge from the 1B Waste Monitor Storage Tank?

A. **Restart the release using permit #04-365 with periodic grab samples in lieu of an OPERABLE radiation monitor.**

B. **Issue a new release permit with independent sample and lineup verifications.**

C. **Restart the release using permit #04-365, after independently verifying the release rate calculations.**

D. **Issue a new release permit using periodic grab samples in lieu of an OPERABLE radiation monitor.**

1 Pt(s)

Unit 2 was conducting a turbine load increase to full power following an outage where the high-pressure turbine had been replaced. Given the following events and conditions:

- Turbine load was 200 MW – increasing slowly at 0.25 MW/min
- Annunciator D-15 (TURBINE VIBRATION ABNORMAL) alarmed
- The turbine supervisory indicated 15 mils turbine vibration and was confirmed to be valid
- Annunciator L-29 (LOSS OF LOAD/LCL PWR DENS CHANNEL TRIP BYPASSED) is dark

Which one of the following statements correctly describes the operator action(s) that are required to respond to this condition?

A. **Open the main generator breakers to determine if the main generator causes the vibration – if the vibrations continue, trip the turbine.**

B. **Raise turbine load away from the destructive resonance point and request an engineering evaluation.**

C. **Immediately trip the turbine.**

D. **Trip the reactor and verify turbine trip.**

SRO Exam References

SRO Exam References:

ADM-05.01 Appendix C page 29
AP 0010721
EPIP-01 (Classification of Emergency)
EPIP-02 (Duties and Responsibilities of the Emergency Coordinator)
EPIP-08 Attachments 1A & 2
HPP-30 Appendixes 1, 6, & 11
Unit 2 ONP-0120039 Figure 3 Unit 1 Tech Specs 3.4.13
Unit 1 Tech Spec 3.7.1.2
Unit 1 Tech Spec 3.8.1.1
Unit 2 Tech Specs 3.4.9.3

ADMs

APPENDIX C
TEDE ALARA EVALUATION FORM
(Page 1 of 2)

RWP Number ____________________

Job Description ____

____

A. Respiratory Device ____________________ **Protection Factor** ________________

B. TEDE ALARA Estimate Evaluation

1. Exposure Estimate with Respiratory Device

   a. $\dfrac{\_\_\_\_\_\_\ \text{rem/hr} \times \_\_\_\_\_\_\ \text{hrs} \times 1.2^{*}}{\text{(area dose rate)} \quad \text{(time w/o respirator)}}$ = _______ rem

   b. $\dfrac{0.0025\ \text{rem/DAC} \times \_\_\_\_\_\_\ \text{hrs} \times \_\_\_\_\_\_\ \text{DAC}}{\_\_\_\_\_\_\ \text{PF (respirator protection factor)}}$ = _______ rem

   c. Add lines a and b _______ total rem

2. Exposure Estimate without Respiratory Device

   a. $\dfrac{\_\_\_\_\_\_\ \text{rem/hr} \times \_\_\_\_\_\_\ \text{hrs}}{\text{(area dose rate)} \quad \text{(time w/o respirator)}}$ = _______ rem

   * "time w/respirator" normally 1.2 times "time w/o respirator"

   b. 0.0025 rem/DAC x _______ hrs x _______ DAC = _______ rem

   c. Add lines a and b _______ total rem

C. Exposure Savings / Costs

1. ________ rem Exposure estimate with respiratory (line 1.c)
2. - ________ rem (subtract) Exposure estimate without respiratory (line 2.c)
3. ________ rem Exposure savings (+) or cost (-) with respiratory

D. Respiratory device use approved? ☐ Yes ☐ No

APs

FPL	**ST. LUCIE PLANT** **ADMINISTRATIVE PROCEDURE** QUALITY RELATED	Procedure No. **0010721** Current Revision No. **52** Effective Date **03/26/04**

Title:

NRC REQUIRED NON-ROUTINE NOTIFICATIONS AND REPORTS

Responsible Department: **OPERATIONS**

REVISION SUMMARY:

Revision 52 - Incorporated PCR 04-0782 to change SAFETY RELATED to QUALITY RELATED. (J. S. Napier, 03/04/04/)

AND

Incorporated PCR 04-0273 for CR 04-0044 to revise reporting guidance for 10CFR50.72 reports. (G. R. Madden, 02/20/04)

Revision 51A - Incorporated PCR 03-2028 to correct procedure number reference. (Bonnie Wooldridge, 07/23/03)

Revision 51 - Incorporated PCR 03-3339 to remove special reporting requirements. (Terri Taylor, 12/04/03)

Revision 50A - Incorporated PCR 03-2357 to delete requirement for NPS to make state notification on sea turtle stranding. (N. Whiting, 08/18/03)

Revision 50 - Incorporated PCR 03-0412 for CR 03-0390 to enhance existing guidance to Operations concerning 8 hour reports for "major loss of emergency preparedness capability." (K. W. Frehafer, 03/13/03)

Revision 49 - Incorporated PCR 03-0214 to remove Health Physics Training Building (B-11) Lobby. (T. Taylor, 02/07/03)

Revision	FRG Review Date	Approved By	Approval Date
0	07/31/75	K. N. Harris Plant General Manager	07/31/75
Revision	**FRG Review Date**	**Approved By**	**Approval Date**
52	02/20/04	G. L. Johnston Plant General Manager	02/20/04
		N/A Designated Approver	
		N/A Designated Approver (Minor Correction)	

S___OPS	
DATE	
DOCT	PROCEDURE
DOCN	0010721
SYS	
COM	COMPLETED
ITM	52

TABLE OF CONTENTS

1.0 TITLE

NRC REQUIRED NON-ROUTINE NOTIFICATIONS AND REPORTS

2.0 REVIEW AND APPROVAL

See cover page

3.0 SCOPE

3.1 Purpose

This procedure provides instructions for making notifications and reporting and recording Safety Limit Violations, Licensee Event Reports, Special Reports, Radiological Events and Security Events. Instructions concerning recovery from abnormal conditions are located in the appropriate Off-Normal Operating Procedure.

3.2 Authority

1. Units 1 and 2 Operating Licenses
2. Appendix A, "Technical Specifications."
3. Title 10 of Code of Federal Regulations.
4. Title 29 of Code of Federal Regulations
5. Title 49 of Code of Federal Regulations

3.3 Definitions:

1. Non-Routine Reports

   Reports of abnormal conditions of occurrences as required by the Technical Specifications, the Off-site Dose Calculation Manual (ODCM), the Code of Federal Regulations (CFR), the Operating License, and the Final Updated Safety Analysis Report (FSAR).

2. Immediate Notification

   Notification to the NRC via Emergency Notification System (ENS) or telephone, within a specified time frame (usually hours), as defined by the Technical Specifications, the ODCM, the CFR and/or the FSAR.

3.3 Definitions: (continued)

3. Licensee Event Report

A 60-day written report to NRC on events and conditions as defined by 10 CFR 50.73.

4. Safety Limit

Limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity (see Section 2.1 of Appendix A, "Technical Specifications").

5. Special Reports

Is defined in Technical Specifications Section 6.9.2 as, "Special reports shall be submitted to the NRC within the time period specified for each report".

6. Posting of Notices to Workers

As defined in 10 CFR 19, 10 CFR 21, QI 16-PR/PSL-4, Subsection 223B of the Atomic Energy Act of 1954, Appendix A to 29 CFR 24, and Sections 206 and 211 of the Energy Reorganization Act of 1974 and, shown in Appendices E, F, H, I and J.

A. 10 CFR 19.11 requires posting of any notice of violation involving radiological working conditions, proposed imposition of civil penalty, or order issued pursuant to Subpart B of 10 CFR 2 and any response from FPL.

B. Documents to be posted in accordance with the above paragraph shall be posted within 2 working days of receipt from the NRC.

C. FPL responses to the above, if any, shall be posted within 2 working days of dispatch.

D. Documents shall remain posted for at least 5 working days or until action correcting the violation has been completed which ever is later.

3.3 Definitions: (continued)

7. Valid Engineered Safety Feature (ESF) Actuations

Valid ESF actuations are those actuations that result from valid signals or from intentional manual initiation, unless it is part of a preplanned test. Valid signals are those signals that are initiated in response to actual plant conditions or parameters satisfying the requirements for ESF initiation. Refer to Nureg-1022, Rev. 2, Section 3.2.6 for further guidance.

8. Invalid Engineered Safety Feature (ESF) Actuations

Invalid actuations are by definition those that do not meet the criteria for being valid. Thus, invalid actuations include actuations that are not intentional manual actuations. Invalid actuations include instances where instrument drift, spurious signals, human error or other invalid signals caused actuation of the ESF (e.g., jarring a cabinet, an error in use of jumpers or lifted leads, an error in actuation of switches or controls, equipment failure or radio frequency interference). Refer to Nureg-1022, Rev. 2, Section 3.2.6 for further guidance.

3.4 10 CFR 21 Items

These reports shall be handled in accordance with QI 16-PR/PSL-4.

3.5 10 CFR 30.9, 40.9, 50.9, 55.9, 70.9, & 71.6a.

1. Information provided to the NRC as an applicant for an operator's or facility license or as a licensee or information required by statue or by NRC regulations, orders or license conditions to be maintained by the applicant or licensee shall be complete and accurate in all material respects.

2. The Regional Administrator shall be notified when it is identified that incomplete or inaccurate information has been provided to the NRC. This requirement is not applicable to information which is already required to be provided to the NRC by other reporting or updating requirements.

4.0 PRECAUTIONS

4.1 Failure to make required federal reports is a violation of the Facility License.

4.2 Time is of the essence as some required notifications are required to be made immediately.

5.0 RESPONSIBILITIES

5.1 St. Lucie Plant supervisors should be familiar with the NRC required non-routine notifications and reports in their areas of responsibility as defined in this procedure.

5.2 Any individual aware of an incident, event, condition or activity that could require a non-routine report shall immediately inform his/her foreman or supervisor. A Condition Report shall be written to ensure that the incident, event, condition or activity is properly documented.

5.3 Any foreman or supervisor knowing of a possible non-routine report item shall immediately inform the cognizant department head and the Nuclear Plant Supervisor. The Foreman or Supervisor shall ensure that a Condition Report is written documenting the incident, event, condition or activity.

5.4 The SM is responsible for determining reportability for those events that do not require analysis, evaluation and/or interpretation from Engineering and the Facility Review Group (FRG). Reportable events and potential reportable events shall be documented in a Condition Report.

5.5 For those potential reportable events, that require an analysis and evaluation from Engineering, a Condition Report shall be written for Engineering to analyze and evaluate.

5.6 For events that require analysis and evaluation from Engineering, the FRG is responsible for determining reportability (through approval or disapproval of the Condition Report Disposition).

1. Degraded Condition or Unanalyzed Condition
2. Event That Could Have Prevented Fulfillment of a Safety Function
3. Common Cause or Condition Resulting in Independent Trains or Channels Becoming Inoperable

5.7 The Facility Review Group has the ultimate responsibility for interpretation of reportability per this procedure.

5.8 The Facility Review Group is responsible for recommending and approving retraction of reports made.

5.9 The Nuclear Plant Supervisor (or designee) is responsible for making required report notifications.

5.10 The Nuclear Plant Supervisor shall inform the Plant General Manager if necessary. Section 8.4 specifies the requirements for these notifications.

5.11 The Plant General Manager is responsible to ensure that all requirements of the facilities license and all records and notifications as per Section 7.0 are met.

5.12 The Licensing Manager shall ensure the posting required by Section 3.3.6 of this procedure are maintained at the locations specified in Section 8.10.

5.13 The Nuclear Plant Supervisor is responsible for ensuring that immediate reports to the NRC are made within the allowable time frame.

1. The NRC Duty Officer will have a form similar to Appendix G from which he may ask questions. It may be helpful, but is not required, to complete this attachment prior to making Event Classification type and 1-hour NRC ENS notifications. Appendix G should be completed prior to the NRC ENS notifications for 4-hour Non-Emergency 10 CFR 5072(b)(2) and 8-hour Non-Emergency 10 CFR 50.72(b)(3) failures or events. /R52

2. More than one failure or event may and should be reported in a single ENS notification provided (1) the failures or events are related (i.e., they have the same general cause or consequences) and (2) they occurred during a single activity (e.g., during or as a result of a reactor trip) over a reasonably short time (e.g., within 4 hours or 8 hours for ENS notifications. /R52

3. Unrelated failures or vents should be reported as separate ENS notifications (with separate copies of Appendix G) to be given unique ENS numbers by the NRC. However, multiple ENS notifications may be addressed in a single NRC ENS phone call. /R52

5.14 The SM (or designee) should notify the Work Control Group and Licensing if equipment is declared out of service that could require a Special Report to the NRC. Specifically, if the Technical Specifications require a Special Report be submitted if equipment is out of service greater than a specified time, the SM (or designee) should inform the Work Control Group such that appropriate priority is assigned for repairs. The SM (or designee) should also inform Licensing such that they are aware of the potential reportability.

5.15 The Licensing Manager is responsible for posting copies of any violation involving radiological work conditions, any notice of proposed imposition of civil penalties, or NRC order issued pursuant to 10 CFR 2.202 and the FPL responses to any of the above documents. Posting is required within 2 working days of receipt or dispatch by FPL.

5.16 The Land Utilization Group shall provide the SM with the completed Stranding form, of AP 0005762, for any event that is required to be reported to the Florida Fish and Wildlife Conservation Commission. The SM shall then make the required 50.72 notification.

6.0 REFERENCES

NOTE

One or more of the following symbols may be used in this procedure:

§ Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, Operating Experience, License Renewal, etc. and shall NOT be revised without Facility Review Group review and Plant General Manager approval.

¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.

Ψ Indicates a step that requires a sign off on an attachment.

6.1 Regulatory Guide 10.1, Compilation of Reporting Requirements for Persons Subject to NRC Regulations

6.2 Appendix A, Technical Specifications

6.3 NUREG-1022, Rev. 2, Event Reporting Guidelines 10 CFR 50.72 and 50.73

6.4 Letter JNS-EP-87-080, Definition of Significant Degradation of Alert and Notification System

6.5 C-200, Offsite Dose Calculation Manual (ODCM)

6.6 Title 10 Code of Federal Regulations

6.7 EPIP-01, Classification of Emergencies

6.8 QI 16-PR/PSL-4, Evaluation and Reporting Defects and Failures to Comply for Substantial Safety Hazards in accordance with 10 CFR Part 21

6.9 10 CFR 19.11(c), Posting Notices to Workers

6.10 Section 206 of the Energy Reorganization Act of 1974

6.11 Section 223B of the Atomic Energy Act of 1954

6.12 Section 211 of the Energy Reorganization Act of 1974

6.13 NP-808, Evaluation and Reporting Defects and Failures to Comply for substantial Safety Hazards in accordance with 10 CFR 21

6.14 CR 00-0061

6.15 NP-303, Notification of Chief Nuclear Officer

6.16 NP-803, Company Nuclear Review Board

6.17 CR 00-0618

6.18 10 CFR 19.11.a, Posting of notices to workers

6.19 CR 02-0505

6.20 NRC NRR Office Instruction LIC-100, Control of Licensing Bases for Operating Reactors, Revision 00-a

7.0 RECORDS AND NOTIFICATIONS

7.1 Notifications and reports shall be made in accordance with Appendix A, Appendix B, Appendix C and Appendix D of this procedure.

7.2 Originals or copies of all plant generated telephone conference memoranda of non-routine report items, follow-up telegrams or facsimile transmission reports, letter reports, supplementary reports, etc., should be forwarded to Document Control.

8.0 INSTRUCTIONS

8.1 Events and conditions requiring non-routine NRC and other notifications or reports are listed in Appendix A, Appendix B and Appendix C of this procedure.

NOTE

Certain reportable events may require analysis or evaluation to determine reportability. In the event one of the following reportable conditions is evaluated to exist by Engineering, FRG review and approval of the condition is required to establish reportability:

- Degraded condition or unanalyzed condition
- Event that could have prevented fulfillment of a safety function
- Common cause or condition resulting in independent trains or channels becoming inoperable

8.2 The first individual to become aware of an event, condition or activity which could require a non-routine NRC notification or report shall immediately inform his/her foreman or supervisor.

8.3 The cognizant foreman or supervisor shall immediately notify the applicable department head and the Nuclear Plant Supervisor and submit a Condition Report on the event or condition.

8.4 The SM (or designee) shall determine if the event or condition meets the requirement for a non-routine NRC notification, using Appendix A, Appendix B, and Appendix C of this procedure, or if the event or condition requires additional analysis to determine reportability.

1. If the event or condition is determined to meet the requirement for a non-routine NRC notification, Then the SM shall ensure that the required notification is made within the allowable time frame.

2. If it is determined that the event or condition requires additional analysis to determine reportability, Then the Condition Report should be designated as "Potentially Reportable" and processed per NAP-400, Condition Reports.

3. If the event or condition is determined to be NOT reportable, Then no further action is required by this procedure.

8.5 If the event is of a complex nature, or if an unplanned reactor trip has occurred, the SM should ensure that appropriate personnel interviews are performed, relevant documents, information/data are collected in order to obtain a full understanding of the event. If the event results in the formation of an Event Response Team (ERT), Then the responsibilities for event reconstruction should be turned over to the ERT when available.

NOTE
For some events, the NRC Duty Officer will request an open line of continuous communications be maintained to the NRC Operations Center. This communications channel shall be closed only when authorized by the NRC.

8.6 Prior to contacting the NRC, the Event Notification Worksheet (Appendix G) should be completed to the maximum extent possible. This completed form should then be followed for information to transmit to the NRC.

1. Check all applicable reporting criteria on the Event Notification Worksheet (Appendix G) when **related failures or events** are reported in a single ENS notification (e.g., AFAS actuates after or as a result of a reactor trip). See Section 5.13 for guidance on reporting multiple failures or events in a single ENS notification. /R52

2. Multiple ENS notifications may be addressed in a single NRC ENS phone call. However, **unrelated failures or events** should be reported as separate ENS notifications with separate Event Notification Worksheets (Appendix G) and should be given unique ENS numbers by the NRC. /R52

8.7 The SM shall refer to Operation Policy OPS-105 for any additional notification requirements.

8.8 Licensee Event Reports will be prepared by the Site Licensing Department in accordance with the guidelines of NUREG 1022.

8.9 All non-routine NRC written reports must be reviewed by the Facility Review Group prior to submittal to the NRC. All non-routine NRC reports should be submitted to the Site Licensing Department Staff at least (5) days prior to the NRC due date. A copy of each report shall be sent to document control for retention in the plant files in accordance with QI-17-PSL-1. The Nuclear Assurance Department Staff shall ensure CNRB Review.

8.10 See below:

1. Forms similar to Appendices E, F, H, I, J and Form NRC-3 shall be maintained in accordance with paragraph 5.12 in the following locations:

   A. East Security Building

   B. North Security Building

   C. Nuclear Training Center Lobby

   D. Unit 1 RAB Entrance

   E. Unit 2 RAB Entrance

2. NRC Notices of Violation involving radiological working conditions, proposed imposition of civil penalty or order issued pursuant to 10 CFR 2, Subpart B and their required responses shall be posted by the Licensing Manager. These documents shall be posted within 2 working days after receipt of the documents from the Commission; FPL's response, if any, shall be posted within 2 working days after dispatch. Such documents shall remain posted for a minimum of 5 working days or until action correcting the violation has been completed, whichever is later. These documents shall be posted in the following locations:

   A. Unit 1 RAB Entrance

   B. Unit 2 RAB Entrance

   C. Any temporary RCA entrance for the duration of its use.

8.11 See below:

1. Certain reportable events may require analysis or evaluation to determine reportability. In the event one of the following reportable conditions is evaluated to exist by Engineering, FRG review and approval of the condition is required to establish reportability:

   - Degraded condition or unanalyzed condition
   - Event that could have prevented fulfillment of a safety function
   - Common cause or condition resulting in independent trains or channels becoming inoperable

2. If an event has been reported to the NRC but subsequent evaluation of the condition concludes that the event was not reportable, the retraction notification shall be reviewed and recommended for approval by the FRG prior to notifying the NRC of the retraction.

8.12 Additional Guidance for Major Loss of Emergency Preparedness Capability

A major loss of emergency preparedness capability includes those events that would significantly impair the safety assessment capability or would significantly impair the fulfillment of the approved emergency plan. Loss of capability with appropriate compensatory measures would not be considered a major loss.

If an emergency assessment structure, system, or component is governed by a Technical Specification (TS), ODCM, or UFSAR Chapter 13 requirement, and operation is within LCO and applicable ACTION statement requirements, then there is not a major loss of assessment capability (e.g., accident monitoring instrumentation, radiation monitors, meteorological tower, etc.). If the governing document requires NRC notification or plant shutdown because of unavailable assessment instrumentation, then that condition would be considered a major loss of assessment capability and shall be reported as such.

An unplanned loss of SPDS output for greater than one hour, or the loss of SPDS output concurrent with the loss of other major emergency assessment equipment (i.e., loss of plant instrumentation such that safety function(s) can not be verified) shall be reported as a major loss of assessment and communication capability.

Loss of the meteorological tower is not reportable as long as the National Weather Service is available to provide required information.

Upon notification that the 12 month cumulative running siren availability decreased to less than 90 percent, or the siren availability during any 2 week period is determined to be less than 75 percent, a NRC notification will be made as a major loss of communication capability.

The loss of all primary and backup communication channels to a state or local government agency or emergency response facility [Control Room, Technical Support Center (TSC), Operational Support Center (OSC), Emergency Operations Facility (EOF)] meets the EPIP-01 requirements for the declaration of an unusual event. The communication channels include dedicated telephone communication links (i.e., Florida State Warning Point), commercial telephone lines, and offsite emergency radio communication system.

If a dedicated NRC primary communication channel is lost (i.e., either the ENS, HPN, or ERDS data link) for any amount of time, then the event is considered to be a major loss of emergency communication capability.

In addition to the off-site communications above, the total loss of the in-plant paging, and in-plant radio systems required for safe plant operation would be reportable as a major loss of emergency communication capability.

APPENDIX A
10 CFR 50.72 AND 50.73 REPORTS
(Page 1 of 6)

Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
EMERGENCY CLASS	Immediately after notification of State and local agencies, but not later than 1 hour after declaration of emergency class defined in EPIP-01 {(50.72(a)(1), (a)(2), (a)(3) and (a)(4)}		Note - Although not specifically mentioned in 10 CFR 50.73, many emergency class events involve reportable events involve reportable situations.	3.1.1
TECHNICAL SPECIFICATIONS (TS):				
Plant shutdown required by TS	Initiations of shutdown required by TS {50.72(b)(2)(i)} (4 hour report)		Completion of shutdown required by TS {50.73(a)(2)(i)(A)}	3.2.1
TS prohibited operations or conditions			Operation or condition prohibited by TS {50.73(a)(2)(i)(B)} except when: (1) The Technical Specification is administrative in nature; (2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.	3.2.2
TS deviation authorized by 50.54(x)	Deviation from TS authorized by 50.54(x) {50.72(b)(1)} (1 hour report)		Criterion {50.73(a)(2)(i)(C)} same as ENS 1 hour	3.2.3

APPENDIX A
10 CFR 50.72 AND 50.73 REPORTS
(Page 2 of 6)

Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
DEGRADED OR UNANALYZED				
Plant, including its principal safety barriers, seriously degraded		Any condition that results in the plant, including principal safety barriers, being seriously degraded {50.72(b)(3)(ii)(A)}	Condition of plant, including principal safety barriers, seriously degraded {50.73(a)(2)(ii)(A)}	3.2.4
Plant in an unanalyzed condition that significantly degrades plant safety		Any condition that results in an unanalyzed condition that significantly degrades plant safety {50.72(b)(3)(ii)(B)}	Unanalyzed condition that significantly degrades plant safety {50.73(a)(2)(ii)(B)}	3.2.4
EXTERNAL THREAT TO PLANT SAFETY			Any natural phenomenon or other external condition that poses an actual threat to the safety of the plant or significantly hampers site personnel in performance of duties necessary for its safe operation {50.73(a)(2)(iii)}	3.2.5

APPENDIX A
10 CFR 50.72 AND 50.73 REPORTS
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Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
EMERGENCY SYSTEM ACTUATION	Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal {50.72(b)(2)(iv)(A)} or results in actuation of the reactor protection system (RPS) when the reactor is critical {50.72(b)(2)(iv)(B)}, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operations. (4 hour report)	Any event or condition that results in valid actuation of any of the: reactor protection system (RPS), general containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs); emergency core cooling systems (ECCS); auxiliary or emergency feedwater system; containment spray and fan cooler systems; and emergency diesel generators (EDGs), except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. {50.72(b)(3)(iv)(A)} NOTE: Invalid actuations (other than actuation of the reactor protection system (RPS) when the reactor is critical) are not reportable under 10 CFR 50.72.	Any event or condition that resulted in manual or automatic actuation of the: reactor protection system (RPS), general containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs); emergency core cooling systems (ECCS); auxiliary or emergency feedwater system; containment spray and fan cooler systems; and emergency diesel generators (EDGs), except when: (1) actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or (2) the actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had been already completed. {50.73(a)(2)(iv)(A)} NOTE: Invalid actuations (other than actuation of the reactor protection system (RPS) when the reactor is critical), may be reported to the NRC Operations Center via telephone notification within 60 days after discovery of the event instead of submitting a written LER.	3.2.6

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10 CFR 50.72 AND 50.73 REPORTS
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Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
EVENT OR CONDITION THAT COULD HAVE PREVENTED FULFILLMENT OF A SAFETY FUNCTION		Event or condition at the time of discovery that could have prevented fulfillment of safety function of system needed for shutdown of the reactor, maintenance of a safe shutdown condition, residual heat removal (RHR), control of release of radioactive materials, or mitigation of the consequences of an accident {50.72(b)(3)(v)} Individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function. {50.72(b)(3)(vi)}	Except for tense, criterion {50.73(a)(2)(v)} same as ENS 8 hours. Need not report individual component failures under this paragraph if redundant equipment in same system was operable and available {50.73(a)(2)(vi)}	3.2.7
COMMON CAUSE OR CONDITION RESULTING IN INDEPENDENT TRAINS OR CHANNELS BECOMING INOPERABLE			Single cause or condition caused inoperability of at least one independent train or channel in multiple systems or two independent trains and channels in a single system designed for safe shutdown, RHR, radiation release control, or accident mitigation {5.73(a)(2)(vii)}	3.2.8

APPENDIX A
10 CFR 50.72 AND 50.73 REPORTS
(Page 5 of 6)

Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
RADIOACTIVE RELEASES				
Airborne radioactivity releases			Airborne radioactivity released to an unrestricted area exceeds 20 times the concentration specified in 10 CFR 20, Appendix B, Table 2, averaged over 1 hour 50.73(a)(2)(viii)(A)}	3.2.9
Liquid effluent releases			Liquid effluent released to an unrestricted area exceeds 20 times the concentration specified in 10 CFR 20, Appendix B, Table 2, for all radionuclides except tritium and dissolved noble gases, averaged over 1 hour {50.73(a)(2)(viii)(B)}	3.2.9
INTERNAL THREAT TO PLANT SAFETY			Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases. {50.73(a)(2)(x)}	3.2.10

APPENDIX A
10 CFR 50.72 AND 50.73 REPORTS
(Page 6 of 6)

Event or Condition	Immediate ENS notifications and notifications to be made as soon as practical (within 1 or 4 hours)	ENS notification as soon as practical and in all cases within 8 hours and other NRC notifications	60-day LER and Special Reports	NUREG-1022 Section
LOSS OF EMERGENCY ASSESSMENT, OFFSITE RESPONSE, OR COMMUNICATIONS CAPABILITY		Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system). {50.72(b)(3)(xiii)} See Section 8.12 for additional guidance.		3.2.13
TRANSPORT OF CONTAMINATED PERSON TO OFFSITE MEDICAL FACILITY		A radioactively contaminated person is transported to an offsite medical facility {50.72(b)(3)(xii)}		3.2.11
NEWS RELEASE/OTHER GOVERNMENT NOTIFICATIONS	A news release is planned or other government agencies have been or will be notified of an event related to the health and safety of the public or onsite personnel, or the protection of the environment {50.72(b)(2)(xi)} (4 hour)			3.2.12
SINGLE CAUSE THAT COULD HAVE PREVENTED SAFETY FUNCTIONS OF TRAINS OR CHANNELS OF DIFFERENT SYSTEMS			Any event or condition as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to: shut down and maintain reactor in a safe shutdown condition; remove residual heat; control the release of radioactive material; or mitigate the consequences of an accident. {50.73(a)(2)(ix)(A)}	3.2.14

APPENDIX B
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(Page 1 of 11)

EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
SAFETY LIMIT VIOLATIONS	Violation of any Safety Limit specified by TS 2.0. Note: Critical operation of the unit shall not be resumed until authorized by the NRC {50.36(c)(1), (6) and (7)} NOTE: - Immediate NRC notification required - Notify the Chief Nuclear Officer within 24 hours - Notify the CNRB within 24 hours	Submit a Safety Limit Violation Report. (Technical Specification 6.7.1) Note: -Report shall be submitted to the FRG, Chief Nuclear Officer, and CNRB within 14 days of the violation. Critical operation shall not be resumed until authorized by the NRC	Unit 1 Technical Specification Section 6.7.1 Unit 2 Technical Specification Section 6.7.1 10 CFR 50.36(c)(1)(6) and (7)
RADIOACTIVE MATERIALS TRANSPORTATION	Removable surface contamination on non-exclusive use vehicles in excess of the limits in 49 CFR 173.443(b) {10 CFR 20.1906(d)(1)} Removable surface contamination on exclusive use vehicles in excess of the limits in 49 CFR 173.443(b) {10 CFR 20.1906(d)(1)} External radiation levels at any point on the surface of a package in excess of the limits of 71.47(a) and 71.47(b) delivered by a non-exclusive use or exclusive use transport vehicle. NOTE: Immediate notification required		49 CFR 173.443(a)(2) {10 CFR 20.1906(d)(1)} 49 CFR 173.443(b) {10 CFR 20.1906(d)(1)} 71.47(a) and 71.47(b)

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
BYPRODUCT, SOURCE AND SPECIAL NUCLEAR MATERIALS EXPOSURE AND RELEASE	Event that causes of threatens to cause exposure in excess of the limits in 20.2202(a) {20.2202(a)} Release of radioactive material in concentrations averaged over a 24 hour period, would exceed five times the annual limit on intake in 20.2202(a) {20.2202(a)} NOTE: this is a 24 hour ENS notification	Criterion {20.2202(a)} same as ENS immediate notification Criterion {20.2202(a)} same as ENS immediate notification	10 CFR 20.22002
LICENSED MATERIALS EXPOSURES AND RELEASES	Any event involving licensed material possessed by FPL that may have caused or threatens to cause external exposure or intake in excess of the limits in 20.2202(b) {20.2202(b)} NOTE: This is a 24 hour ENS notification	Criterion {20.2202(a)} same as ENS 24 hour notification	10 CFR 20.2202

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
RADIOLOGICAL EVENT		Any incident for which notification is required by 20.2202 {20.2203(a)} Doses in excess of the limits in 20.1201, or 20.1207, or 20.1208, or 20.1301, or any applicable limit as required by the Operating Licenses {20.2203(a)} Levels of radiation or concentration of radioactive material in a restricted area in excess of an applicable limit of the Operating Licenses {20.2203(a)} Levels of radiation or concentration of radioactive material in an unrestricted area in excess of any applicable limit set forth in 10 CFR Part 20 or in the Operating Licenses {20.2203(a)} Levels of radiation or releases of radioactive material in excess of limits in 40 CFR Part 190 or in excess of license conditions related to compliance with 40 CFR Part 190 {20.2203(a)}	10 CFR 20.2202 10 CFR 20.2203

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OTHER REGULATORY REPORTS
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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
THEFT OR LOSS OF LICENSED MATERIAL	Any lost, stolen, or missing licensed material in a quantity equal to or greater than 1000 times the quantity specified in Appendix C to 20.1001 - 20.24.01 under circumstances that it appears that an exposure could result to persons in unrestricted areas {20.2201(a)} NOTE: Immediate notification required Any lost, stolen, or missing licensed material in a quantity equal to or greater than 10 times the quantity specified in Appendix C to 20.1001-20.2401 {20.2201(a)} NOTE: This report is required to be made by ENS phone 30 days after FPL becomes aware that the material is missing if the material is still missing.	Criterion {20.2201(a)} same as ENS 24 hour notification {20.2201(b)}	10 CFR 20.2201

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
SAFEGUARDS EVENTS (SECURITY)	Reports are made in accordance with Security Procedure 0006125, "Reporting of Safeguards Events," {10 CFR 73.71(a) and 10 CFR 73.71(b)}	Criterion {10 CFR 73.71(d)} same as ENS 24 hour notification	10 CFR 73.71
INCOMPLETE AND INACCURATE INFORMATION	Identification of information having been provided to the NRC which is incomplete or inaccurate in some material respect and being identified as having a significant implication for public health and safety or common defense and security. {10 CFR 50.9(b)} NOTE: Notification is required to be made to the Regional Administrator, NRC Region II within two working days of identifying the information.		10 CFR 50.9

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
FAILURE TO COMPLY OR DEFECT	Identification of a defect or a failure to comply associated with a substantial safety hazard (10 CFR 21.21(d)(3)(i)) NOTE: Initial notification is to be made by facsimile (preferred method) or by telephone to the NRC Operations Center within two days following receipt of information under 10 CFR 21.21(a)(1) {QI 16-PR/PSL-4, "Evaluation and Reporting Defects and Failures to Comply for Substantial Safety Hazards," in accordance with 10 CFR Part 21}	Criterion {10 CFR 21.21(d)(3)(ii)} same as ENS 2-day notification Evaluate deviations and failures to comply associated with substantial safety hazards as soon as practical, and in all cases within 60 days of discovery. If an evaluation of an identified deviation or failure to comply potentially associated with a substantial safety hazard, submit an interim report. {10 CFR 21.21(a)(1) and (a)(2)} NOTE: Notification of the potential substantial safety hazard is required to be made if the evaluation can not be completed within 60 days from discovery. {QI 16-PR/PSL-4, "Evaluation and Reporting Defects and Failures to Comply for Substantial Safety Hazards," in accordance with 10 CFR Part 21}	10 CFR 21

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
FATALITY OR HOSPITALIZATION	Any event or situation resulting in fatality or the hospitalization of three or more employees as a result of a work related incident {29 CFR 1904.8} NOTE: This phone report is to be made to the Occupational Safety and Health Administration (OSHA) within 8 hours of the event or situation (305-424-0242 or 800-321-6742)		29 CFR 1904.8
BODILY INJURY OR PROPERTY DAMAGE RELATED TO RADIOACTIVE MATERIAL		Report bodily injury or property damage arising out of or in connection with the possession or use of radioactive material at the plant or in the course of transportation or in the event any claim is made therefore. {10 CFR 140.6(a)} NOTE: Written notice is required "...as promptly as practicable."	10 CFR 140.6

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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
UNIT 2 OPERATING LICENSE CONDITIONS (continued on next page)	Notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission. {Unit 2 Operating License Section 2, Item G} NOTE: Notification required as soon as possible but not later than one hour. Report any violations of the following Unit 2 Operating License requirements: - Maximum Power Level: Operation of ·vels not in excess of 2700 megawatts - Antitrust Conditions: Non-compliance ·es C and D to the Unit 2 Operating	Criterion {Unit 2 Operating License Condition 2.F.} same as 24 hour notification NOTE: Provide a written follow-up report within fourteen (14) days.	Unit 2 Operating License

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OTHER REGULATORY REPORTS
(Page 9 of 11)

EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
UNIT 2 OPERATING LICENSE CONDITIONS (continued)	- Security and Safeguards: Failure to fully implement and maintain in effect all provisions of the NRC approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). - Construction and Operational Activities Affection Environmental Compliance: Additional construction or operational activities which result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement dated April 1982. {Unit 2 Operating License Section 2, Items C.(1), C.(3) through C.(17), D. and E} NOTE: Report within 24 hours by telephone and confirm by telegram, mailgram or facsimile transmission to the NRC Regional Administrator, Region II, or his designee no later than the first working day following the violation.	Criterion {Unit 2 Operating License Condition 2.F.} same as 24 hour notification NOTE: Provide a written follow-up report within fourteen (14) days.	Unit 2 Operating License

APPENDIX B
OTHER REGULATORY REPORTS
(Page 10 of 11)

EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
UNUSUAL OR IMPORTANT ENVIRONMENTAL EVENTS	Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to station operation shall be recorded and promptly **reported to the NRC within 72 hours**, followed by a written report within 30 days. If an outside government agency (local, state or federal) is required to be notified, because of the environmental event, Then a four (4) hour ENS Notification is required in accordance with 10 CFR 50.72(b)(2)(xi). No routine monitoring programs are required to implement this condition. The following are examples of unusual or important events: excessive bird impaction, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, unusual fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.	Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to station operation shall be recorded and promptly reported to the NRC within 72 hours, **followed by a written report within 30 days.** No routine monitoring programs are required to implement this condition. The following are examples of unusual or important events: excessive bird impaction, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, unusual fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.	Environmental Protection Plant Section 4.1 AP 0005762, "Plant Guide to Routine/Non-Routine Environmental Reporting and Significant Events."

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OTHER REGULATORY REPORTS
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EVENT OF CONDITION	NRC PHONE NOTIFICATION	NRC WRITTEN NOTIFICATION	CRITERIA FOR REPORTABILITY
FITNESS FOR DUTY	Report significant fitness-for-duty events including: Sale, use or possession of illegal drugs within the protected area, and Any acts by any person licensed under 10 CFR Part 55 to operate a power reactor or by any supervisory personnel assigned to perform duties within the scope of part 26 involving the sale, use or possession of a controlled substance, resulting in confirmed positive tests on such persons, involving the use of alcohol within the protected area, or resulting in a determination of unfitness for scheduled work due to the consumption of alcohol. NOTE: Notifications must be made within 24 hours of the discovery of the event.		10 CFR 26.73
DEGRADED SPENT FUEL STORAGE CASK OR CONFINEMENT SYSTEM	A defect in any spent fuel storage cask structure, system or component that is important to safety. (4 hour report)	30 day written report	72.75(b)(2)
	A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask under a general license issued under 10 CFR 72.210. (4 hours report)	30 day written report	72.75(b)(3)

APPENDIX C
TECHNICAL SPECIFICATION AND FSAR SPECIAL REPORTS
(Page 1 of 2)

AUTHORITY UNIT 1/2 T.S.	EVENT	REQUIRED ACTION
Unit 1, 3.3.3.1 Table 3.3-6, Action 15 Unit 2, 3.3.3.1 Table 3.3-6, Action 27	With any of the following Radiation Monitoring Instrumentation having less than the minimum number of channels operable for greater than 72 hours: 1.d. Containment Area Hi Range Area Monitor 2.c. Noble Gas Effluent Monitors	Submit Special Report IAW T.S. 6.9.2 within 14 days.
Unit 1, 3.3.3.8 Table 3.3-11, Action 4 Unit 2, 3.3.3.6 Action c	With the number of Operable channels of the Reactor Vessel Level Monitoring System or the Containment Sump Water Level (narrow range or wide range) instruments 1 less than the total number of channels for greater than 7 days (if repairs are feasible without shutting down the reactor).	Submit a Special Report IAW T.S. 6.9.2 within 30 days.
Unit 1 3.3.3.8 Table 3.3-11 Action 5 Unit 2 3.3.3.6 Action d	With the number of Operable channels of the Reactor Vessel Level Monitoring System or the Containment Sump Water Level (narrow range or wide range) instruments 1 less than the minimum number of channels for greater than 48 days (if repairs are feasible without shutting down the reactor).	Submit a Special Report IAW T.S. 6.9.2 within 30 days.
Unit 1/2 4.4.5.5.a	Following the completion of each inservice inspection of steam generator tubes.	Submit a Special Report IAW T.S. 6.9.2 within 15 days.
Unit 2 4.4.5.5.b	After completion of the steam generator tube inservice inspection.	Submit a Special Report IAW T.S. 6.9.2 within 12 months.

APPENDIX C
TECHNICAL SPECIFICATION AND FSAR SPECIAL REPORTS
(Page 2 of 2)

AUTHORITY UNIT 1/2 T.S.	EVENT	REQUIRED ACTION
Unit 1 3.5.2 Action b or 3.5.3 Action c Unit 2 3.5.2 Action b or 3.5.3 Action b	In the event ECCS is actuated and injects water into the RCS.	Submit a Special Report IAW T.S. 6.9.2 within 90 days.
Unit 1 4.7.9.1.3 Unit 2 4.7.10.3	If sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries or removable contamination.	Unit 1 - submit a Special Report IAW T.S. 6.9.2 within 90 days. Unit 2 - submit a report to the NRC on an annual basis.

APPENDIX D
OFFSITE DOSECALCULATION MANUAL (ODCM) SPECIAL REPORTS
(Page 1 of 2)

AUTHORITY ODCM	EVENT	REQUIRED ACTION
Control Section 3.11.1.2 Action a	With calculated dose from the release of radioactive materials in liquid effluents exceeding any of the ODCM Control Section limits.	Submit a Special Report within 30 days.
Control Section 3.11.1.3 Action a	With radioactive liquid wastes being discharged without treatment and in excess of the ODCM Control Section limits and any portion of the Liquid Radwaste Treatment System not in operation.	Submit a Special Report within 30 days.
Control Section 3.11.2.2 Action a	With the calculated air dose from radioactive noble gas in gaseous effluents exceeding any of the ODCM Control Section limits.	Submit a Special Report within 30 days.

APPENDIX D
OFFSITE DOSE CALCULATION MANUAL (ODCM) SPECIAL REPORTS
(Page 2 of 2)

AUTHORITY ODCM	EVENT	REQUIRED ACTION
Control Section 3.11.2.3 Action a	With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents exceeding any of the ODCM Control Section limits.	Submit a Special Report within 30 days.
Control Section 3.11.2.4 Action a	With gaseous waste being discharged without treatment and in excess of the ODCM Control Section limits.	Submit a Special Report within 30 days.
Control Section 3.11.4 Action a & 10 CFR 20.2203.a	With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Control Section 3.11.1.2.a., 3.11.1.2.b., 3.11.2.2.a., 3.11.2.2.b., 3.33.2.3.a. or 3.11.2.3.b.	Submit a Special Report within 30 days.
Control Section 3.12.1 Action b	With the confirmed level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding reporting level of ODCM Table 3.12-2 when averaged over any calendar quarter.	Submit a Special Report within 30 days.

APPENDIX E
POSTING OF REQUIRED DOCUMENTS PURSUANT TO 10CFR19
(Page 1 of 1)

TO: Plant Personnel, FP&L Personnel Contractors, Vendors and Visitors

LOCATION: St. Lucie Plant

DATE:

FROM: Plant General Manager

COPIES TO: File

SUBJECT: **POSTING OF REQUIRED DOCUMENTS PURSUANT TO 10CFR19**

1) To meet the posting requirements of Section 223B of the Atomic Energy Act of 1954, Sections 206 and 211 of the Energy Reorganization Act of 1974, 10 CFR 19.11(c), and Appendix A to 29 CFR 24, Appendices E, F, H, I, and J of this procedure and NRC Form 3 shall be posted at this location.

2) In accordance with 10 CFR 19.11.a.4, copies of the following documents shall be posted at this location.

   a) Any NRC notice of violation involving radiological working conditions.

   b) Any NRC notice of proposed imposition of civil penalty.

   c) Any NRC order issued pursuant to 10 CFR 2.202.

   d) Any FPL response to the above documents.

3) In accordance with 10CFR19.11.a., copies of the following documents are available by contacting the Plant Licensing.

   a) 10CFR19 Notices, Instructions and Reports to Workers; Inspections.

   b) 10CFR20 Standards for Protection Against Radiation.

   c) Operating License DPR-67 and NPF-16

   d) Operating procedures applicable to licensed activities

   e) Unit 1 and Unit 2 Final Safety Analysis Reports

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Plant General Manager
St. Lucie Plant

APPENDIX F
SUBSECTION 223B OF THE ATOMIC ENERGY ACT OF 1954, AS AMENDED
(Page 1 of 1)

ANY INDIVIDUAL DIRECTOR, OFFICER OR EMPLOYEE OF A FIRM CONSTRUCTING OR SUPPLYING THE COMPONENTS OF ANY UTILIZATION FACILITY REQUIRED TO BE LICENSED UNDER SECTION 103 OR 104 B OF THIS ACT WHO BY ACT OR OMISSION, IN CONNECTION WITH SUCH CONSTRUCTION OR SUPPLY, KNOWINGLY AND WILLFULLY VIOLATES OR CAUSES TO BE VIOLATED, ANY SECTION OF THIS ACT, ANY RULE, REGULATION OR ORDER ISSUED THEREUNDER OR ANY LICENSE CONDITION, WHICH VIOLATION RESULTS OR IF UNDETECTED COULD HAVE RESULTED, IN A SIGNIFICANT IMPAIRMENT OF A BASIC COMPONENT OF SUCH A FACILITY SHALL, UPON CONVICTION, BE SUBJECT TO A FINE OF NOT MORE THAN $25,000 FOR EACH DAY OF VIOLATION OR TO IMPRISONMENT NOT TO EXCEED TWO YEARS OR BOTH. IF THE CONVICTION IS FOR A VIOLATION COMMITTED AFTER A FIRST CONVICTION UNDER THIS SUBSECTION, PUNISHMENT SHALL BE A FINE OF NOT MORE THAN $50,000 PER DAY OF VIOLATION OR IMPRISONMENT FOR NOT MORE THAN TWO YEARS OR BOTH. FOR THE PURPOSES OF THIS SUBSECTION, THE TERM BASIC COMPONENT MEANS A FACILITY STRUCTURE, SYSTEM, COMPONENT OR PART THEREOF NECESSARY TO ASSURE

(1) THE INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY,

(2) THE CAPABILITY TO SHUT-DOWN THE FACILITY AND MAINTAIN IT IN A SAFE SHUT-DOWN CONDITION OR

(3) THE CAPABILITY TO PREVENT OR MITIGATE THE CONSEQUENCES OF ACCIDENTS WHICH COULD RESULT IN AN UNPLANNED OFFSITE RELEASE OF QUANTITIES OF FISSION PRODUCTS IN EXCESS OF THE LIMITS ESTABLISHED BY THE COMMISSION.

THE PROVISIONS OF THIS SUBSECTION SHALL BE PROMINENTLY POSTED AT EACH SITE WHERE A UTILIZATION FACILITY REQUIRED TO BE LICENSED UNDER SECTION 103 OR 104 B OF THIS ACT IS UNDER CONSTRUCTION AND ON THE PREMISES OF EACH PLANT WHERE COMPONENTS FOR SUCH A FACILITY ARE FABRICATED.

APPENDIX G
EVENT NOTIFICATION WORKSHEET
(Page 1 of 2)

Name of NDDO contacted: ___________________

NRC FORM 361
(12-2000)

U.S. NUCLEAR REGULATORY COMMISSION
OPERATIONS CENTER

REACTOR PLANT
EVENT NOTIFICATION WORKSHEET

EN #

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #

EVENT TIME & ZONE	EVENT DATE	POWERMODE BEFORE	POWERMODE AFTER

	EVENT CLASSIFICATIONS			**1-Hr. Non-Emergency 10 CFR 50.72(b)(1)**					
	GENERAL EMERGENCY	GEN/AAEC		TS Deviation	ADEV		(v)(A)	Safe S/D Capability	AINA
	SITE AREA EMERGENCY	SIT/AAEC		**4-Hr. Non-Emergency 10 CFR 50.72(b)(2)**			(v)(B)	RHR Capability	AINB
	ALERT	ALE/AAEC		(i) TS Required S/D	ASHU		(v)(C)	Control of Rad Release	AINC
	UNUSUAL EVENT	UNU/AAEC		(iv)(A) ECCS Discharge to RCS	ACCS		(v)(D)	Accident Mitigation	AIND
	50.72 NON-EMERGENCY	(see next columns)		(iv)(B) RPS Actuation (scram)	ARPS		(xii)	Offsite Medical	AMED
	PHYSICAL SECURITY (73.71)	DDDD		(xi) Offsite Notification	APRE		(xiii)	Loss Comm/Asmt / Resp	ACOM
	MATERIAL/EXPOSURE	B???		**8-Hr. Non-Emergency 10 CFR 50.72(b)(3)**			**60-Day Optional 10 CFR 50.73(a)(1)**		
	FITNESS FOR DUTY	HFIT		(ii)(A) Degraded Condition	ADEG		Invalid Specified System Actuation		AINA
	OTHER UNSPECIFIED REQMT.	(see last column)		(ii)(B) Unanalyzed Condition	AUNA		**Other Unspecified Requirement (Identify)**		
	INFORMATION ONLY	NNF		(iv)(A) Specified System Actuation	AESF				NONR
									NONR

DESCRIPTION

Include: Systems affected, actuations and their initiating signals, causes, effect of event on plant, actions taken or planned, etc. (Continue on back)

NOTIFICATIONS	YES	NO	WILL BE			
NRC RESIDENT				ANYTHING UNUSUAL OR NOT UNDERSTOOD?	☐ YES (Explain above)	☐ NO
STATE(s)				DID ALL SYSTEMS FUNCTION AS REQUIRED?	☐ YES	☐ NO (Explain above)
LOCAL						
OTHER GOV AGENCIES				MODE OF OPERATION UNTIL CORRECTED:	ESTIMATED RESTART DATE:	ADDITIONAL INFO ON BACK ☐ YES ☐ NO
MEDIA/PRESS RELEASE						

APPENDIX G
EVENT NOTIFICATION WORKSHEET
(Page 2 of 2)

RADIOLOGICAL RELEASES: CHECK OR FILL IN APPLICABLE ITEMS (specific details/explanations should be covered in event description)

	LIQUID RELEASE		GASEOUS RELEASE		UNPLANNED RELEASE		PLANNED RELEASE		ONGOING	TERMINATED
	MONITORED		UNMONITORED		OFFSITE RELEASE		T.S. EXCEEDED		RM ALARMS	AREAS EVACUATED
	PERSONNEL EXPOSED OR CONTAMINATED				OFFSITE PROTECTIVE ACTIONS RECOMMENDED				*State release path in description	

	Release Rate (Ci/sec)	% T.S. LIMIT	HOO GUIDE	Total Activity (Ci)	% T.S. LIMIT	HOO GUIDE
Noble Gas			0.1 Ci/sec			1000 Ci
Iodine			10 uCi/sec			0.01 Ci
Particulate			1 uCi/sec			1 mCi
Liquid (excluding tritium and dissolved noble gases)			10 uCi/min			0.1 Ci
Liquid (tritium)			0.2 Ci/min			5 Ci
Total Activity						

	PLANT STACK	CONDENSER/AIR EJECTOR	MAIN STEAM LINE	SG BLOWDOWN	OTHER
RAD MONITOR READINGS					
ALARM SETPOINTS					
% T.S. LIMIT (if applicable)					

RCS OR SG TUBE LEAKS: CHECK OR FILL IN APPLICABLE ITEMS: (specific details/explanations should be covered in event description)

LOCATION OF THE LEAK (e.g., SG #, valve, pipe, etc.)			
LEAK RATE	UNITS: gpm/gpd	T.S. LIMITS	SUDDEN OR LONG-TERM DEVELOPMENT
LEAK START DATE	TIME	COOLANT ACTIVITY AND UNITS:	PRIMARY / SECONDARY
LIST OF SAFETY RELATED EQUIPMENT NOT OPERATIONAL			

EVENT DESCRIPTION (Continued from front)

APPENDIX H
SECTION 211 OF THE ENERGY REORGANIZATION ACT OF 1974
(42 U.S.C. § 5851)
EMPLOYEE PROTECTION
(Page 1 of 3)

Sec. 211. (a)(1) No employer may discharge any employee or otherwise discriminate against any employee with respect to his compensation, terms, conditions or privileges of employment because the employee (or any person action pursuant to a request of the employee)--

A. notified his employer of an alleged violation of this Act or the Atomic Energy Act of 1954 (42 U.S.C. 2011 et seq.);
B. refused to engage in any practice made unlawful by this Act or the Atomic Energy Act of 1954, if the employee has identified the alleged illegality to the employer;
C. testified before Congress or at any Federal or State proceeding regarding any provision (or proposed provision) of this Act or the Atomic Energy Act of 1954;
D. commenced, caused to be commenced or is about to commence or cause to be commenced a proceeding under this Act or the Atomic Energy Act of 1954, as amended or a proceeding for the administration or enforcement of any requirement imposed under this Act or the Atomic Energy Act of 1954, as amended;
E. testified or is about to testify in any such proceeding or;
F. assisted or participated or is about to assist or participate in any manner in such a proceeding or in any other manner in such a proceeding or in any other action to carry out the purposes of this Act or the Atomic Energy Act of 1954, as amended.

2. For the purposes of this section, the term employer includes --
   A. a license of the Commission or of an agreement State under section 274 of the Atomic Act of 1954 (42 U.S.C. 2021);
   B. an applicant for a license from the Commission or such an agreement State;
   C. a contractor or subcontractor of such a licensee or applicant; and
   D. a contractor or subcontractor of the Department of Energy that is indemnified by the Department under section 170d. of the Atomic Energy Act of 1954 (42 U.S.C. 2210(d)), but such term shall not include any contractor or subcontractor covered by Executive Order No. 12344.

b1 Any employee who believes that he has been discharged or otherwise discriminated against by any person in violation of subsection (a) may, within 180 days after such violation occurs, file (or have any person file on his behalf) a complaint with the Secretary of Labor (in this section referred to as the Secretary) alleging such discharge or discrimination. Upon receipt of such a complaint, the Secretary shall notify the person named in the complaint of the filing of the complaint, the Commission and the Department of Energy.

APPENDIX H
SECTION 211 OF THE ENERGY REORGANIZATION ACT OF 1974
(42 U.S.C. § 5851)
EMPLOYEE PROTECTION
(Page 2 of 3)

2A Upon receipt of a complaint filed under paragraph (1), the Secretary shall conduct an investigation of the violation alleged in the complaint. Within thirty days of the receipt of such complaint, the Secretary shall complete such investigation and shall notify in writing the complainant (and any person acting in his behalf) and the person alleged to have committed such violation of the results of the investigation conducted pursuant to this subparagraph. Within ninety days of the receipt of such complaint the Secretary shall, unless the proceeding on the complaint is terminated by the Secretary on the basis of a settlement entered into by the Secretary and the person alleged to have committed such violation, issue an order either providing the relief prescribed by subparagraph (B) or denying the complaint. An order of the Secretary shall be made on the record after notice and opportunity for public hearing. Upon the conclusion of such hearing and the issuance of a recommended decision that the complaint has merit, the Secretary shall issue a preliminary order providing the relief prescribed in subparagraph (B), but may not order compensatory damages pending a final order. The Secretary may not enter into a settlement terminating a proceeding on a complaint without the participation and consent of the complainant.

B. If, in response to a complaint filed under paragraph (1), the Secretary determines that a violation of subsection (a) has occurred, the Secretary shall order the person who committed such violation to (i) take affirmative action to abate the violation and (ii) reinstate the complainant to his former position together with the compensation (including back pay), terms, conditions and privileges of his employment and the Secretary may order such person to provide compensatory damages to the complainant. If an order is issued under this paragraph, the Secretary, at the request of the complainant shall assess against the person whom the order is issued a sum equal to the aggregate amount of all costs and expenses (including attorney's and expert witness fees) reasonably incurred, as determined by the Secretary, by the complainant for or in connection with, the bringing of the complaint upon which the order was issued.

3A The Secretary shall dismiss a complaint filed under paragraph (1) and shall not conduct the investigation required under paragraph (2), unless the complainant has made a prima facie showing that any behavior described in subparagraphs (A) through (F) of subsection (a)(1) was a contributing factor in the unfavorable personnel action alleged in the complainant.

B. Notwithstanding a finding by the Secretary that the complainant has made the showing required by subparagraph (A), no investigation required under paragraph (2) shall be conducted if the employer demonstrates, by clear and convincing evidence, that it would have taken the same unfavorable personnel action in the absence of such behavior.

C. The Secretary may determine that a violation of subsection (a) has occurred only if the complainant has demonstrated that any behavior described in subparagraphs (A) through (F) of subsection (a)(1) was a contributing factor in the unfavorable personnel action alleged in the complaint.

D. Relief may not be ordered under paragraph (2) if the employer demonstrates by clear and convincing evidence that it would have taken the same unfavorable personnel action in the absence of such behavior.

c1. Any person adversely affected or aggrieved by an order issued under subsection (b) may obtain review of the order in the United States court of appeals for the circuit in which the violation, with respect to which the order was issued, allegedly occurred. The petition of review must be filed within sixty days from the issuance of the Secretary's order. Review shall conform to chapter 7 of title 5 of the United States Code. The commencement of proceedings under this subparagraph shall not, unless ordered by the court, operate as a stay of the Secretary's order.

APPENDIX H
SECTION 211 OF THE ENERGY REORGANIZATION ACT OF 1974 (42 U.S.C. § 5851)
EMPLOYEE PROTECTION
(Page 3 of 3)

2. An order of the Secretary with respect to which review could have been obtained under paragraph (1) shall not be subject to judicial review in any criminal or other civil proceeding.

d. Whenever a person has failed to comply with an order issued under subsection (b)(2), the Secretary may file a civil action in United States district court for the district in which the violation was found to occur to enforce such order. In actions brought under this subsection, the district courts shall have jurisdiction to grant all appropriate relief including, but not limited to, injunctive relief, compensatory and exemplary damages.

e1. Any person on whose behalf an order was issued under paragraph (2) of subsection (b) may commence a civil action against the person to whom such order was issued to require compliance with such order. The appropriate United States district court shall have jurisdiction, without regard to the amount in controversy or the citizenship of the parties, to enforce such order.

2. The court, in issuing any final order under this subsection, any award costs of litigation (including reasonable attorney and expert witness fees) to any part whenever the court determines such award is appropriate.

f. Any non-discretionary duty imposed by this section shall be enforceable in a mandamus proceeding brought under section 1361 of title 28 of the United States Code.

g. Subsection (a) shall not apply with respect to any employee who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of this Act or the Atomic Energy Act of 1954, as amended. [42 U.S.C. 5851.]

h. This section may not be construed to expand, diminish or other wise affect any right otherwise available to an employee under Federal or State law to redress the employee's discharge or other discriminatory action taken by the employer against the employee.

i. The provisions of this section shall be prominently posted in any place of employment to which this section applies.

j1. The Commission or the Department of Energy shall not delay taking appropriate action with respect to an allegation of a substantial safety hazard on the basis of--
   A. the filing of a complaint under subsection (b1) arising from such allegation; or
   B. any investigation by the Secretary or other action, under this section in response to such complaint.

2. A determination by the Secretary under this section that a violation of subsection (a) has not occurred shall not be considered by the Commission or the Department of Energy in its determination of whether a substantial safety hazard exists.

APPENDIX I
APPENDIX A TO 29 CFR PART 24 - YOUR RIGHTS UNDER THE ENERGY REORGANIZATION ACT
(Page 1 of 1)

YOUR RIGHTS UNDER THE ERA

The Energy Reorganization Act (ERA), makes it illegal for an employer covered by the act - including a licensee of the Nuclear Regulatory Commission (NRC) or an agreement state, an applicant for a license, a contractor or subcontractor of a licensee or applicant and a contractor or subcontractor of the Department Of Energy (DOE) under the Atomic Energy Act (AEA) - to discharge or otherwise discriminate against an employee in terms of compensation, conditions or privileges of employment because the employee or any person acting at an employee's request performs a protected activity.

Right to raise a safety concern: you are engaged in protected activity when you:

(1) Notify your employer of an alleged violation of the ERA or the AEA;
(2) Refuse to engage in any practice made unlawful by the ERA or the AEA, if you have identified the alleged illegality to the employer;
(3) Testify before congress or at any federal or state proceeding regarding any provision or proposed provision of the ERA or the AEA;
(4) Commence or cause to be commenced a proceeding under the ERA, or a proceeding for the administration or enforcement of any requirement imposed under the ERA;
(5) Testify or are about to testify in any such proceeding; or
(6) Assist or participate in such a proceeding or in any other action to carry out the purposes of the ERA or the AEA.

Unlawful acts by employers: It is unlawful for an employer to intimidate, threaten, restrain, coerce, blacklist, discharge or in any other manner discriminate against any employee because the employee has engaged in protected activity.

Complaint: An employee or employee representative may file a complaint charging discrimination in violation of the ERA within 180 days of the discriminatory action. A complaint must be in writing and should include a full statement of facts, including the protected activity engaged in by the employee, knowledge by the employer of the protected activity, and the basis for believing that the activity resulted in discrimination against the employee by the employer. A complaint may be filed in person or by mail at the nearest local office of the Occupational Safety and Health Administration (OSHA), U.S. Government, Department of Labor, or with the office of the Assistant Secretary, OSHA, U.S. Department of Labor, Washington, D.C. 20210.

Enforcement: OSHA will review the complaint to ensure that it makes an initial showing of discrimination. If not, or if the employer provides clear and convincing evidence that there was no discrimination, there will be no investigation. If the required showing is made, OSHA will notify the employer and conduct an investigation to determine whether a violation has occurred. Either the employee or the employer may request a hearing before an ALJ.

Relief: If discrimination is found, the employer will be required to provide appropriate relief, including reinstatement (even for the period between the ALJ decision and appeal), back wages or compensation for injury suffered from the discrimination, and attorney's fees and costs.

Caution: The preceding protections and remedies are not available to employees who engage in deliberate violations of the ERA or the AEA.

For additional Information: Contact the Occupational Safety and Health Administration, U.S. Government, Department of Labor (listed in telephone directories), or see the Department of Labor's web site at: WWW.OSHA.GOV.

Employers are required to display this poster where employees can readily see it.

APPENDIX J
REPORTING OF DEFECTS AND NONCOMPLIANCE
(Page 1 of 1)

Pursuant to Florida Power & Light Company's intent to meet the requirements of the Energy Reorganization Ac of 1974 and Title 10 Code of Federal Regulations Part 21 (10CFR21), any individual director or responsible officer involved with the design, construction, or operation of a facility regulated by the Atomic Energy Act of 1954 shall comply with Section 206 of Energy Reorganization Act of 1974 below:

NONCOMPLIANCE

SEC.206.

(a) Any individual director, or responsible officer of a firm constructing, owning, operating, or supplying the components of any facility or activity which is licensed or otherwise regulating pursuant to the Atomic Energy Act of 1954, as amended, or pursuant to this Act, who obtains information reasonably indicating that such facility or activity or basic components supplied to such facility or activity:

(1) Fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable, rule, regulation, order, or license of the Commission relating to substantial safety hazards, or

(2) Contains a defect which could create a substantial safety hazard, as defined by regulations which the Commission shall promulgate, shall immediately notify the Commission of such failure to comply, or of such defect, unless such person has actual knowledge that the Commission has been adequately informed of such defect or failure to comply.

(b) Any person who knowingly and consciously fails to provide the notice required by subsection (a) of this section shall be subject to a civil penalty in an amount equal to the amount provided by section 234 of the Atomic Energy Act of 1954, as amended.

(c) The requirements of this section shall be prominently posted on the premises of any facility licensed or otherwise regulated pursuant to the Atomic Energy Action of 1954, as amended.

(d) The Commission is authorized to conduct such reasonable inspections and other enforcement activities as needed to insure compliance with the provision of this section.

To implement Section 206, the Nuclear Regulatory Commission published 10CFR21 which has the following stated purpose:

Sec.21.1 Purpose - The regulation in this part establish procedures and requirements for implementation of Section 206 of the Energy Reorganization Act of 1974. That section requires any individual director or responsible officer of a firm constructing, owning, operating, or supplying the components of any facility or activity which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act or 1974, who obtains information reasonably indicating: (a) that the facility, activity or basic component supplied to such facility or activity fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the commission relating to substantial safety hazards or (b) that the facility, activity or basic component supplied to such facility or activity contains defects, which could create a substantial safety hazard, to immediately notify the Commission of such failure to comply or such defect, unless he has actual knowledge that the Commission has been adequately informed of such defect or failure to comply.

Florida Power & Light Company has adopted Corporate Procedure GO 90, *Reporting Nuclear Power Plant Deficiencies*, Nuclear Division Nuclear Policy NP 808, *Evaluating and Reporting Defects and Failures to Comply for Substantial Safety Hazards in Accordance with 10 CFR 21;* Nuclear Division Interdepartmental Procedure IP 801, *Evaluating and Reporting Defects and Failures to Comply for Substantial Safety Hazards in Accordance with 10 CFR 21;* and St. Lucie Quality Instruction QI 16-PR/PSL-4, *Evaluating and Reporting Defects and Failures to Comply for Substantial Safety Hazards in Accordance with 10 CFR 21* to implement Section 206 and 10CFR21. These procedures provide detailed responsibilities for identification, processing, and reporting a defect, and noncompliance covered under these requirements. Each individual is responsible for reporting defects or a noncompliance to their supervisors.

Copies of 10CFR21 and the applicable procedure(s) can be obtained from the Plant Licensing. Any individual who identifies, or obtains in-house information regarding a potential defect or failure to comply shall generate a Condition Report (CR) to notify the Plant General Manager. The information may also be reported through the FPL Nuclear Safety Speakout Program or the NRC as indicated on NRC Form 3. Any individual receiving information from a supplier concerning a potential defect or failure to comply shall forward all information, in writing, to the Quality Supervisor - Performance Assessment, Juno Beach. This includes software error notices.

EPIPs

FPL	**ST. LUCIE PLANT** **EMERGENCY PLAN IMPLEMENTING PROCEDURE** SAFETY RELATED	Procedure No. **EPIP-01** Current Revision No. **6** Effective Date **04/18/03**

Title:

CLASSIFICATION OF EMERGENCIES

Responsible Department: **EMERGENCY PLANNING**

REVISION SUMMARY:

Revision 6 – Incorporated PCR #03-0403 to delete wording regarding technical specification limits and correctly place symbols. (J. R. Walker, 02/21/03)

Revision 5 – Clarified EALs under alert. (J. R. Walker, 07/25/02)

Revision 4 - Revised IAW revision to E-Plan (R40). Revised initiating condition for RCS leakage. Added EALs under security threat initiating condition. Added definitions for EAL and IC. Added guidance for multiple and dual unit events. Made editorial and administrative changes. (J.R. Walker, 05/23/02)

Revision 3 - Added PMAI references, added definitions for OCA, PA and power block, clarified classification guidance and made editorial/administrative changes. (J. R. Walker, 02/09/01)

Revision 2 - Clarified initiating conditions and emergency action levels to correspond to changes in the PSL emergency plan in accordance with PMAI PM99-09-154, defined classification table and made editorial changes. (J. R. Walker, 10/13/00)

Revision 1 - Revised to RCS EAL for alert based on NESP007 guidance. (J. R. Walker, 04/21/00)

Revision	FRG Review Date	Approved By	Approval Date
0	12/15/97	J. Scarola Plant General Manager	12/15/97

Revision	FRG Review Date	Approved By	Approval Date
6	02/20/03	R. E. Rose Plant General Manager N/A Designated Approver N/A Designated Approver (Minor Correction)	02/21/03

S___OPS	
DATE	
DOCT	PROCEDURE
DOCN	EPIP-01
SYS	
COM	COMPLETED
ITM	6

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides instructions on the classification of emergencies at St. Lucie Plant.

Emergency classifications in order of increasing seriousness are:

- Unusual Event
- Alert
- Site Area Emergency
- General Emergency

Specific criteria are provided to assure proper escalation and de-escalation between emergency classification levels.

2.0 REFERENCES / RECORDS REQUIRED / COMMITMENT DOCUMENTS

NOTE

One or more of the following symbols may be used in this procedure:

§ Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, Operating Experience, License Renewal, etc. and shall NOT be revised without Facility Review Group review and Plant General Manager approval.

¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.

Ψ Indicates a step that requires a sign off on an attachment.

2.1 References

1. St. Lucie Plant Radiological Emergency Plan (E-Plan)
2. E-Plan Implementing Procedures (EPIP 00-13)
3. C-200, Offsite Dose Calculation Manual (ODCM)
4. AP 0010502, Oil and Hazardous Material Emergency Response Plan
5. ¶$_1$ NUREG-1022, Section 3.1.1

2.1 References (continued)

6. ¶2 NRC IEN No. 85-80, Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notifications, October 15, 1985

7. ¶3 NRC EPPOS No. 2, Emergency Preparedness Position (EPPOS) on Timeliness of Classification of Emergency Conditions, August, 1995

8. ¶4 PMAI PM98-01-017, Loss of Seismic Monitoring Capability

2.2 Records Required

The basis for classifying an emergency condition shall be recorded in appropriate emergency logs.

2.3 Commitment Documents

- §1 CR 00-0614 (RCS leakage during shutdown cooling)

- §2 PMAI PM99-09-154 (IC and EAL changes submitted under FPL letter L-98-2000).

3.0 RESPONSIBILITIES

3.1 Nuclear Plant Supervisor (NPS)

1. The Nuclear Plant Supervisor is responsible to promptly classify abnormal situations into one of the four defined categories.

2. If an emergency has been declared, the Nuclear Plant Supervisor is responsible for assuming the position of Emergency Coordinator and retaining this position until relieved.

3.2 Emergency Coordinator (EC)

The Emergency Coordinator is responsible to continually evaluate changes in plant conditions against the classification table in this procedure.

4.0 DEFINITIONS

4.1 Emergency Action Level (EAL)

1. A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (on-site or off-site); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

4.2 Emergency Classes

1. Unusual Event

   This classification is represented by off-normal events or conditions at the plant for which no significant degradation of the level of safety of the plant has occurred or is expected. Any releases of radioactive material which may have occurred or which may be expected are minor and constitute no appreciable health hazard.

2. Alert

   This classification is represented by events which involve an actual or potential substantial degradation of the level of safety of the plant combined with a potential for limited uncontrolled releases of radioactivity from the plant.

3. Site Area Emergency

   This classification is composed of events which involve actual or likely major failures of plant functions needed for protection of the public combined with a potential for significant uncontrolled releases of radioactivity from the plant.

4. General Emergency

   This classification is composed of events which involve actual or imminent substantial core degradation and potential loss of containment integrity combined with a likelihood of significant uncontrolled releases of radioactivity from the plant.

4.3 Classification Table

A composite of Initiating Conditions (ICs) and their Emergency Action Levels (EALs) used to evaluate off normal/emergency conditions resulting in declaration of one of the four Emergency Classes, as appropriate. The Table is arranged in the following categories:

1. Events Affecting Primary Pressure
   - A. Abnormal Primary Leak Rate
   - B. Abnormal Primary/Secondary Leak Rate
   - C. Loss of Secondary Coolant
2. Abnormal Radiation, Contamination and Effluent Releases
   - A. Uncontrolled Effluent Release
   - B. High Radiation Levels in Plant
3. Fires, Explosions
4. Accident Involving Fuel
   - A. Fuel Element Failure
   - B. Fuel Handling
5. Natural Emergencies
   - A. Earthquake
   - B. Hurricane
   - C. Tornado
   - D. Abnormal Water Level
6. Miscellaneous Events
   - A. Increased Awareness or Potential Core Melt
7. Electrical Malfunctions
   - A. Loss of Power

4.3 Classification Table (continued)

8. Degradation of Control Capabilities

A. Loss of Plant Control Functions

B. Loss of Alarms, Communications, Monitoring

9. Hazards to Station Operation

A. Aircraft, Missile

B. Turbine Failure

C. Toxic or Flammable Gas

10. Security Threat

4.4 Initiating Condition (IC)

1. One of a predetermined subset of nuclear power plant conditions where either the potential exists for a radiological emergency, or such an emergency has occurred.

4.5 **Plant** - The St. Lucie Plant, Unit 1 and Unit 2

4.6 **Site** - A general term referring to the location of the St. Lucie Nuclear Power Plant. Other terms related to the site are given below:

1. **Owner Controlled Area** - That portion of FPL property surrounding and including the St. Lucie Nuclear Power Plant which is subject to limited access and control as deemed appropriate by FPL.

2. **Protected Area** - The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed with the security perimeter fence. The area within which accountability of personnel is maintained in an emergency.

4.6 Site (continued)

3. §2 **Power Block** - Structures, systems or components in the areas listed below that support the production of power. This includes any equipment needed for the direct generation of power or necessary for safe operation and/or shutdown of one or both of the reactors.

A. Reactor Containment and Shield Buildings

B. Reactor Auxiliary Buildings including the following areas:

1. Refueling Water Tank (RWT)
2. Component Cooling Water (CCW) platform area
3. Diesel Generator Buildings and Fuel Oil Storage Tanks
4. Fuel Handling Building
5. Primary Water Tank and Pumps

C. Intake Area

D. Discharge Canal & Headwall

E. Ultimate Heat Sink Structure

F. Fire Protection System including the fire pumps and the City Water Storage Tanks (CWST), but not including parts of the system associated with the North or South Service Buildings or other outlying facilities.

G. Turbine Buildings (all levels)

H. Condensate Storage Tanks (CST)

I. Main, Auxiliary and Startup Transformers

J. Steam Trestles

K. Turbine Lube Oil Storage Tanks

L. Gas House

5.0 INSTRUCTIONS

5.1 Direct Initial Investigative and Mitigating Actions to Address the Event

1. If the event involves entry into the Off-Normal Operating Procedures (ONOPs) or Emergency Operating Procedures (EOPs), Then perform steps per ONOPs or EOPs until appropriate or directed to classify event.

2. If the event involves a release of hazardous materials to the environment, Then respond per AP 0010502, Oil and Hazardous Material Emergency Response Plan.

3. If the event involves a release of radioactive material to the environment, Then direct Chemistry personnel to implement EPIP-09, Off-site Dose Calculations.

END OF SECTION 5.1

NOTE

Initiating Conditions / Emergency Action Levels are applicable to all modes unless otherwise indicated.

5.2 Classifying the Event

1. ¶3 A goal of fifteen (15) minutes should be used for assessing and classifying an emergency once indications (Emergency Action Levels (EALs)) are available to Control Room Operators that an Initiating Condition (IC) has been met and/or exceeded.

   A. This goal should allow time for determination of indications (leak rate, etc.) and detailed review of Attachment 1, Emergency Classification Table.

2. Use the best information available when working through the Emergency Classification Table. When confronted with conflicting information for which resolution is not apparent, classify the condition at the highest appropriate emergency class.

3. If, in the judgement of the Nuclear Plant Supervisor (NPS)/Emergency Coordinator (EC), a situation is more serious than indicated by instrument readings or other parameters, Then classify the emergency condition at the more serious level (i.e., at the highest appropriate emergency class).

4. Multiple and Dual Unit Events

CAUTION

There can not be two concurrent declared emergency classes under the St. Lucie Plant Radiological Emergency Plan.

   A. If one Unit is in a classified event and the same or the other Unit enters into an event where the same or lesser Emergency Class would apply, Then a new classification should NOT be declared. The event should be documented on a SNF as "Additional Information or Update" and issued as soon as practicable.

   B. If one Unit is in a classified event and the other Unit enters into a more severe event in which a higher Emergency Class would apply, Then the new classification shall be declared and promptly, within the regulatory time limits, issued to the State, Counties and the NRC.

5.2 Classifying the Event (continued)

5. ¶2 If an EAL was met and the condition completely cleared prior to an emergency classification being declared, Then:

A. Classify the event in accordance with Attachment 1.

B. Termination of the event

1. An event classified as an Unusual Event or Alert may be terminated at the time of declaration by the EC.

2. An event classified as a Site Area Emergency or General Emergency may only be downgraded and/or terminated by the Recovery Manager (RM).

END OF SECTION 5.2

/R6

5.3 §1 ¶1 Classification of An Event Based On Subsequent Information

1. If subsequent information of a more detailed nature (e.g., sampling results) becomes available after the initial classification has been made, Then reclassify as appropriate.

2. If results of a protracted review (i.e., Engineering Evaluation, CR disposition, etc.) of an event indicate that conditions were met for an Emergency Classification, and the condition has completely cleared prior to recognition of possible classification, Then notify NRC within one hour of discovery of the undeclared event.

   A. Contact Emergency Preparedness for briefing of state and local agencies.

END OF SECTION 5.3

ATTACHMENT 1
EMERGENCY CLASSIFICATION TABLE
(Page 1 of 20)

CAUTION
§2 Section 1.A should not be used for a steam generator tube leak / rupture.

EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.A. ABNORMAL PRIMARY LEAK RATE (Page 1 of 2)	Reactor Coolant System (RCS) Leakage 1. RCS leakage GREATER THAN 10 gpm as indicated by: A. Control Room observation OR B. Inventory balance calculation OR C. Field observation OR D. Emergency Coordinator judgement OR 2. Indication of leaking RCS safety or relief valve which causes RCS pressure to drop below setpoints: - Unit 1 - 1600 psia - Unit 2 - 1736 psia	§1 RCS Leakage GREATER THAN 50 gpm 1. Unisolable RCS leakage as indicated by Charging/letdown mismatch greater than 50 gpm but less than available charging pump capacity. OR 2. Unisolable measured RCS leakage indicating greater than 50 gpm but less than available charging pump capacity.	LOCA GREATER THAN capacity of charging pumps 1. RCS leakage greater than available charging pump capacity occurring with RCS pressure above HPSI shutoff head. OR 2. RCS leakage greater than available makeup occurring with RCS pressure below HPSI shutoff head. OR 3. Loss of RCS subcooled margin due to RCS leakage (saturated conditions). OR 4. Containment High Range Radiation Monitors indicate 7.3 X 10^3 R/hr (If CHRRM inoperable, Post-LOCA monitors indicate between 100 and 1000 mR/hr).	A release has occurred or is in progress resulting in: 1. Containment High Range Radiation monitor greater than 1.46 X 10^5 R/hr (If CHRRM inoperable, Post-LOCA monitors greater than 1000 mR/hr). OR 2. Performance of EPIP-09 (Off-site Dose Calculations) or measured dose rates from off-site surveys indicate site boundary (1 mile) exposure levels have been exceeded as indicated by either A, B, C or D below: A. 1000 mrem/hr (total dose rate) B. 1000 mrem (total dose - TEDE) C. 5000 mrem/hr (thyroid dose rate) D. 5000 mrem (thyroid dose - CDE) (continued on next page)
1.A. ABNORMAL PRIMARY LEAK RATE				

AFTER CLASSIFYING, GO TO EPIP-02, DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.A. ABNORMAL PRIMARY LEAK RATE (Page 2 of 2)				Loss of 2 of the 3 fission product barriers with imminent loss of the third (any two of the following exist and the third is imminent). 1. Fuel element failure (confirmed DEQ I-131 activity greater than 275 µCi/mL). AND 2. LOCA or Tube rupture on unisolable steam generator. AND 3. Containment Integrity Breached. **NOTE** Also refer to Potential Core Melt Event / Class 6.A.

1.A. ABNORMAL PRIMARY LEAK RATE

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.B. ABNORMAL PRIMARY TO SECONDARY LEAK RATE (Page 1 of 2)	RCS PRI/SEC Leakage 1. Measured RCS to secondary leakage exceeds Tech. Spec. limits. AND 2. Secondary plant activity is detected.	Rapid gross failure of one steam generator tube (WITHIN charging pump capacity) with loss of offsite power 1. Measured RCS to secondary leakage greater than Tech. Spec. Limits and within charging pump capacity. AND 2. Secondary plant activity is detected. AND 3. Loss of both Non-Vital 4.16 KV buses. (continued on next page)	Rapid gross failure of steam generator tubes (GREATER THAN charging pump capacity) with a loss of offsite power 1. Measured RCS to secondary leakage is greater than charging pump capacity. AND 2. Secondary plant activity is detected. AND 3. Loss of both Non-Vital 4.16 KV buses. (continued on next page)	Loss of 2 of the 3 fission product barriers with imminent loss of the third (any two of the following exist and the third is imminent). 1. Fuel element failure (confirmed DEQ I-131 activity greater than 275 µCi/mL). AND 2. LOCA or Tube rupture on unisolable steam generator. AND 3. Containment integrity breached. **NOTE** Also refer to Potential Core Melt Event/Class 6.A.

1.B. ABNORMAL PRIMARY TO SECONDARY LEAK RATE

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.B. ABNORMAL PRIMARY TO SECONDARY LEAK RATE (Page 2 of 2)		Rapid failure of steam generator tubes (GREATER THAN charging pump capacity) 1. Measured RCS to secondary leakage greater than charging pump capacity. AND 2. Secondary plant activity is detected.	§2 Rapid failure of steam generator tube(s) (GREATER THAN charging pump capacity) with steam release in progress 1. Measured RCS to secondary leakage greater than charging pump capacity. AND 2. Secondary plant activity is detected. AND 3. Secondary steam release in progress from affected generator (i.e., ADVs, stuck steam safety(s) or unisolable leak.)	

1.B. ABNORMAL PRIMARY TO SECONDARY LEAK RATE

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.C. LOSS OF SECONDARY COOLANT (Page 1 of 2)	Rapid depressurization of secondary plant 1. Rapid drop in either steam generator pressure to less than 600 psia.	Major steam leak with GREATER THAN 10 gpm primary/secondary leakage 1. Rapid drop in either steam generator pressure to less than 600 psia. AND 2. Known pri/sec leak of greater than 10 gpm. AND 3. Secondary plant activity is detected. Total loss of feedwater 1. No main or auxiliary feedwater flow available for greater than 15 minutes when required for heat removal. AND 2. Steam Generator levels are less than 40% wide range.	Major steam leak with GREATER THAN 50 gpm primary/secondary leakage and fuel damage indicated 1. Rapid drop in either steam generator pressure to less than 600 psia. AND 2. Known pri/sec leak of greater than 50 gpm. AND 3. Secondary plant activity is detected. AND 4. Fuel element damage is indicated (Refer to Fuel Element Failure Event/Class 4.A). TLOF with once-through cooling initiated 1. No main or auxiliary feedwater flow available. AND 2. PORV(s) have been opened to facilitate core heat removal.	A release has occurred or is in progress resulting in: 1. Containment High Range Radiation monitor greater than 1.46×10^5 R/hr (If CHRRM inoperable, Post-LOCA monitors greater than 1000 mR/hr). OR 2. Performance of EPIP-09 (Off-site Dose Calculations) or measured dose rates from off-site surveys indicate site boundary (1 mile) exposure levels have been exceeded as indicated by either A, B, C or D below: A. 1000 mrem/hr (total dose rate) B. 1000 mrem (total dose - TEDE) C. 5000 mrem/hr (thyroid dose rate) D. 5000 mrem (thyroid dose-CDE) (continued on next page)
1.C. LOSS OF SECONDARY COOLANT				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1.C. LOSS OF SECONDARY COOLANT (Page 2 of 2)				Loss of 2 of the 3 fission product barriers with imminent loss of the third (any two of the following exist and the third is imminent). 1. Fuel element failure (confirmed DEQ I-131 activity greater than 275 μCi/mL). AND 2. LOCA or Tube rupture on unisolable steam generator. AND 3. Containment Integrity Breached. **NOTE** Also refer to Potential Core Melt Event/Class 6.A.

1.C. LOSS OF SECONDARY COOLANT

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2.A. UNCONTROLLED EFFLUENT RELEASE	Radiological effluent limits exceeded 1. Plant effluent monitor(s) exceed alarm setpoint(s). AND 2. Confirmed analysis results for gaseous or liquid release which exceeds ODCM limits. **NOTE** If analysis is not available within one hour and it is expected that release is greater than ODCM limit, classify as UNUSUAL EVENT.	A release has occurred or is in progress that is 10 times the effluent limit 1. Plant effluent monitor(s) significantly exceed alarm setpoints. AND 2. Confirmed analysis results for gaseous or liquid release which exceeds 10 times ODCM limits. **NOTE** If analysis is not available within one hour and it is expected that release is equal to or greater than 10 times ODCM limit, classify as ALERT.	§2 A release has occurred or is in progress resulting in: 1. Containment High Range Radiation Monitor greater than 7.3 X 10^3 R/hr (Post-LOCA monitors indicate between 100 and 1000 mR/hr, if CHRRM inoperable). OR 2. Measured Dose Rates or Offsite Dose Calculation (EPIP-09) worksheet values at one mile in excess of: A. 50 mrem/hr (total dose rate) or 250 mrem/hr (thyroid dose rate) for 1/2 hour. OR B. 500 mrem/hr (total dose rate) or 2500 mrem/hr (thyroid dose rate) for two minutes at one mile.	A release has occurred or is in progress resulting in: 1. Containment High Range Radiation monitor greater than 1.46 X 10^5 R/hr (If CHRRM inoperable, Post-LOCA monitors greater than 1000 mR/hr). OR 2. Performance of EPIP-09 (Off-site Dose Calculations) or measured dose rates from off-site surveys indicate site boundary (1 mile) exposure levels have been exceeded as indicated by either A, B, C or D below: A. 1000 mrem/hr (total dose rate) B. 1000 mrem (total dose - TEDE) C. 5000 mrem/hr (thyroid dose rate) D. 5000 mrem (thyroid dose-CDE)

ODCM - refers to Chemistry Procedure C-200, Offsite Dose Calculation Manual (ODCM)

2.A. UNCONTROLLED EFFLUENT RELEASE

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2.B. HIGH RADIATION LEVELS IN PLANT		High radiation levels or high airborne contamination which indicates a severe degradation in the control of radioactive materials 1. Any valid area monitor alarm from an unplanned source with meter near or greater than full scale deflection (10^3 mR/hr). OR 2. Unexpected plant iodine or particulate airborne concentration of 1000 DAC as seen in routine surveying or sampling. OR 3. Unexpected direct radiation dose rate reading or unexpected airborne radioactivity concentration from an unplanned source in excess of 1000 times normal levels.		

2.B. HIGH RADIATION LEVELS IN PLANT

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
3. FIRE	§2 Uncontrolled fire within the Power Block lasting more than 10 minutes.	Uncontrolled fire 1. Potentially affecting safety systems. AND 2. Requiring off-site support in the opinion of the NPS/EC.	§2 Fire compromising the function of safety systems (e.g., both trains rendered inoperable).	**NOTE** Refer to Potential Core Melt Event/Class 6.A.

NOTE
§2 Explosion is defined as a rapid chemical reaction resulting in noise, heat and rapid expansion of gas.

EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
EXPLOSION	Occurrence of an explosion within the Owner Controlled Area.	§2 Damage to structures/components in the Protected Area by explosion which affects plant operation.	§2 Severe damage to safe shutdown equipment from explosion (e.g., both trains rendered inoperable.	

3. FIRE

EXPLOSION

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4.A. FUEL ELEMENT FAILURE	Fuel element damage 1. Process monitors or area radiation surveys indicate increased letdown activity AND 2. Confirmed RCS sample indicating: A. Coolant activity greater than the Tech Spec limit for iodine spike (Tech Spec Figure 3.4-1.). OR B. Coolant activity greater than 100/Ē µCi/gram specific activity. **NOTE** If analysis is not available within one hour and it is expected that activity is greater than Tech Spec limit, classify as UNUSUAL EVENT.	Fuel element failure 1. Process monitors or area radiation surveys indicate increased letdown activity and confirmed RCS Samples indicating DEQ I-131 activity greater than or equal to 275 µCi/mL. **NOTE** If analysis is not available within one hour and it is expected that RCS activity for DEQ I-131 is greater than 275 µCi/mL, classify as an ALERT.	Fuel element failure with inadequate core cooling 1. RCS DEQ I-131 activity greater than or equal to 275 µCi/mL. AND 2. Highest CET per core quadrant indicates greater than 10°F superheat or 700°F.	A release has occurred or is in progress resulting in: 1. Containment High Range Radiation monitor greater than 1.46 X 10^5 R/hr (If CHRRM inoperable, Post-LOCA monitors greater than 1000 mR/hr). OR 2. Performance of EPIP-09 (Off-site Dose Calculations) or measured dose rates from off-site surveys indicate site boundary (1 mile) exposure levels have been exceeded as indicated by either A, B, C or D below: A. 1000 mrem/hr (total dose rate) B. 1000 mrem (total dose - TEDE) C. 5000 mrem/hr (thyroid dose rate) D. 5000 mrem (thyroid dose - CDE)
4.A FUEL ELEMENT FAILURE				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4.B. FUEL HANDLING ACCIDENT		Fuel handling accident which results in the release of radioactivity to Containment or Fuel Handling Building: 1. NPS/EC determines that an irradiated fuel assembly may have been damaged. AND 2. Associated area or process radiation monitors are in alarm.	§2 Major damage to irradiated fuel in Containment or Fuel Handling Building 1. Affected area radiation monitor greater than 1000 mrem/hr. AND 2. Damage to more than one irradiated fuel assembly. OR Major damage resulting from uncovering of one or more irradiated fuel assemblies in the Spent Fuel Pool.	

4.B. FUEL HANDLING ACCIDENT

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
5.A. EARTHQUAKE	§2 A confirmed earthquake has occurred 1. A confirmed earthquake has been experienced within the Owner Controlled Area. OR 2. ¶4 An earthquake is detected by plant seismic monitor instruments or other means.	§2 A confirmed earthquake has occurred. 1. A confirmed earthquake has occurred which registered GREATER THAN 0.05g within the Owner Controlled Area. OR 2. A confirmed earthquake has occurred that could or has caused trip of the turbine generator or reactor.	§2 A confirmed earthquake has occurred. 1. A confirmed earthquake has occurred which registered GREATER THAN 0.1g within the Owner Controlled Area and the plant not in Cold Shutdown. OR 2. A confirmed earthquake has occurred that has caused loss of any safety system function (e.g., both trains inoperable).	**NOTE** Refer to Potential Core Melt Event / Class 6.A.
5.B. HURRICANE	Hurricane Warning 1. Confirmed hurricane warning is in effect.	Hurricane warning with winds near design basis 1. Confirmed hurricane warning is in effect and winds are expected to exceed 175 mph within the Owner Controlled Area. **NOTE** At FPL's request, NOAA will provide an accurate projection of wind speeds onsite 24 hours prior to the onset of hurricane force winds. If that projection is not available within 12 hours of entering into the warning, classify the event using current track and wind speeds to project onsite conditions. For example, projected onsite wind speed would be less than maximum hurricane wind speed if the track is away from PSL.	Hurricane warning with winds GREATER THAN design basis 1. Plant not at cold shutdown. AND 2. Confirmed hurricane warning is in effect and winds are expected to exceed 194 mph within the Owner Controlled Area. **NOTE** At FPL's request, NOAA will provide an accurate projection of wind speeds onsite 24 hours prior to the onset of hurricane force winds. If that projection is not available within 12 hours of entering into the warning, classify the event using current track and wind speeds to project onsite conditions. For example, projected onsite wind speed would be less than maximum hurricane wind speed if the track is away from PSL.	**NOTE** Refer to Potential Core Melt Event / Class 6.A.

5.A. EARTHQUAKE
5.B. HURRICANE

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
5.C. TORNADO	Notification of a tornado sighted in the Owner Controlled Area	§2 Any tornado striking the Power Block.		**NOTE** Refer to Potential Core Melt Event / Class 6.A.
5.D. ABNORMAL WATER LEVEL	Abnormal water level conditions are expected or occurring 1. Low intake canal level of -10.5 ft. MLW for 1 hour or more. OR 2. Visual sightings by station personnel that water levels are approaching storm drain system capacity.	Flood, low water, hurricane surge or other abnormal water level conditions 1. The storm drain capacity is exceeded during hurricane surge or known flood conditions. OR 2. Low intake canal level of -10.5 ft. MLW for 1 hour or more with emergency barrier valves open.	Flood, low water, hurricane surge or other abnormal water level conditions causing failure of vital equipment 1. Flood/surge water level reaching elevation +19.5 ft. (turbine building / RAB ground floor). OR 2. Low intake canal level has caused the loss of all ICW flow.	

5.C. TORNADO

5.D. ABNORMAL WATER LEVEL

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NOTE
Activation of the Emergency Response Facilities does not require declaration of an emergency or entry into a specific emergency classification.

EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
6.A. INCREASED AWARENESS OR POTENTIAL CORE MELT (Page 1 of 2)	Emergency Coordinator's judgement that plant conditions exist which warrant increased awareness on the part of the operating staff and/or local authorities. 1. The plant is shutdown under abnormal conditions (e.g., exceeding cooldown rates or primary system pipe cracks are found during operation). OR 2. Any plant shutdown required by Technical Specifications in which the required shutdown is not reached within action limits.	§2 Emergency Coordinator's judgement that plant conditions exist which have a potential to degrade the level of safety at the plant.	§2 Emergency Coordinator's judgement that plant conditions exist which are significantly degrading in an uncontrollable manner.	§2 Emergency Coordinator's judgement that plant conditions exist that make release of large amounts of radioactivity in a short period appear possible or likely. (Any core melt situation.) 1. LOCA with failure of ECCS leading to severe core degradation or melt. OR 2. LOCA with initially successful ECCS and subsequent failure of containment heat removal systems for greater than 2 hours. OR 3. Total loss of feedwater followed by failure of once-through-cooling (ECCS) to adequately cool the core. OR 4. Failure of off-site and on-site power along with total loss of feedwater makeup capability for greater than 2 hours. OR 5. ATWS occurs which results in core damage or causes failure of core cooling and make-up systems. OR 6. Any major internal or external event (e.g., fire, earthquake or tornado substantially beyond design basis) which in the ECs opinion has or could cause massive damage to plant systems resulting in any of the above. (continued on next page)
6.A. INCREASED AWARENESS OR POTENTIAL CORE MELT				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
6.A. INCREASED AWARENESS OR POTENTIAL CORE MELT (Page 2 of 2)				**NOTES** 1. Most likely containment failure mode is melt-through with release of gases only. Quicker releases are expected for failure of containment isolation system. 2. General Emergency must be declared for the above listed events. The likelihood of corrective action (repair of AFW pump, etc.) should not be considered.
6.A. INCREASED AWARENESS OR POTENTIAL CORE MELT				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
7.A. LOSS OF POWER	Loss of off-site power or loss of all on-site AC power capability. 1. Loss of off-site AC power. OR 2. Loss of capability to power at least one vital 4.16 kv bus from any available emergency diesel generator.	§2 Station Blackout (Total Loss of AC) 1. Loss of off-site AC power. AND 2. Failure of both emergency diesel generators to start or load. --- Loss of all on-site DC power 1. Drop in A and B DC bus voltages to less than 70 VDC.	§2 Station Blackout (Total Loss of AC) for GREATER THAN 15 minutes 1. Loss of offsite AC power. AND 2. Sustained failure of both emergency diesel generators to start or load. AND 3. Failure to restore AC power to at least one vital 4.16 kv bus within 15 minutes. --- Loss of all vital on-site DC for greater than 15 minutes 1. Sustained drop in A and B DC bus voltages to 70 VDC for greater than 15 minutes.	**NOTE** Refer to Potential Core Melt Event / Class 6.A.

7.A. LOSS OF POWER

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
8.A. LOSS OF PLANT CONTROL FUNCTIONS		Loss of Plant Control Functions 1. Complete loss of any function needed for plant cold shutdown. OR 2. Failure of the Reactor Protection System to bring the reactor subcritical when needed. OR 3. Control Room is evacuated (for other than drill purposes) with control established locally at the Hot Shutdown Control Panel. Loss of Shutdown Cooling 1. Complete loss of functions needed to maintain cold shutdown. A. Failure of shutdown cooling systems, resulting in loss of cold shutdown conditions. AND B. RCS subcooling can NOT be maintained greater than 0°F.	Critical Loss of Plant Control Functions 1. Loss of any function or system which, in the opinion of the Emergency Coordinator, precludes placing the plant in Hot Shutdown. OR 2. Failure of the RPS to trip the reactor when needed and operator actions fail to bring the reactor subcritical. OR 3. Control Room is evacuated (for other than drill purposes) and control cannot be established locally at the Hot Shutdown Control Panel within 15 minutes.	**NOTE** Refer to Potential Core Melt Event / Class 6.A.
8.A. LOSS OF PLANT CONTROL FUNCTIONS				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
8.B. LOSS OF ALARMS / COMMUNICATION / MONITORING	§2 Significant loss of effluent monitoring capability, communications, indication and alarm panels, etc., which impairs ability to perform accident or emergency assessment. 1. Loss of effluent or radiological monitoring capability requiring plant shutdown. OR 2. Loss of all primary and backup communication capability with offsite locations. OR 3. Unplanned loss of most (greater than 75%) or all Safety System annunciators for greater than 15 minutes.	§2 Loss of alarms 1. Unplanned loss of most (greater than 75%) or all safety system annunciators. AND 2. Plant transient in progress.	Loss of alarms/monitoring 1. Inability to monitor* a significant transient in progress. *Monitoring means loss of ERDADS, QSPDS and/or the inability to determine any one of the following: reactivity control, core cooling, RCS status or containment integrity.	
8.B. LOSS OF ALARMS / COMMUNICATION / MONITORING				

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
9.A. AIRCRAFT / MISSILE	Unusual aircraft activity 1. Aircraft crash in the Owner Controlled Area or unusual aircraft activity over facility that in the opinion of the NPS/EC, could threaten the safety of the plant or personnel.	§2 Aircraft/missile impact 1. Aircraft crash into the Power Block. OR 2. Visual or audible indication of missile impact on the Power Block.	§2 Damage to vital systems from aircraft/missiles 1. Aircraft crash into the Power Block damaging vital plant systems. OR 2. Damage resulting in loss of safe shutdown equipment from any missile.	
9.B. TURBINE FAILURE	Turbine rotating component failure causing rapid plant shutdown.	Visual indication that the turbine casing has been penetrated by blading.		
9.C. TOXIC OR FLAMMABLE GAS	Unplanned/uncontrolled toxic or flammable gas release in the Owner Controlled Area that could affect plant/personnel safety.	Entry of toxic or flammable gas into areas potentially affecting plant operation.	§2 Toxic or flammable gas has diffused into vital areas compromising the function of safety related equipment (e.g., both trains rendered inoperable).	
9.A. AIRCRAFT / MISSILE				
9.B. TURBINE FAILURE				
9.C. TOXIC OR FLAMMABLE GAS				

AFTER CLASSIFYING, GO TO EPIP-02, DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR

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EVENT/CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
10. SECURITY THREAT	A SECURITY ALERT has been called by the Security Force in response to one or more of the items listed below. 1. Bomb threat 2. Attack threat 3. Security threat 4. Protected Area intrusion attempt 5. Sabotage attempt 6. Internal disturbance 7. Civil disturbance 8. Vital Area intrusion 9. Security Force strike 10. Credible site-specific Security threat notification	A SECURITY EMERGENCY has been called by the Security Force as defined in the Safeguards Contingency Plan.	A SECURITY EMERGENCY involving imminent occupancy of the control room or other area(s) vital to the operation of the reactor as defined in the Safeguards Contingency Plan.	A successful takeover of the plant including the Control Room or any other area(s) vital to the operation of the reactor (as per the Security Plan).

10. SECURITY THREAT

AFTER CLASSIFYING, GO TO EPIP-02, DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR

FPL	**ST. LUCIE PLANT** **EMERGENCY PLAN IMPLEMENTING PROCEDURE** SAFETY RELATED	Procedure No. **EPIP-02** Current Revision No. **12** Effective Date **11/17/03**

Title:

DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR

Responsible Department: **EMERGENCY PLANNING**

REVISION SUMMARY:

Revision 12 - Incorporated PCR 03/1634 for PMAI 03-04-082 to incorporate shift communicator position. (A. Terezakis, 08/06/03)

AND

Incorporated PCR 03-0581 for CR 03-0246, CR 03-0953 to remove navigational degrees and supplement instructions regarding security events. (J. R. Walker, 07/18/03)

Revision 11 - Incorporated PCR #03-0024 for MAI MA 02-12-041 to clarify guidance regarding multiple events, add step to SAE checklist, and reorder plant announcement step. (J. Walker, 02/07/03)

Revision 10 - Clarified duties of DCS as phonetalker. Clarified checklists regarding steps not necessary in TSC. Made editorial/administrative changes. (J. R. Walker, 07/26/02)

Revision 9 – THIS PROCEDURE HAS BEEN COMPLETELY REWRITTEN. Added responsibility for information services to update checklists in CRs, deleted basis of exposure guidelines, added statement on EC coverage during a prolonged event, deleted emergency declaration checklist, revised emergency class checklists, added reference to EPIP-07 and made editorial and administrative changes. (J. R. Walker, 06/11/01)

Revision 8 - Revised off-site assembly area to Jensen public beach parking area in accordance with revision 38 of the E-Plan and made administrative changed. (J. R. Walker, 10/13/00)

Revision	FRG Review Date	Approved By	Approval Date
0	12/15/97	J. Scarola Plant General Manager	12/15/97
Revision	**FRG Review Date**	**Approved By**	**Approval Date**
12	07/31/03	R. E. Rose Plant General Manager	08/06/03
		N/A Designated Approver	
		N/A Designated Approver (Minor Correction)	

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1.0 PURPOSE

1.1 This procedure provides guidance and instructions to be followed by the Emergency Coordinator when an emergency occurs that requires the implementation of the Radiological Emergency Plan for St. Lucie Plant.

2.0 REFERENCES / RECORDS REQUIRED / COMMITMENT DOCUMENTS

NOTE

One or more of the following symbols may be used in this procedure:

§ Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, Operating Experience, License Renewal, etc. and shall NOT be revised without Facility Review Group review and Plant General Manager approval.

¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.

Ψ Indicates a step that requires a sign off on an attachment.

2.1 References

1. St. Lucie Plant Updated Final Safety Analysis Report (UFSAR) Unit 1 and Unit 2 (Section 9.5.A.7.2)
2. §1 St. Lucie Plant Radiological Emergency Plan (E-Plan)
3. St. Lucie Plant Physical Security Plan
4. St. Lucie Plant Safeguards Contingency Plan
5. E-Plan Implementing Procedures (EPIP 00-13)
6. 10 CFR 50, Domestic Licensing of Production and Utilization Facilities.
7. NUREG/BR-0150, Vol. 1, Response Technical Manual (USNRC).
8. NUREG-0654, FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.
9. EPA 400-R-92-001, Manual of Protective Actions Guides and Protective Actions for Nuclear Incidents, October, 1991.

2.2 Records Required

¶10 A copy of the checklists or data generated by this procedure shall be maintained in the plant files in accordance with QI-17-PSL-1, Quality Assurance Records. Records include:

1. Emergency Class Checklists

2.3 Commitment Documents

1. ¶1 PMAI PM96-04-165, "ITR 96-006" (Unusual Event Declared Due to Dropped Rod)
2. ¶2 NRC Inspection Report 91-01, Closure of IFIs 89-31-03 and 89-31-01
3. ¶3 PMAI PM96-09-185, Condition Report CR-96-1750 (Off-site Notification Using Commercial Phone)
4. ¶5 PMAI PM96-05-233, Off-site Notification Process.
5. ¶6 Condition Report CR 96-2389, Off-site Dose Calculations.
6. ¶7 Condition Report CR 98-1536, EC Responsibilities Remain in the Control Room.
7. ¶8 PMAI PM98-09-006, Control of NLOs Under E-Plan.
8. ¶9 Condition Report CR 99-1406, Field Operator Dosimetry Under E-Plan.
9. ¶10 PMAI PM99-10-191, Condition Report CR 99-1656 (Quality Records, Downpower Guidance Due to Hurricanes).
10. ¶11 PMAI PM99-10-142, Condition Report CR 99-1647 (EC Turnover).
11. ¶12 PMAI PM99-09-016, (PARs Based on FMT Data, Completion of NRC Notification Form).
12. ¶13 PMAI PM00-01-043, Gai-Tronics E-Plan Alarm.
13. ¶14 PMAI PM00-03-122, Early Activation of ERFs.
14. ¶15 Condition Report CR 02-0333, Role of the Duty Call Supervisor.
15. ¶16 NRC Interim Compensatory Measures (ICM), 25 February, 2002 (Response to Item B.5.d) /R12
16. ¶17 Condition Report CR 03-0246, (NRC Recommended Procedure Improvement Regarding Security Events) /R12

3.0 RESPONSIBILITIES

3.1 The Nuclear Plant Supervisor (NPS) and the shift operating staff represent the first line of response to any developing emergency condition. The primary responsibility of the NPS is to control the condition as well as possible.

3.2 The NPS upon declaration of an emergency classification becomes the Emergency Coordinator (EC). The NPS remains the EC until the position is turned over.

Specific Responsibilities of the EC are:

Direction of the on-site emergency organization to bring the emergency under control.

Notification of off-site agencies within specific time limits as mandated by regulations.

Changes in Emergency Classification based on changing conditions.

Protective Action Recommendations (PARs) until turnover to the Recovery Manager.

Interfaces with the Nuclear Regulatory Commission (NRC) Reactor Safety Operations Coordinator (RSOC) when the NRC site team arrives at the TSC.

3.3 Information Services maintains user copies, in the Unit 1 and Unit 2 Control Rooms, of the following checklists used for implementing the Emergency Plan:

- Unusual Event Declaration Checklist
- Alert Declaration Checklist
- Site Area Emergency Declaration Checklist
- General Emergency Declaration Checklist

4.0 DEFINITIONS

4.1 §1 **Duty Call Supervisor (DCS)** – The Duty Call Supervisor is a designated and trained supervisor assigned from the nuclear plant staff to provide 24-hour response to any emergency upon notification by the Nuclear Plant Supervisor. The Duty Call Supervisor (DCS) is responsible for notifying the Emergency Response Organization and, as requested, Plant management in the event of an emergency.

4.2 **Owner Controlled Area Evacuation** (= Site Evacuation) - The evacuation from the owner controlled area of all personnel except those required to place the plant in a safe condition, the Emergency Response Organization (ERO), and Security personnel to fulfill responsibilities for evacuation.

4.3 **Release** (during any declared emergency)

1. Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.

OR

2. Health Physics detecting airborne radioactivity levels in excess of 25% derived air concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

4.4 **Shift Communicator** - a specific shiftly designated individual trained and qualified to assist the Nuclear Plant Supervisor / Emergency Coordinator in the control room in making emergency off-site notifications, and performing other activities as directed. /R12

5.0 INSTRUCTIONS

5.1 General Overview

1. ¶7,11 Upon Declaration of an emergency classification the NPS becomes the EC.

   To ensure access to the EC for direction and control decisions and so that the responsibilities of the position can be successfully completed, the EC position shall remain, initially in the affected Control Room and then in the Technical Support Center (TSC), when it goes operational.

   Prior to the TSC being operational or in cases when there is a prolonged event such as a hurricane, the duties and responsibilities of the EC, while a Control Room position, may be turned over to another qualified EC:

CAUTION

There can NOT be two concurrent declared emergency classes under the St. Lucie Plant Radiological Emergency Plan.

- If the site is in a dual Unit event, the EC should locate in the Unit 1 Control Room (due to proximity to the TSC). If both units are experiencing independent and classifiable conditions, the EC should locate in the Unit Control Room with the [highest] classified event.

   If the TSC is activated, Then the EC position is turned over to an EC qualified member of plant management and the position relocated to the TSC. The prospective EC receives a turnover (refer to Attachment 3, Turnover Guidelines) from the Control Room EC and then reports to the TSC. Following verification of TSC operational readiness, the prospective EC accepts EC responsibility from the Control Room EC. The TSC EC may temporarily turnover responsibility to the TSC OPS Coordinator as the need arises.

2. To meet the above responsibilities, plus others described in this procedure, the EC will likely need to delegate many tasks. Although delegated, the completion of these tasks is still the responsibility of the EC.

   The EC shall not delegate the following responsibilities prior to Emergency Operations Facility (EOF) being declared operational:

   A. Classification of the emergency.

   B. The decision to notify state and local authorities and the content of those notifications.

5.1 General Overview (continued)

2. (continued)

   C. Recommendation of protective actions for the public.

   Once the EOF is operational and proper turnover has been conducted, the Recovery Manager (RM) will assume responsibility for off-site notifications to the state and local authorities and for recommending off-site protective actions.

3. Order of Succession

   If the NPS is incapacitated, Then the EC shall be (in order of succession):

   A. Assistant Nuclear Plant Supervisor (ANPS) (from the affected unit)

   B. Nuclear Watch Engineer (NWE)

   C. Any other member of the plant staff with an active SRO license.

4. Watch Relief

   A. The EC shall grant permission for watch relief, including his/her own, only when it is safe in his/her judgement to do so.

5. The Emergency Coordinator (EC) shall consider plant and radiological conditions as they relate to the emergency prior to ordering an evacuation and / or activation of the Emergency Response Organization (ERO). As conditions warrant, the EC may delay, postpone or institute special arrangements concerning, but not limited to:

   A. Emergency Response Facility (ERF) activation

   B. Local or Site Evacuation

   C. Protected Area (PA) and / or Radiation Controlled Area (RCA) access

   D. Operator field activities

   E. Unit shutdown

/R12

5.1 General Overview (continued)

6. Some examples of special circumstances and considerations are as follows:

A. Radiological Conditions

1. Duration of release ("puff" versus prolonged)

2. Meteorological conditions

3. Evacuation route availability

4. Sheltering

5. Route to ERFs

6. Plant conditions

7. Other information pertinent to radiation protection considerations

B. ¶16,17 Security Event

1. Site Security and Local Law Enforcement Agencies (LLEA) will take the lead in response to a Security Event in accordance with the Security Plan.

2. Security events when known hazards or dangers (e.g., armed intruders, bomb threats, etc.) are perceived, consider:

a. Location of intruders

b. Bomb threat location

c. Modification of plant announcements if it is determined that such announcements may cause intruders to panic or make them aware of plant / Security personnel locations and / or responses

d. Directing ERO members to alternate locations

e. Special instructions for non-essential plant personnel regarding movement on the plant site, sheltering or evacuation

/R12

5.1 General Overview (continued)

7. ¶14 Early Activation of Emergency Response Facilities

It may be useful to have technical and/or operational support available early in an emergency prior to when the Technical Support Center (TSC), Operational Support Center (OSC), or Emergency Operations Facility (EOF) is required to be operational. Activation of any of these facilities does not require declaration of an emergency class or entry into a specific emergency classification. If early activation of one or more of the facilities is desired, then follow these guidelines:

A. This is an option during normal working hours only.

B. A page announcement should be made to request that appropriate Emergency Response Organization personnel to report to the [identify what facility/facilities is/are to be activated early].

C. Turnover of EC responsibilities is done in accordance with Step 5.1.1, above.

D. The E-Plan Activation Alarm is used only when the Emergency Response Facilities (ERFs) are to be activated in accordance with the requirements of the Emergency Plan (i.e., at the Alert or higher emergency level) and is provided for in the checklist included in this procedure.

E. Staff augmentation due to actual facility activation is to be done in accordance with the Alert Declaration Checklist, Site Area Emergency Declaration Checklist or the General Emergency Declaration Checklist which are part of this procedure.

/R12

5.1 General Overview (continued)

8. Severe Weather Considerations

¶10 If a hurricane warning is in effect, and either one or both Unit(s) is/are in Mode 1, 2 or 3, Then use the following criteria for unit shutdown:

> **NOTE**
> Sustained hurricane force winds are sustained winds of 74 mph (64 kt or 119 kph) or greater.

A. For storms projected to reach a Category 1 or 2, the Unit(s) shall be placed in HOT STANDBY (Mode 3) or below at least two (2) hours before the projected onset of sustained hurricane force winds within the Owner Controlled Area and both Units shall remain off-line for the duration of the hurricane force winds (or restoration of reliable offsite power).

B. For storms projected to reach Category 3, 4 and 5 prior to landfall, the Units shall be shut down to a temperature less than 350 degrees T ave. at least two (2) hours before the projected onset of sustained hurricane force winds within the Owner Controlled Area and both Units shall remain off-line for the duration of the hurricane force winds (or restoration of reliable offsite power).

C. Establish an acceptable update frequency with state and local officials.

9. Drill Messages

A. During exercises, drills, or tests, **ALL MESSAGES** shall begin and end with **THIS IS A DRILL** or **THIS IS AN EXERCISE** or **THIS IS A TEST.**

END OF SECTION 5.1

5.2 Unusual Event Declaration Checklist **TIME / INITIAL**

Date: ____/____/____

CAUTION
Notification to the State Warning Point (SWP) shall occur within 15 minutes of declaration of the emergency classification.

NOTE

- Steps should be performed in the order presented. When conditions warrant, steps may be performed out of sequence.
- PA announcements are provided as a guideline. Actual announcements may vary from the text provided.
- Not Applicable (N/A) may be used for tasks / steps previously accomplished / satisfied.

1. Determine the following:

   A. Shift Technical Advisor (STA) present Y / N

   B. Duty Call Supervisor (DCS) present Y / N

   C. Shift Communicator present Y / N /R12

   D. Wind direction (from) ____°

NOTE
During any declared emergency, a release is occurring if one of the following is true:

- Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.

OR

- Health Physics detecting airborne radioactivity levels in excess of 25 percent Derived Air Concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

   E. Release in progress Y / N ____/____

2. Mobilize emergency team personnel (i.e., Fire Team, First Aid Team) as required using Gai-tronics and boost function. ____/____

5.2 Unusual Event Declaration Checklist (continued) TIME / INITIAL

3. The NPS shall declare the emergency to the Control Room staff and formally announce that he / she is the Emergency Coordinator. ___/___

4. Notify plant personnel using Gai-tronics and boost function as follows:

> "Attention all plant personnel, Unit 1 / 2 has declared an UNUSUAL EVENT. All personnel are to limit radio and phone use and listen for future instructions and further information."

Repeat the announcement. ___/___

5. Notify the Shift Technical Advisor, Duty Call Supervisor, and the Shift Communicator, as appropriate to report to the Control Room using Gai-tronics and boost function. (N/A if already performed) /R12

> "Shift Technical Advisor report to the Unit 1 / 2 Control Room." ___/___

> "Duty Call Supervisor report to the Unit 1 / 2 Control Room." ___/___

> "Shift Communicator report to the Unit 1 / 2 Control Room." ___/___ /R12

6. ¶6 If a release of radioactive material has occurred or is in progress, Then notify Chemistry to promptly implement EPIP-09, Off-site Dose Calculations, and report the results to the Emergency Coordinator (EC). ___/___

7. If a Chemist is unavailable, Then call-out a Chemist (this may be accomplished by the DCS). ___/___

8. If evacuation of an area is necessary (refer to Attachment 2, Criteria for Evacuation), Then initiate a local evacuation. ___/___

9. ¶15 Complete required notifications in accordance with Appendix A, Notifications from the Affected Control Room, in EPIP-08, Off-site Notifications and Protective Action Recommendations. The DCS may be utilized as a phonetalker.

State Warning Point ___/___

NRC ___/___

5.2 Unusual Event Declaration Checklist (continued) TIME / INITIAL

10. Ensure notification of the following: (this may be accomplished by the DCS)

Plant Management ___/___

Security ___/___

Nuclear Division Duty Officer (NDDO) ___/___

11. Utilize Attachment 3, Turnover Guidelines when relinquishing duties to the oncoming EC. ___/___

NOTE

¶2 New notification forms shall be completed for all updates.

12. ¶15 If a State / Local notification frequency has been negotiated, Then provide an update, as necessary utilizing a new notification form. The DCS may be utilized as a phonetalker. (Repeat as necessary) ___/___

___/___

___/___

___/___

___/___

13. If the event can be terminated, Then complete the notification forms (State, NRC) and notify the following: (this may be accomplished by the DCS)

State Warning Point (SWP) ___/___

Plant Management ___/___

Security ___/___

NDDO ___/___

NRC ___/___

14. UNUSUAL EVENT Declaration Checklist complete (emergency upgraded or event terminated). ___/___

END OF SECTION 5.2

5.3 Alert Declaration Checklist

TIME / INITIAL

Date: ____/____/____

CAUTION

Notification to the State Warning Point (SWP) shall occur within 15 minutes of declaration of the emergency classification.

NOTE

- Steps should be performed in the order presented. When conditions warrant, steps may be performed out of sequence.
- PA announcements are provided as a guideline. Actual announcements may vary from the text provided.
- For assistance with exposure control, refer to:
  - Attachment 4, Field Operator Re-entry Guidelines
  - Attachment 5, Exposure Limits for Emergency Response Personnel
- Not Applicable (N/A) may be used for tasks / steps previously accomplished / satisfied.

1. Determine the following:

A. Shift Technical Advisor (STA) present Y / N

B. Duty Call Supervisor (DCS) present Y / N

C. Shift Communicator present Y / N /R12

D. Wind direction (from) ____°

5.3 Alert Declaration Checklist (continued) **TIME / INITIAL**

1. (continued)

> **NOTE**
>
> During any declared emergency, a release is occurring if one of the following is true:
>
> - Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.
>
> OR
>
> - Health Physics detecting airborne radioactivity levels in excess of 25 percent Derived Air Concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

E. Release in progress Y / N

F. E-Plan Alarm sounded and Emergency Response Facilities (ERFs) activated Y / N ___/___

2. Mobilize emergency team personnel (i.e., Fire Team, First Aid Team) as required using Gai-tronics and boost function. ___/___

3. The NPS shall declare the emergency to the Control Room staff and formally announce that he / she is the Emergency Coordinator. ___/___

4. ¶2 If a release of radioactive material is in progress, Then review personnel access with Health Physics personnel and notify Security personnel with any special instructions. ___/___

5. ¶13 Sound the Emergency Plan (E-Plan) Activation Alarm. ___/___

5.3 Alert Declaration Checklist (continued) **TIME / INITIAL**

6. Notify plant personnel using Gai-tronics and boost function as follows:

   "Attention all plant personnel, Unit 1 / 2 has declared an ALERT. All emergency response personnel report at once to your assigned emergency response facility."

   "All non-emergency response personnel report to your normal work location or contact your supervisor. Please limit radio and phone use and listen for further instructions and further information."

   Repeat the announcement. ___/___

7. Notify the Shift Technical Advisor, Duty Call Supervisor and the Shift Communicator, as appropriate to report to the Control Room using Gai-tronics and boost function. (N/A if already performed) /R12

   "Shift Technical Advisor report to the Unit 1 / 2 Control Room." ___/___

   "Duty Call Supervisor report to the Unit 1 / 2 Control Room." ___/___

   "Shift Communicator report to the Unit 1 / 2 Control Room." ___/___ /R12

8. Initiate the call-out process in accordance with EPIP-03, Emergency Response Organization Notification / Staff Augmentation (this may be accomplished by the DCS). ___/___

9. ¶6 If a release of radioactive material has occurred or is in progress, Then notify Chemistry to promptly implement EPIP-09, Off-site Dose Calculations, and report the results to the Emergency Coordinator (EC). ___/___

10. If a Chemist is unavailable, Then call-out a Chemist (this may be accomplished by the DCS). ___/___

11. If evacuation of an area is necessary (refer to Attachment 2, Criteria for Evacuation), Then initiate a local evacuation. ___/___

5.3 Alert Declaration Checklist (continued) TIME / INITIAL

12. ¶15 Complete required notifications in accordance with Appendix A, Notifications from the Affected Control Room, in EPIP-08, Off-site Notifications and Protective Action Recommendations. The DCS may be utilized as a phonetalker.

State Warning Point ___/___

NRC ___/___

13. Ensure notification of the following: (this may be accomplished by the DCS)

Plant Management ___/___

Security ___/___

Nuclear Division Duty Officer (NDDO) ___/___

14. Initiate the Operations Department Accountability Aid for both Unit 1 and Unit 2 and provide this list to the TSC when requested. (this may be accomplished by the DCS) ___/___

15. ¶9 Ensure Operations field personnel return to their assigned Control Room and obtain emergency Electronic Personal Dosimetry (EPD) from the HP Emergency Kit. ___/___

16. Utilize Attachment 3, Turnover Guidelines when relinquishing duties to the oncoming EC. ___/___

NOTE
¶2 New notification forms shall be completed for all updates.

17. ¶15 If State / Local notification has not been completed in the last 60 minutes, Then provide a routine update utilizing a new notification form. The DCS may be utilized as a phonetalker. (Repeat as necessary) ___/___

___/___

___/___

___/___

___/___

5.3 Alert Declaration Checklist (continued) TIME / INITIAL

18. If the event can be terminated, Then complete the notification forms (State, NRC) and notify the following: (this may be accomplished by the DCS)

State Warning Point (SWP) ___/___

Plant Management ___/___

Security ___/___

NDDO ___/___

NRC ___/___

19. ALERT Declaration Checklist complete (emergency upgraded or event terminated). ___/___

END OF SECTION 5.3

5.4 Site Area Emergency Declaration Checklist **TIME / INITIAL**

Date: ____/____/____

CAUTION

Notification to the State Warning Point (SWP) shall occur within 15 minutes of declaration of the emergency classification.

NOTE

- Steps should be performed in the order presented. When conditions warrant, steps may be performed out of sequence.
- Steps with an asterisk are NOT applicable in the TSC.
- The Duty Call Supervisor (DSC) is available in the Control Room only.
- All Gai-tronics alarms and announcements require Control Room assistance.
- PA announcements are provided as a guideline. Actual announcements may vary from the text provided.
- For assistance with exposure control, refer to:
  - Attachment 4, Field Operator Re-entry Guidelines
  - Attachment 5, Exposure Limits for Emergency Response Personnel
- Not Applicable (N/A) may be used for tasks / steps previously accomplished / satisfied.

1. Determine the following:

   * **A.** Shift Technical Advisor (STA) present Y / N

   * **B.** Duty Call Supervisor (DCS) present Y / N

   * **C.** Shift Communicator present Y / N

   D. Wind direction (from) ____°

/R12

5.4 Site Area Emergency Declaration Checklist (continued) TIME / INITIAL

1. (continued)

NOTE

During any declared emergency, a release is occurring if one of the following is true:

- Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.

OR

- Health Physics detecting airborne radioactivity levels in excess of 25 percent Derived Air Concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

E. Release in progress Y / N

* F. E-Plan Alarm sounded and Emergency Response Facilities (ERFs) activated Y / N

G. Site evacuated Y / N

H. Site accountability Not Requested / In Progress / Complete

2. Mobilize emergency team personnel (i.e., Fire Team, First Aid Team) as required using Gai-tronics and boost function. ___/___

3. The NPS shall declare the emergency to the facility staff and, as necessary, formally announce that he / she is the Emergency Coordinator. ___/___

4. ¶2 If a release of radioactive material is in progress, Then review personnel access with Health Physics personnel and notify Security personnel with any special instructions. ___/___

5. ¶13 Sound the Emergency Plan (E-Plan) Activation Alarm. (N/A if already preformed) ___/___

5.4 Site Area Emergency Declaration Checklist (continued) TIME / INITIAL

6. Notify plant personnel using Gai-tronics and boost function as follows: (N/A if facilities already activated)

"Attention all plant personnel, Unit 1 / 2 has declared a SITE AREA EMERGENCY. All emergency response personnel report at once to your assigned emergency response facility."

Repeat the announcement. ___/___

7. If a SITE AREA EMERGENCY plant announcement has NOT been made, Then notify plant personnel using Gai-tronics and boost function:

"Attention all plant personnel, Unit 1/2 has declared a SITE AREA EMERGENCY."

Repeat the announcement. ___/___

* 8. Notify the Shift Technical Advisor, Duty Call Supervisor and the Shift Communicator, as appropriate to report to the Control Room using Gai-tronics and boost function. (N/A if already performed) /R12

"Shift Technical Advisor report to the Unit 1 / 2 Control Room." ___/___

"Duty Call Supervisor report to the Unit 1 / 2 Control Room." ___/___

"Shift Communicator report to the Unit 1 / 2 Control Room." ___/___ /R12

5.4 Site Area Emergency Declaration Checklist (continued) TIME / INITIAL

NOTE
Site Evacuation Guidance

No release of radioactive material – send personnel home.

Current or prior release of radioactive material send personnel to the off-site assembly area.

- North to Jaycee Park if wind is from 240° through 60° (clock-wise direction)
- South to Jensen Public Beach Parking Area if wind is from 60° through 240° (clock-wise direction)

/R12

9. Sound the Site Evacuation Alarm. (N/A if already performed) ___/___

10. Notify plant personnel using Gai-tronics and boost function as follows: (N/A if already performed)

"Attention all non-emergency response plant personnel, you are directed to commence evacuation of the Owner Controlled Area, report to your vehicles and (Choose one):

Proceed to your homes.

OR

Proceed North / South away form the plant to Jaycee Park / Jensen Public Beach Parking Area for contamination check, accountability and further instructions."

Repeat the announcement. ___/___

* 11. Initiate the call-out process in accordance with EPIP-03, Emergency Response Organization Notification / Staff Augmentation. (this may be accomplished by the DCS) (N/A if already performed) ___/___

12. ¶6 If a release of radioactive material has occurred or is in progress, Then notify Chemistry to promptly implement EPIP-09, Off-site Dose Calculations, and report the results to the Emergency Coordinator (EC). ___/___

5.4 Site Area Emergency Declaration Checklist (continued) TIME / INITIAL

13. If a Chemist is unavailable, Then call-out a Chemist (this may be accomplished by the DCS). ___/___

14. ¶15 Complete required notifications in accordance with Appendix A, Notifications from the Affected Control Room, in EPIP-08, Off-site Notifications and Protective Action Recommendations. The DCS may be utilized as a phonetalker.

    State Warning Point ___/___

    NRC ___/___

15. Ensure notification of the following: (this may be accomplished by the DCS)

    Plant Management ___/___

    Security ___/___

    Nuclear Division Duty Officer (NDDO) ___/___

* 16. Initiate the Operations Department Accountability Aid for both Unit 1 and Unit 2 and provide this list to the TSC when requested. (this may be accomplished by the DCS) (N/A if already performed) ___/___

* 17. ¶9 Ensure Operations field personnel return to their assigned Control Room and obtain emergency Electronic Personal Dosimetry (EPD) from the HP Emergency Kit. (N/A if already performed) ___/___

18. ¶8 Direct all Non-licensed Operators (NLOs), from **both** Units to report to the OSC (when operational) following evacuation of the Owner Controlled Area and completion of immediate Operator actions. ___/___

19. Verify with Security that the evacuation of the Owner Controlled Area has been completed and all personnel have been accounted for. ___/___

5.4 Site Area Emergency Declaration Checklist (continued) TIME / INITIAL

20. Notify off-site agencies when evacuation is complete: (N/A if already performed)

State Warning Point ___/___

NRC ___/___

21. Utilize Attachment 3, Turnover Guidelines when relinquishing duties to the oncoming EC. ___/___

> **NOTE**
> ¶2 New notification forms shall be completed for all updates.

22. ¶15 If State / Local notification has not been completed in the last 60 minutes, Then provide a routine update utilizing a new notification form. The DCS may be utilized as a phonetalker. (Repeat as necessary) ___/___

___/___

___/___

___/___

___/___

23. Turnover off-site interface responsibilities (notifications and Protective Action Recommendations (PARs)) to the Recovery Manager (RM) when the EOF goes operational. ___/___

24. At the direction of the RM, coordinate termination of the emergency and initiation of recovery planning. ___/___

25. SITE AREA EMERGENCY Declaration Checklist complete (emergency upgraded or event terminated). ___/___

END OF SECTION 5.4

5.5 General Emergency Declaration Checklist **TIME / INITIAL**

Date: ____/____/____

CAUTION

- Protective Action Recommendations (PARs) are required for a General Emergency.
- Notification to the State Warning Point (SWP) shall occur within 15 minutes of declaration of the emergency classification.

NOTE

- Steps should be performed in the order presented. When conditions warrant, steps may be performed out of sequence.
- Steps with an asterisk are NOT applicable in the TSC.
- The Duty Call Supervisor (DSC) is available in the Control Room only.
- All Gai-tronics alarms and announcements require Control Room assistance.
- PA announcements are provided as a guideline. Actual announcements may vary from the text provided.
- For assistance with exposure control, refer to:
  - Attachment 4, Field Operator Re-entry Guidelines
  - Attachment 5, Exposure Limits for Emergency Response Personnel
- Not Applicable (N/A) may be used for tasks / steps previously accomplished / satisfied.

1. Determine the following:

   * **A.** Shift Technical Advisor (STA) present Y / N

   * **B.** Duty Call Supervisor (DCS) present Y / N

   * **C.** Shift Communicator present Y / N

   D. Wind direction (from) ____°

/R12

5.5 General Emergency Declaration Checklist (continued) TIME / INITIAL

1. (continued)

> **NOTE**
> During any declared emergency a release is occurring if one of the following is true:
>
> - Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.
>
> OR
>
> - Health Physics detecting airborne radioactivity levels in excess of 25 percent Derived Air Concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

E. Release in progress Y / N

* F. E-Plan Alarm sounded and Emergency Response Facilities (ERFs) activated Y / N

G. Site Evacuation Alarm sounded and site evacuated Y / N

H. Site accountability Not Requested / In Progress / Complete

2. Mobilize emergency team personnel (i.e., Fire Team, First Aid Team) as required using Gai-tronics and boost function. ___/___

3. The NPS shall declare the emergency to the facility staff and, as necessary, formally announce that he / she is the Emergency Coordinator. ___/___

4. ¶2 If a radioactive release is in progress, Then review personnel access with Health Physics personnel and notify Security personnel with any special instructions. ___/___

5. ¶13 Sound the Emergency Plan (E-Plan) Activation Alarm. (N/A if already preformed) ___/___

5.5 General Emergency Declaration Checklist (continued) TIME / INITIAL

6. Notify plant personnel using Gai-tronics and boost function as follows: (N/A if facilities already activated)

"Attention all plant personnel, Unit 1 / 2 has declared a GENERAL EMERGENCY. All emergency response personnel report at once to your assigned emergency response facility."

Repeat the announcement. ___/___

7. If a GENERAL EMERGENCY plant announcement has not been made, Then notify plant personnel using Gai-tronics and boost function:

"Attention all plant personnel, Unit 1 / 2 has declared a GENERAL EMERGENCY."

Repeat the announcement. ___/___

* **8.** Notify the Shift Technical Advisor, Duty Call Supervisor and the Shift Communicator, as appropriate to report to the Control Room using Gai-tronics and boost function. (N/A if already performed) /R12

"Shift Technical Advisor report to the Unit 1 / 2 Control Room." ___/___

"Duty Call Supervisor report to the Unit 1 / 2 Control Room." ___/___

"Shift Communicator report to the Unit 1 / 2 Control Room." ___/___ /R12

5.5 General Emergency Declaration Checklist (continued) TIME / INITIAL

NOTE
Site Evacuation Guidance

No release of radioactive material – send personnel home.

Current or prior release of radioactive material send personnel to the off-site assembly area.

- North to Jaycee Park if wind is from 240° through 60° (clock-wise direction)
- South to Jensen Public Beach Parking Area if wind is from 60° through 240° (clock-wise direction)

/R12

9. Sound the Site Evacuation Alarm. (N/A if already performed) ___/___

10. Notify plant personnel using Gai-tronics and boost function as follows: (N/A if site evacuated)

"Attention all plant personnel, Unit 1 / 2 has declare a GENERAL EMERGENCY, all non-emergency response plant personnel are directed to commence evacuation of the Owner Controlled Area, report to your vehicles and (Choose one):

Proceed to your homes.

OR

Proceed North / South away form the plant to Jaycee Park / Jensen Public Beach Parking Area for contamination check, accountability and further instructions."

Repeat the announcement. ___/___

* **11.** Initiate the call-out process in accordance with EPIP-03, Emergency Response Organization Notification / Staff Augmentation. (this may be accomplished by the DCS) (N/A if already performed) ___/___

5.5 General Emergency Declaration Checklist (continued) TIME / INITIAL

12. ¶6 If a radioactive release has occurred or is in progress, Then notify Chemistry to promptly implement EPIP-09, Off-site Dose Calculations, and report the results to the Emergency Coordinator (EC). ___/___

13. If a Chemist is unavailable, Then call-out a Chemist (this may be accomplished by the DCS). ___/___

14. Complete required notifications in accordance with Appendix A, Notifications from the Affected Control Room, in EPIP-08, Off-site Notifications and Protective Action Recommendations. The DCS may be utilized as a phonetalker.

State Warning Point ___/___

NRC ___/___

15. Ensure notification of the following: (this may be accomplished by the DCS)

Plant Management ___/___

Security ___/___

Nuclear Division Duty Officer (NDDO) ___/___

* **16.** Initiate the Operations Department Accountability Aid for both Unit 1 and Unit 2 and provide this list to the TSC when requested. (this may be accomplished by the DCS) (N/A if already performed) ___/___

* **17.** ¶9 Ensure Operations field personnel return to their assigned Control Room and obtain emergency Electronic Personal Dosimetry (EPD) from the HP Emergency Kit. (N/A if already performed) ___/___

18. ¶8 Direct all Non-licensed Operators (NLOs), from **both** Units to report to the OSC (when operational) following evacuation of the Owner Controlled Area and completion of immediate Operator actions. (N/A if already performed) ___/___

19. Verify with Security that the evacuation of the Owner Controlled Area has been completed and all personnel have been accounted for. (N/A if already performed) ___/___

5.5 General Emergency Declaration Checklist (continued) TIME / INITIAL

20. Notify off-site agencies when evacuation is complete: (N/A if already performed)

State Warning Point ___/___

NRC ___/___

21. Utilize Attachment 3, Turnover Guidelines when relinquishing duties to the oncoming EC. ___/___

NOTE

¶2 New notification forms shall be completed for all updates.

22. ¶15 If State / Local notification has not been completed in the last 60 minutes, Then provide a routine update utilizing a new notification form. The DCS may be utilized as a phonetalker. (Repeat as necessary) ___/___

___/___

___/___

___/___

___/___

23. Turnover off-site interface responsibilities (notifications and Protective Action Recommendations (PARs)) to the Recovery Manager (RM) when the EOF goes operational. ___/___

24. At the direction of the RM, coordinate termination of the emergency and initiation of recovery planning. ___/___

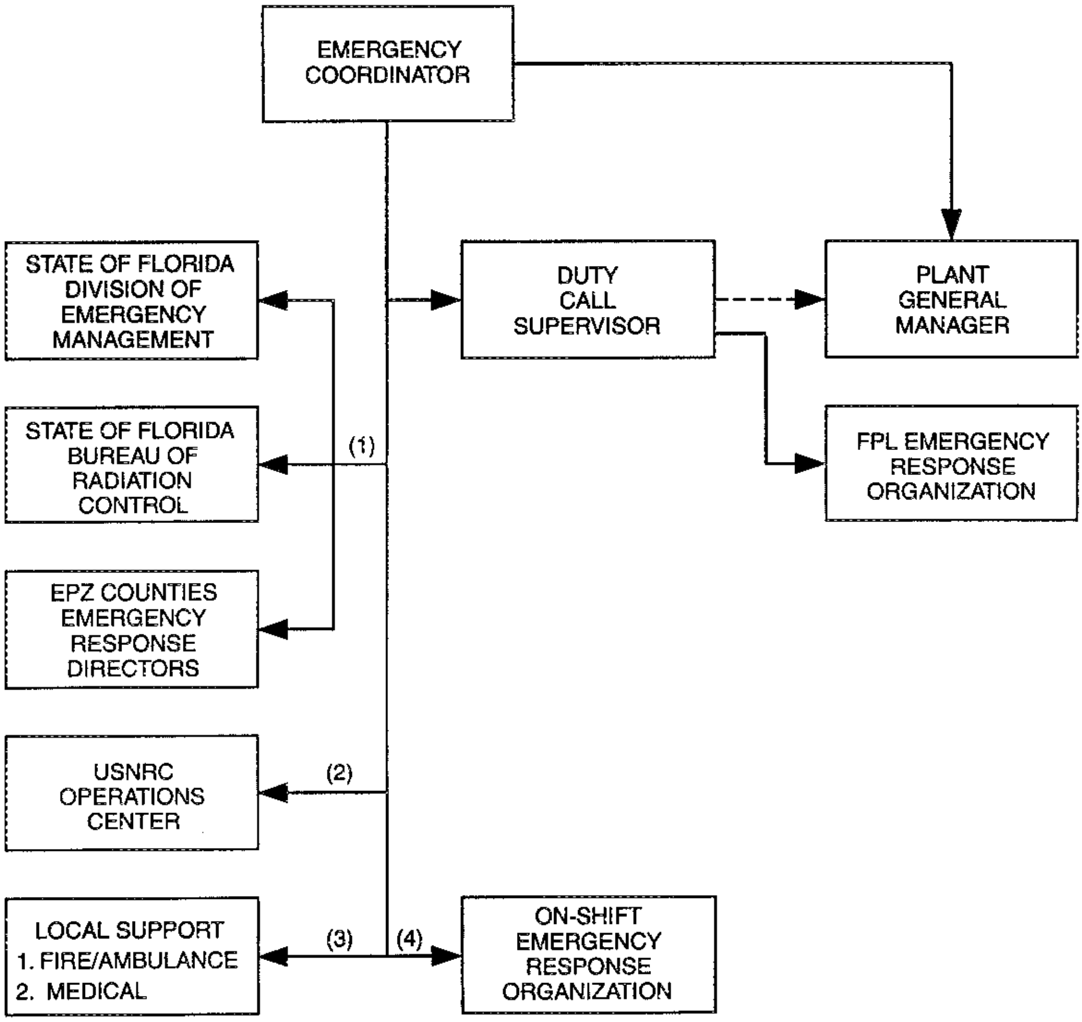
25. GENERAL EMERGENCY Declaration Checklist complete (event terminated). ___/___

END OF SECTION 5.5

ATTACHMENT 1

INITIAL NOTIFICATION FLOW

(Page 1 of 1)



Legend:
—— Primary Notification Pathway
- - - Alternate Notification Pathway

(1) Via State Hot Ring Down Telephone (HRD)
(2) Via Emergency Notification System (ENS)
(3) Medical & Fire Emergencies Only, As Needed
(4) Via Plant Public Address System (PA)

(D/PS/ EPLAN-F1.2-R35)

END OF ATTACHMENT 1

ATTACHMENT 2

CRITERIA FOR EVACUATION

(Page 1 of 1)

A. Criteria for Local Evacuation

The need for Local Evacuation should be determined in accordance with the following criteria:

Evacuate the affected local area in which any of the following conditions occur:

1. Area Radiation Monitor Alarm.
2. Containment Evacuation Alarm.
3. Unevaluated direct radiation dose rate increase in excess of 100 mRem/hour above normal levels.
4. Unexpected airborne radioactivity concentration in excess of 1×10^{-9} micro Ci/cc.
5. Removable radioactive surface contamination in an unposted area in excess of 1000 dpm/100 cm^2 beta-gamma over an area of 100 ft^2.
6. Removable radioactive surface contamination in an unposted area in excess of 50 dpm/100cm^2 alpha over an area of 100 ft^2.
7. The Emergency Coordinator determines that a situation exists for which Local Evacuation is appropriate.

B. Criteria for Owner Controlled Area Evacuation

The Owner Controlled Area shall be evacuated in the following circumstances:

1. Site Area Emergency
2. General Emergency
3. If the Emergency Coordinator determines that the entire Owner Controlled Area should be evacuated.

C. Refer to EPIP-07, Conduct of Evacuations / Assembly, for more information.

END OF ATTACHMENT 2

ATTACHMENT 3
TURNOVER GUIDELINES
(Page 1 of 2)

Upon arrival at the affected Control Room, the prospective Emergency Coordinator should review the following items/issues with the Control Room Emergency Coordinator (not in a particular order):

> **NOTE**
> This information (1-10 below) should be reviewed with the Duty Call Supervisor.

/R12

1. Type of accident or incident
2. Plant status
3. Equipment out-of-service
4. Operator actions underway
5. Radiological conditions
6. Meteorological conditions
7. Procedure status
8. Emergency Plan activities underway, including any on-site or off-site protective actions
9. Conditions and/or trends of concern
10. Personnel injuries or radiation exposures

For an Alert or higher emergency, complete the following:

1. Prior to leaving Control Room verify the status of the following:
   - **A.** Emergency classification
   - **B.** Off-site notifications

ATTACHMENT 3
TURNOVER GUIDELINES
(Page 2 of 2)

2. Bring the following items to the Technical Support Center:

A. Copy of RCO log (entries from start of the event)

B. Completed notification forms (State and NRC)

C. Operations Accountability Aid (only if completed)

END OF ATTACHMENT 3

ATTACHMENT 4
FIELD OPERATOR RE-ENTRY GUIDELINES
(Page 1 of 1)

CAUTION

As specified in ADM-17.09, Invoking 10 CFR 50.54(x), the Emergency Coordinator (EC) may (with the concurrence of a licensed senior operator) waive re-entry requirements to place the plant in a safe shutdown condition or mitigate a release, if this immediate action is needed to protect the health and safety of the public.

1. **Prior to evacuation and with the Operational Support Center (OSC) NOT operational.**

   Re-entry guidelines do not apply.

2. **Prior to evacuation and with the OSC operational.**

   ¶8 Operators in the field should return to the Control Rooms and obtain an Electronic Personal Dosimeter (EPD) from the Health Physics Emergency Kit prior to returning to field.

3. ¶8 **Evacuation ordered and with the OSC NOT operational.**

   Operator actions in the field must be viewed as re-entry activities. Operators shall return to the Control Rooms following the evacuation order. Operators shall obtain an Electronic Personal Dosimeter (EPD) from the Health Physics Emergency Kit, if not done previously. Re-entry into the plant requires:

   A. The EC (initially the NPS) authorize the entry.

   B. Maintenance of appropriate radiological and safety measures.

   C. Tracking the whereabouts of the team.

4. **Evacuation ordered and with the OSC operational**

   A. NLOs, from both Units, are to report to the OSC once it is declared operational.

   B. All field activities are re-entries and shall be coordinated and controlled by the OSC.

END OF ATTACHMENT 4

ATTACHMENT 5

§1 **EXPOSURE LIMITS FOR EMERGENCY RESPONSE PERSONNEL**

(Page 1 of 1)

NOTE

1. Both Total Dose (TEDE) and Thyroid Dose (CDE) should be used for purposes of controlling exposure.
2. Protective clothing, including respirators, should be used where appropriate.

For the following missions, the exposure limit is(1):	Total Dose(2) (TEDE)	THYROID(3) (CDE)
Performance of actions that would not directly mitigate the event, minimize escalation, or minimize effluent releases.	5 REM	50 REM
Performance of actions that mitigate the escalation to the event, rescue persons from a non-life threatening situation, minimize exposures or minimize effluent releases.	10 REM	100 REM
Performance of actions that decrease the severity of the event or terminate the processes causing the event in an attempt to control effluent releases to avoid extensive exposure of large populations. Also, rescue of persons from a life-threatening situation.	25 REM	250 REM
Rescue of person from a life-threatening situation. (Volunteers(4) should be above the age of 45.)	(5)	(5)

---

(1) Exposure limits to the lens of the eye are 3 times the Total Dose (TEDE) values listed.

(2) Total Dose (TEDE) is the total whole body exposure from both external and internal (weighted) sources - Total Effective Dose Equivalent.

(3) Thyroid Dose (CDE) commitment from internal sources - Committed Dose Equivalent. The same dose limits also apply to other organs (CDE), skin (Shallow Dose Equivalent) and extremities (Extremity Dose Equivalent).

(4) Volunteers with full awareness of risks involved including numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

(5) No upper limit for Total Dose (TEDE) and/or Thyroid Dose (CDE) exposure has been established because it is not possible to prejudge the risks that one person should be allowed to take to save the life of another. Also, no specific limit is given for thyroid exposure since in the extreme case, complete thyroid loss might be an acceptable sacrifice for a life saved. This should not be necessary if respirators and/or thyroid protection for rescue personnel are available as the result of adequate planning.

END OF ATTACHMENT 5

FPL	ST. LUCIE PLANT EMERGENCY PLAN IMPLEMENTING PROCEDURE SAFETY RELATED	Procedure No. **EPIP-08** Current Revision No. **6A** Effective Date **01/08/04**

Title:

OFF-SITE NOTIFICATIONS AND PROTECTIVE ACTION RECOMMENDATIONS

Responsible Department: **EMERGENCY PLANNING**

REVISION SUMMARY:

Revision 6A - Incorporated PCR 03-3535 to put Attachment 2 in forms database. (M. Cooper, 12/10/03)

Revision 6 – Incorporated PCR 03-2272 for CR 03-2568 to revise State Notification form. Delete supplemental data sheet. Revise instructions for completing State form. Improve guidance relative to changing PARs (RIS 2003-12). (J.R. Walker, 08/29/03)

AND

Incorporated PCR 03-1637 for MA 03-04-082 to incorporate shift communicator position. (A. Terezakis, 08/06/03)

Revision 5 – Clarified duties, made editorial / administrative changes and removed local government radio. (J. R. Walker, 07/26/02)

Revision 4 – Clarified instructions regarding notification of rapidly degrading events. Clarified stability class instructions. Made administrative/editorial changes. (J.R. Walker, 10/11/01)

Revision	FRG Review Date	Approved By	Approval Date
0	05/30/00	R. G. West Plant General Manager	05/31/00
6A	08/28/03	R.E. Rose Plant General Manager	08/29/03
		N/A Designated Approver	
		D. Calabrese Designated Approver (Minor Correction)	12/10/03

S__OPS	
DATE	
DOCT	PROCEDURE
DOCN	EPIP-08
SYS	
COM	COMPLETED
ITM	6A

TABLE OF CONTENTS

1.0 PURPOSE

1.1 Discussion

1. This procedure provides information and instructions for undertaking notifications of the State Warning Point (SWP) and the Nuclear Regulatory Commission (NRC) and for determination of Protective Action Recommendations (PARS).

2. This procedure is for use in the Control Room, Technical Support Center (TSC) and Emergency Operations Facility (EOF).

3. Upon declaration of an emergency classification the Nuclear Plant Supervisor (NPS) assumes the duties of the Emergency Coordinator (EC). The EC has initial responsibility for off-site notifications and PARs.

4. Once the EOF is operational and proper turnover has been conducted, the Recovery Manager (RM) assumes responsibility for off-site notifications and PARs from the EC.

5. At an Alert or higher level emergency, communications with the NRC transition to an open phone line from the TSC and the EOF (at a Site Area Emergency of higher level emergency).

6. The following table illustrates which facility has a responsibility for Classification, Notification or PARs.

	Control Room (**X** until EC function transfers to the TSC)	**TSC** (**X** when operational)	**EOF** (**X** when operational)
Classifications	**X** transfers →	**X**	
Notifications	**X** transfers →	**X** transfers →	**X**
PARs	**X** transfers →	**X** transfers →	**X**

7. Off-site Notification

   A. Purpose of Off-Site Notifications

   FPL is required to notify off-site agencies in the event of any emergency that could threaten the health and safety of the public. These notifications provide an early warning to agencies responsible for public protection.

1.1 Discussion (continued)

7. (continued)

NOTE
The State Department of Health (Bureau of Radiation Control) may not have their office staffed on a 24-hour basis. In the event that they do not answer the Hot Ring Down (HRD) telephone, the State Warning Point (SWP) assumes responsibility for notifying their duty officer.

B. Who Shall Be Notified

- State Division of Emergency Management
- State Department of Health (Bureau of Radiation Control)
- St. Lucie County Emergency Operations Center
- Martin County Emergency Operations Center
- NRC

1. State and County Notification

a. State and local agencies are notified by using the Hot Ring Down (HRD) telephone. The HRD rings the State Warning Point (SWP). The SWP puts the other agencies on line and reduces the need for individual calls.

b. ¶4 After the State Coordinating Officer (SCO) arrives in the EOF, he / she can transfer "NET Control" to the EOF. When this occurs, the Recovery Manager's PAR Briefing becomes the primary notification method for the State and Counties. The Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1) shall still be completed and provided to the SCO or his / her designee in the EOF. The EOP HRD Communicator should no longer contact the State Warning Point (SWP). /R6

1.1 Discussion (continued)

7. B. (continued)

2. NRC Notification

a. The NRC is notified using the Emergency Notification System (ENS) telephone.

b. NRC notifications occur through an open line of communication in the TSC and, when operational, the EOF.

C. Emergency Follow-up Information Requests from State and local agencies.

1. Incoming calls should come via the SWP over the HRD phone. If the HRD is inoperable, the SWP may use commercial telephone or ESATCOM (emergency satellite phone). If an off-site authority contacts the plant without going through the SWP, request that they contact the SWP. SWP shall verify that the agency calling is a risk county or the Department of Health (DOH) and shall notify other county and state agencies of the updated information, thus reducing the number of calls that may be directed to the plant.

2. Long, detailed explanations of plant systems or reactor theory should be avoided. If prompted for this kind of information by the State Duty Officer, he / she should be referred to the Nuclear Division Duty Officer (NDDO).

3. If the State or one of the Counties provides either the TSC or EOF with new or pertinent information, Then bring that information to the attention of the EC or EC Assistant / Logkeeper in the TSC or the RM or the RM OPS Advisor / Logkeeper in the EOF.

1.1 Discussion (continued)

8. Protective Action Recommendations

A. Protective actions for the general public are ordinarily NOT required prior to declaration of a General Emergency. It is possible however, that due to unusually stable and constant meteorological conditions, protective actions could be recommended at a Site Area Emergency based on projected doses. This is the exception rather than the rule.

Protective actions for the general public are required to be recommended if a General Emergency is declared. Initial Protective Action Recommendations (PARs) are normally based on plant conditions. This would NOT be true if the General Emergency was declared based on off-site dose (either measured or projected) or a Security Emergency (per the Security Plan). The predetermined minimum PARs (based on plant conditions) are as given below.

B. General Emergency - Minimum PARs

1. In any case where a GENERAL EMERGENCY has been declared, the minimum PAR shall be:

Shelter all people within a 2-mile radius and out to 5 miles in the sectors affected. The sectors affected are at least three, the downwind sector plus the two adjacent sectors.

2. If a GENERAL EMERGENCY has been declared due to actual or projected severe core damage, the minimum PAR shall be:

Evacuate all people within a 2-mile radius from the plant and out to 5 miles in the sectors affected. Shelter all people in the remaining sectors from 2 to 5 miles and from 5 to 10 miles from the plant.

3. If a GENERAL EMERGENCY has been declared due to loss of physical control of the plant to intruders, including the Control Room or any other area(s) vital to the operation of the reactor system (as defined in the Security Plan), the minimum PAR shall be:

Evacuate all people within a 2-mile radius from the plant and out to 5 miles in the sectors affected. Shelter all people in the remaining sectors from 2 to 5 miles and from 5 to 10 miles from the plant.

1.1 Discussion (continued)

8. (continued)

C. Once a release of radioactive material occurs, dose assessment should be utilized when evaluating PARs. The final determination of the PAR should consider all available information including off-site dose projections, plant conditions and field monitoring data. The most conservative recommendation shall be made.

D. If it is anticipated that a PAR threshold will be exceeded, DO NOT wait until the threshold is exceeded to make that PAR.

E. ¶12 Conditions (plant information, dose projections and field monitoring results) are to be continually assessed and PARs expanded, as necessary, to ensure that adequate (most conservative) PARs are issued. /R6

F. ¶12 Previously issued PARs, unless found to be less conservative, are to remain in effect until the threat is fully under control and the event is being de-escalated. /R6

G. ¶12 Only State and County officials can implement, change and/or terminate protective actions. /R6

2.0 REFERENCES / RECORDS REQUIRED / COMMITMENT DOCUMENTS

> **NOTE**
>
> One or more of the following symbols may be used in this procedure:
>
> § Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, Operating Experience, License Renewal, etc. and shall NOT be revised without Facility Review Group review and Plant General Manager approval.
>
> ¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.
>
> Ψ Indicates a step that requires a sign off on a data sheet.

2.1 References

1. St. Lucie Plant Updated Final Safety Analysis Report (UFSAR) Unit 1 and Unit 2
2. St. Lucie Plant Technical Specifications Unit 1 and Unit 2
3. §1 St. Lucie Plant Radiological Emergency Plan (E-Plan)
4. E-Plan Implementing Procedures (EPIP 00 – 13)
5. St. Lucie Plant Emergency Response Directory (ERD)
6. QI-17-PSL-1, Quality Assurance Records

2.2 Records Required

1. All PAR worksheets and notifications forms (all attachments) shall be maintained in plant files in accordance with QI-17-PSL-1.

2.3 Commitment Documents

1. ¶1 PMAI PM96-04-165, "ITR 96-006" (Unusual Event Declared Due to Dropped Rod)
2. ¶2 PMAI PM96-09-185, Condition Report CR-96-1750 (Off-site Notification Using Commercial Phone)
3. ¶3 NRC Inspection Report 91-01, Closure of IFIs 89-31-03 and 89-31-01
4. ¶4 Condition Report CR-00-0428 (Evaluated Exercise Critique)

2.3 Commitment Documents (continued)

5. ¶6 PMAI PM96-05-233 (Off-site Notification Process)

6. ¶7 PMAI PM99-09-016 (PARs Based on FMT Data, Completion of NRC Notification Form)

7. ¶8 NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, Section 4.2.4, ENS Event Notification Worksheet (NRC Form 361).

8. ¶9 Condition Reports CR-01-0726 and CR-01-0742 (NOUEs Associated with SDC During SL1-17 Outage)

9. ¶10 Condition Report CR-01-0389 (Alternate Met Data Source)

10. ¶11 Condition Report CR-02-0333 (Role of Duty Call Supervisor)

11. ¶12 Condition Report CR-03-2568 (Response to RIS 2003-12 Regarding PARs) /R6

3.0 RESPONSIBILITIES

3.1 Emergency Coordinator – Responsible for classifications, notifications and PARs.

3.2 Recovery Manager – Responsible for notifications and PARs.

3.3 ¶11 Duty Call Supervisor – Assists the EC as a phonetalker. /R6

3.4 TSC EC Assistant / Logkeeper or TSC OPS Coordinator – Prepares notification forms (Attachment 1, Florida Nuclear Plant Emergency Notification Form, and if necessary, Attachment 3, NRC Reactor Plant Event Notification Worksheets) for EC approval when the TSC is operational. /R6

3.5 EOF RM OPS Advisor / Logkeeper – Prepares notification forms (Attachment 1 and if necessary, Attachment 3) for RM approval when the EOF is operational. /R6

3.6 TSC HRD Communicator – Assists the TSC EC Assistant / Logkeeper or TSC OPS Coordinator with notification form preparation and makes calls to complete notifications to the SWP.

3.7 EOF HRD Communicator – Assists the EOF RM OPS Advisor with form preparation and makes calls to complete notifications to the SWP and the SCO following transfer of Net Control by the Division of Emergency Management (DEM).

3.8 TSC Chemistry Supervisor (in his absence, TSC Dose Assessor) – Assists the EC with radiological dose assessment data and PARS.

3.9 EOF HP Manager (in his absence, EOF Dose Assessor) – Assists the RM with radiological dose assessment data and PARS.

3.10 TSC Supervisor – Oversees communications performed by the TSC Communicators (HRD, ENS, Health Physics Network (HPN), Sound-Powered Phonetalker, EOF and Field Monitoring Team).

3.11 EOF Nuclear Licensing Manager – Oversees EOF communications performed by the EOF Communicators (HRD, ENS, HPN and TSC).

3.12 Information Services – Maintains user copies, in the Unit 1 and Unit 2 Control Rooms, of the following checklist and supporting attachments for making notifications and developing Protective Action Recommendations:

- Appendix A, Notifications from the Affected Control Room
- Attachment 1 – Florida Nuclear Plant Emergency Notification Form
- Attachment 1A – Directions for Completing the Florida Nuclear Plant Emergency Notification Form
- Attachment 2 – Determination of Protective Action Recommendations (PARs) /R6
- Attachment 3 – NRC Reactor Plant Event Notification Worksheet /R6
- Attachment 3A – Directions for Completing the NRC Reactor Plant Event Notification Worksheet /R6

3.13 Shift Communicator – Assists the Nuclear Plant Supervisor/Emergency Coordinator in making emergency off-site notifications and performing other activities, as directed. /R6

4.0 DEFINITIONS

4.1 **Conservative** – Means more extensive or comprehensive action under a given set of circumstances to provide a greater measure of safety. For example, evacuation is more conservative than sheltering.

4.2 **Emergency** – Any off-normal event or condition which is classified into one of the four emergency classes (Unusual Event, Alert, Site Area Emergency, or General Emergency) by the NPS in accordance with EPIP-01, Classification of Emergencies.

4.3 **Emergency Coordinator (EC)** – The title initially assumed by the NPS, until relieved by plant management through proper turnover, in the event of plant conditions that trigger implementation of the Emergency Plan. The EC is responsible for notifying off-site authorities, emergency responders both inside and outside the company and has full authority and responsibility for on-site emergency response actions. The EC is also responsible for Protective Action Recommendations during the initial stages of an emergency.

4.4 **Florida Nuclear Plant Emergency Notification Form** – A predetermined format used by nuclear power plants throughout the State for notification and local authorities.

4.5 **Operational** (status for an emergency facility) – The mandatory minimum staff is present and the facility has taken responsibility for its procedurally assigned functions.

4.6 **Protective Action Recommendations** (PARs) – Recommendations, for action instructions to protect the public, made by the Emergency Coordinator or Recovery Manager to State and County officials. FPL may recommend No Action, Sheltering or Evacuation.

4.7 **Recovery Manager** (RM) – A designated company officer or senior manager, who will have responsibility for the direction and control of the EOF. He / she has the authority to establish policy and to expend funds necessary to cope with emergency situations that trigger the implementation of the Emergency Plan.

4.8 **Release** (during any declared emergency)

1. Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values.

OR

2. Health Physics detecting airborne radioactivity levels in excess of 25% derived air concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

4.9 **Shift Communicator** – A specific shiftly designated individual trained and qualified to assist the Nuclear Plant Supervisor/Emergency Coordinator in the control room in making emergency off-site notifications, and performing other activities as directed. /R6

4.10 **State Notification Form (SNF)** – Less formal, more concise expression used in lieu of Florida Nuclear plant Emergency Notification Form.

5.0 INSTRUCTIONS

5.1 State and County Notification

1. Time Limits

   A. Notification shall be initiated within 15 minutes of any of the following:

      1. Recognition of entry into the Emergency Plan.
      2. Escalation in Emergency Class.
      3. De-escalation of the Emergency Class.
      4. Protective Action Recommendation.
      5. Change in Protective Action Recommendation.

   B. Notification shall be initiated within 60 minutes of any of the following:

      1. At an Alert or higher Emergency Class, the time of the last update (unless a different frequency has been agreed to by the off-site agencies as during a hurricane).
      2. A radiological release has been initiated.
      3. A radiological release has been terminated.
      4. A significant change in plant conditions has occurred (e.g., loss or restoration of off-site power or major plant equipment).
      5. Termination of the emergency.

5.1 State and County Notification (continued)

2. Forms Required for Notifications

CAUTION

Notifications require the use of a form similar to Attachment 1, Florida Nuclear Plant Emergency Notification Form. /R6

A. Notifications with 15 minute time limits shall be made using a form similar to Attachment 1, Florida Nuclear Plant Emergency Notification Form. /R6

B. Notifications with 60 minute time limits shall be made using a form similar to Attachment 1, Florida Nuclear Plant Emergency Notification Form. /R6

3. Special instructions due to extraordinary circumstances.

A. If Emergency Class escalation is necessary due to rapidly degrading conditions, Then provide the State and County authorities with the initial notification information by transmitting lines 1-6, at a minimum, of the SNF and terminate the phone call by stating that a new notification form will be provided within 15 minutes.

CAUTION

There can not be two concurrent declared emergency classes under the St. Lucie Plant Radiological Emergency Plan.

B. If one Unit is in a classified event and the same or the other Unit enters into an event where the same or lesser Emergency Class would apply, Then a new classification should NOT be declared. The event should be documented on a SNF as "Additional Information or Update" and issued as soon as practicable.

C. If one Unit is in a classified event and the other Unit enters into a more severe event in which a higher Emergency Class would apply, Then the new classification shall be declared and promptly, within the regulatory time limits, issued to the State, Counties and the NRC.

5.1 State and County Notification (continued)

4. ¶4 Transfer of NET Control

A. The State Coordinating Officer (SCO) can transfer the control of Hot Ring Down (HRD) NET from the State Warning Point (SWP) to the EOF. When this occurs;

1. The RM shall do face to face communication to satisfy off-site notification requirements for the State and Counties. Calls to the SWP are no longer necessary.

2. The Florida Nuclear Plant Emergency Notification Form (Attachment 1) shall continue to be filled out. /R6

3. Completed notification forms are to be provided to the SCO or his / her designee in the EOF.

END OF SECTION 5.1

5.2 Nuclear Regulatory Commission (NRC) Notification

1. Time Limits

> **NOTE**
> Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

A. The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes (10 CFR 50.72 (a)(3)).

2. Special Instructions

A. Initial notification to the NRC using the Emergency Notification System (ENS) (usually done from the Control Room) should use Attachment 3, NRC Reactor Plant Event Notification Worksheet. /R6

B. At an Alert or higher emergency class, the NRC will want to establish an open line of communication with the Control Room, utilizing an ENS conference bridge tying in the licensee with NRC Headquarters and region personnel. Once the Technical Support Center (TSC) is operational, the Control Room should transfer responsibility for NRC communications to the TSC.

C. The Emergency Operations Facility (EOF) should join the TSC on the ENS conference bridge and take the lead for NRC communications.

D. The TSC and EOF should also utilize the Health Physics Network (HPN) line in a manner similar to the ENS (i.e., establish a conference bridge with the NRC).

E. Both the ENS and HPN Communicators in both facilities should keep logs of information transmitted and received from the NRC in accordance with procedures.

END OF SECTION 5.2

5.3 ¶1 Erroneous Information

1. If erroneous information is transmitted to off-site agencies and the error is discovered prior to event termination, a correction should be provided in an update. The need for and urgency of providing the update is dependent upon the importance of the error.

2. If erroneous information is transmitted to off-site agencies and the error is discovered after event termination, the Licensing Department should be consulted to determine the need and method for contacting the off-site agencies with corrected information.

END OF SECTION 5.3

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 1 of 4)

INITIAL

CAUTION

- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of an Emergency Class.
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

NOTE

¶9 1. Completion of this checklist requires the following Attachments (all from EPIP-08):

Attachment 1 – Florida Nuclear Plant Emergency Notification Form

Attachment 1A – Directions for Completing the Florida Nuclear Plant Emergency Notification Form

Attachment 2 – Determination of Protective Action Recommendations (PARs)

Attachment 3 – NRC Reactor Plant Event Notification Worksheet

Attachment 3A – Directions for Completing the NRC Reactor Plant Event Notification Worksheet

2. Checklist Part 1 is for State Warning Point notification.

3. Checklist Part 2 is for NRC notification.

/R6

1. State Warning Point Notification

A. Prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1) in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. ______

B. Emergency Coordinator (EC) approval. ______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 2 of 4)

1. (continued) INITIAL

> **NOTE**
> 1. Primary notification method to the State Warning Point (SWP) is to use the Hot Ring Down (HRD) phone.
> 2. If the HRD is out-of-service, alternate notification methods are provided in Section E, below.

C. Using the State HOT RING DOWN (HRD) Phone, dial 100. ______

D. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce "This is St. Lucie Nuclear Plant [as applicable (Unit 1, 2)] with an emergency message. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy." Allow the Duty Officer to contact St. Lucie County, Martin County and the Bureau of Radiation Control prior to transmitting the information from the notification form. When the parties are on line, provide the information slowly (e.g., in three word intervals) and deliberately, providing time for the information to be written down. ______ /R6

E. Alternate Notification Methods (in order of priority)

> **NOTE**
> Use of the commercial telephone as an alternate notification method requires callback verification from the State Warning Point. Use of ESATCOM as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

a. Call the State Warning Point using the phone number in the St. Lucie Plant Emergency Response Directory (ERD). Announce "This is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] with an emergency declaration. My callback number is ________." ______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 3 of 4)

1. E. 1. (continued) INITIAL

b. Hang up the phone and standby for the callback. When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______

c. ¶2 Request callback from the State Warning Point to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

2. Alternate 2 - ESATCOM

NOTE
Use ESATCOM only if Alternate 1 – commercial phone is not available.

a. Hold down the "push-to-talk" button on the handset and wait 3-5 seconds to hear a beep before you start talking. This must be done each time you talk. ______

b. Announce "State Warning Point, this is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] with an emergency declaration." Then release the "push-to-talk" button in order to listen. ______

c. When the State Warning Point acknowledges, announce "State Warning Point, this is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] declaring a / an (<u>classification</u>), repeat (<u>classification</u>). I am standing by to transmit Florida Nuclear Plant Emergency Notification Form information when you are ready to copy. When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______

d. Announce "St. Lucie clear" at the end of the conversation. ______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 4 of 4)

INITIAL

> **CAUTION**
> Notification of the NRC is expected immediately after notification of State and local agencies. The one hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

2. §1 NRC Notification

A. Prepare the NRC Reactor Plant Event Notification Worksheet (form similar to Attachment 3) in accordance with Attachment 3A, Directions for Completing the NRC Reactor Plant Event Notification Worksheet. ______ /R6

B. EC approval. ______

> **NOTE**
> 1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
> 2. If the ENS is out-of-service an alternate notification method is provided in Section D, below.

C. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD). ______

D. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX A

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 1 of 5)

INITIAL

CAUTION

- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of an Emergency Class.
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

NOTE

- Checklist Part 1 is for HRD Communications.
- Checklist Part 2 is for ENS Communications.

1. State Warning Point Notification

   A. Prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1) in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form.

   B. Verify the Emergency Coordinator (EC) approval. ______

NOTE

1. Primary notification method to the State Warning Point (SWP) is to use the Hot Ring Down (HRD) phone.
2. If the HRD is out-of-service, alternate notification methods are provided in Section E, below.

   C. Using the State HOT RING DOWN (HRD) Phone, dial 100. ______

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 2 of 5)

1. (continued) INITIAL

D. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce "This is St. Lucie Nuclear Plant Technical Support Center with an emergency message. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy." Allow the Duty Officer to contact St. Lucie County, Martin County and the Bureau of Radiation Control prior to transmitting the information from the notification forms. When the parties are on line, provide the information slowly (e.g., in three word intervals) and deliberately, providing time for the information to be written down. ______ /R6

E. Alternate Notification Methods (in order of priority)

NOTE
Use of the commercial telephone as an alternate notification method requires callback verification from the State Warning Point. Use of ESATCOM as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

   a. Call the State Warning Point using the phone number in the St. Lucie Plant Emergency Response Directory (ERD). Announce "This is St. Lucie Nuclear Plant Technical Support Center with an emergency declaration. My callback number is _________." ______

   b. Hang up the phone and standby for the callback. When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______

   c. ¶2 Request callback from the State Warning Point to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

APPENDIX B

NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)

(Page 3 of 5)

1. E. (continued) INITIAL

2. Alternate 2 - ESATCOM

NOTE

Use ESATCOM only if Alternate 1 – commercial phone is not available.

a. Hold down the "push-to-talk" button on the handset and wait 3-5 seconds to hear a beep before you start talking. This must be done each time you talk. ______

b. Announce "State Warning Point, this is St. Lucie Nuclear Plant Technical Support Center with an emergency declaration." Then release the "push-to-talk" button in order to listen. ______

c. When the State Warning Point acknowledges, announce "State Warning Point, this is St. Lucie Nuclear Plant Technical Support Center declaring a / an (classification), repeat (classification). I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy. When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______ /R6

d. Announce "St. Lucie clear" at the end of the conversation. ______

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 4 of 5)

INITIAL

CAUTION
Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

NOTE
1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
2. If the ENS is out-of-service, an alternate notification method is provided in Section B, below.

2. §1 NRC Notification

A. Choose and complete the appropriate steps, below:

1. If the NRC Reactor Plant Event Notification Worksheet has NOT previously been transmitted from the Control Room, Then request that the EC Assistant / Logkeeper prepare the form. ______
2. Verify EC approval. ______
3. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD), then GO TO the next step to establish an open line of communication with the NRC. ______

OR

4. If the NRC Reactor Plant Event Notification Worksheet has previously been transmitted by the Control Room, Then initiate an open line of communication with the NRC by dialing one of the numbers shown on the phone or in the ERD and request to be placed on the Conference Bridge with the NRC. ______
5. As requested, provide information to the NRC. ______

APPENDIX B

NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)

(Page 5 of 5)

2. (continued) INITIAL

B. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX B

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 1 of 6)

INITIAL

CAUTION

- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of Emergency Class or change in Protective Action Recommendation (PAR).
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

NOTE

- Checklist Part 1 is for HRD Communications.
- Checklist Part 2 is for ENS Communications.

1. State Warning Point Notification

   A. Prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1) in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form.

   B. Verify the Recovery Manager (RM) approval. ______

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 2 of 6)

1. (continued)

INITIAL

NOTE

1. Primary notification method to the State Warning Point (SWP) is to use the Hot Ring Down (HRD) phone.
2. If the HRD is out-of-service, alternate notification methods are provided in Section D, below.
3. State and County representatives means Florida Division of Emergency Management (DEM), Florida Department of Health (DOH), St. Lucie County Department of Public Safety (DPS) and Martin County Department of Emergency Services (DES).
4. Notification forms means the Florida Nuclear Plant Emergency Notification Form. /R6

C. Choose and complete the appropriate step below:

1. If State and County representatives are NOT co-located with the FPL Emergency Response Organization (ERO) in the EOF, Then call the SWP and transmit the notification forms. To contact the SWP, dial 100. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce "this is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency message. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy." Allow the Duty Officer to contact the Bureau of Radiation Control, St. Lucie County DPS and Martin County DES prior to transmitting the information from the notification forms. When the parties are on line, transmit the information slowly, (e.g., in three word intervals) and deliberately, providing time for the information to be written down. _______ /R6

OR

APPENDIX C

NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)

(Page 3 of 6)

1. C. (continued) INITIAL

2. If State and County representatives are co-located with the FPL ERO in the EOF and the State Coordinating Officer (SCO) has NOT assumed Net Control, Then call the SWP and transmit the notification forms – "Time of Contact" corresponds to the start time of the Recovery Manager's Protective Action Recommendation (PAR) Briefing. To contact the SWP, dial 100. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce "this is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency message. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy." Allow the Duty Officer to contact the Bureau of Radiation Control, St. Lucie County DPS and Martin County DES prior to transmitting the information from the notification forms. When the parties are on line, transmit the information slowly, (e.g., in three word intervals) and deliberately, providing time for the information to be written down. ______ /R6

OR

3. If State and County representatives are co-located with the FPL ERO in the EOF and the SCO has transferred Net Control to the EOF, Then the SWP is not called (completed notification forms are given to the SCO – may be accomplished by the RM or RM OPS Advisor / Logkeeper). ______

D. Alternate Notification Methods (in order of priority)

NOTE

Use of the commercial telephone as an alternate notification method requires callback verification from the State Warning Point. Use of ESATCOM as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

APPENDIX C

NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)

(Page 4 of 6)

1. D. 1. (continued) INITIAL

a. Call the State Warning Point using the phone number in the St. Lucie Plant Emergency Response Directory (ERD). Announce "This is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency declaration. My callback number is ________." ______

b. Hang up the phone and standby for the callback. When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______

c. ¶2 Request callback from the State Warning Point to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

2. Alternate 2 - ESATCOM

NOTE
Use ESATCOM only if Alternate 1 – commercial phone is not available.

a. Hold down the "push-to-talk" button on the handset and wait 3-5 seconds to hear a beep before you start talking. This must be done each time you talk. ______

b. Announce "State Warning Point, this is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency declaration." Then release the "push-to-talk" button in order to listen. ______

c. When the State Warning Point acknowledges, announce "State Warning Point, this is St. Lucie Nuclear Plant Emergency Operations Facility declaring a / an (classification), repeat (classification). I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy." When the State Warning Point gives the go-ahead, provide the information from the Florida Nuclear Plant Emergency Notification Form. ______ /R6

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 5 of 6)

1. D. 2. (continued) INITIAL

d. Announce "St. Lucie clear" at the end of the conversation. ______

CAUTION
Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

2. §1 NRC Notification

NOTE
1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
2. If the ENS is out-of-service, an alternate notification method is provided in Section B, below.

A. Choose and complete the appropriate steps, below:

1. If the NRC Reactor Plant Event Notification Worksheet has NOT previously been transmitted from either the Control Room or Technical Support Center (TSC), Then request that the RM OPS Advisor prepare the form. ______

2. Verify RM approval. ______

3. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD), then GO TO the next step to establish an open line of communication with the NRC. ______

OR

4. If the NRC Reactor Plant Event Notification Worksheet has previously been transmitted by either the Control Room or the TSC, Then initiate an open line of communication with the NRC by dialing one of the numbers shown on the phone or in the ERD and request to be placed on the Conference Bridge with the NRC and the St. Lucie TSC. ______

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 6 of 6)

2. A. (continued) — INITIAL

5. Take the lead in providing information to the NRC. ______

B. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX C

ATTACHMENT 1
FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 1 of 1)

On-line Verification: ☐ SWP/DEM ☐ St. Lucie County ☐ Martin County

1. A. ☐ THIS IS A DRILL B. ☐ THIS IS AN EMERGENCY
2. A. Date: __/__/__ B. Contact Time: ______ C. Reported by: (Name) ______
   D. Message Number: ______ E. Reported from: ☐ Control Room ☐ TSC ☐ EOF
3. Site: A. ☐ Crystal River Unit 3 B. ☐ St. Lucie Unit 1 C. ☐ St. Lucie Unit 2
   D. ☐ Turkey Point Unit 3 E. ☐ Turkey Point Unit 4
4. Emergency Classification: A. ☐ Notification of Unusual Event C. ☐ Site Area Emergency
   B. ☐ Alert D. ☐ General Emergency
5. A. ☐ Emergency Declaration: B. ☐ Emergency Termination: Date: __/__/__ Time: ______
6. Reason for Emergency Declaration:* A. ☐ EAL Number OR B. ☐ Description:
7. Additional Information or Update: A. ☐ None OR B. ☐
8. Weather Data: A. Wind direction from ______ degrees B. Downwind Sectors Affected: ______, ______, ______
9. Release Status: A. ☐ None (Go to Item 11) B. ☐ Is occurring C. ☐ Has occurred, but stopped
10. Release Significance Category (at the Site Boundary):
    A. ☐ Information not available at this time.
    B. ☐ Release within normal operating limits (≤ 3.5 E-1 Ci/sec noble gas, ≤ 4.6 E-5 Ci/sec iodine)
    C. ☐ Non-Significant Fraction of PAG Range (release is > normal limits and < 500 mrem TEDE and 1000 mrem CDE)
    D. ☐ PAG Range (≥ 500 mrem TEDE or ≥ 1000 mrem CDE)
11. Utility Recommended Protective Actions:
    A. ☐ No recommended actions at this time.
    B. ☐ The utility recommends the following protective actions:

☐ Miles	No Action Sectors	Evacuate Sectors	Shelter Sectors
0 – 2	______	______	______
2 – 5	______	______	______
5 - 10	______	______	______

OR

Shelter Zones / Areas: ______

Evacuate Zones / Areas: ______

C. Consider Issuance of KI: ☐ Yes ☐ No

If form is completed in the Control Room, go to Item 15. If completed in the TSC or EOF, continue with item 12.

12. Plant Conditions: A. Reactor Shutdown? ☐ Yes ☐ No B. Core Adequately Cooled? ☐ Yes ☐ No
    C. Containment Intact? ☐ Yes ☐ No D. Core Condition: ☐ Stable ☐ Degrading
13. Weather Data: A. Wind Speed ______ mph B. Stability Class ______
14. Additional Release Information: ☐ N/A OR
    A. ☐ Noble Gases ______ Curies per second B. ☐ Iodines ______ Curies per second
    C. Airborne: Date Started __/__/__ Time Started ______ Date Stopped __/__/__ Time Stopped ______
    D. Liquid: Date Started __/__/__ Time Started ______ Date Stopped __/__/__ Time Stopped ______

Distance	Projected Thyroid Dose (CDE) for 1 Hour	Projected Total Dose (TEDE) for 1 Hour
1 Mile (Site Boundary)	E. ______ mrem	F. ______ mrem
2 Miles	G. ______ mrem	H. ______ mrem
5 Miles	I. ______ mrem	J. ______ mrem
10 Miles	K. ______ mrem	L. ______ mrem

EC or RM Approval Signature: ______ Date: __/__/__ Time: ______

15. Message Received By: Name: ______ Date: __/__/__ Time: ______

* If emergency class escalation is known to be necessary and a new notification form will be transmitted within 15 minutes, then you may go to line 15.

PSL-F070 Effective Date: 08/29/03

END OF ATTACHMENT 1

R6

¶6 ATTACHMENT 1A

DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 1 of 7)

ITEM ENTRY

On-line Verification - Check the appropriate boxes as the State Warning Point (Florida Division of Emergency Management) requests that St. Lucie County Department of Public Safety and the Martin County Division of Emergency Management get on the line, prior to initiating the notification. All three agencies must be notified through the SWP or alternate means.

1. Check appropriate box for drill or actual emergency as the case may be. During exercises, drills, or tests, each message shall be checked **THIS IS A DRILL.**

2A. Enter today's date.

2B. Enter the time (using the official time, normally synchronized with ERDADS) when contact is made with the State Warning Point or the start time of the RM PAR Briefing. For initial notification of classification, this shall be within 15 minutes of the "Emergency Declaration" time in item 5.

2C. Enter the name of the person making the notification call.

2D. Enter the message number beginning with #1 and following sequentially in all facilities (e.g., if the Control Room transmitted two messages the TSC would start with #3).

2E. Check the box for the facility from which the notification is being made.

3. Site
Check the box for the appropriate plant site for the emergency declaration (both St Lucie boxes might need to be checked for dual unit events such as approach of a hurricane).

4. Emergency Classification
Check the box corresponding to current accident classification declared.

5. Emergency Declaration or Emergency Termination
Enter the **date** and **time** when the current emergency classification was declared (A) or (B) when the emergency was terminated.

/R6

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 2 of 7)

6. Reason for Emergency Declaration
Enter the Emergency Action Level (EAL) number (A) (This option is currently not being used at Plant St. Lucie) or (B) enter wording like that found in the EAL information in EPIP-01, Classification Of Emergencies. Wording should be brief yet descriptive enough for the off-site agencies to gain an understanding of the event. It should be clear from the incident description which EAL has necessitated the emergency declaration. Wording should be as non-technical as possible with no acronyms or abbreviations. This information should remain the same throughout update messages, unless there is a classification change.

"*" asterisk and instruction provided at the bottom of form - If Emergency Class escalation is necessary due to rapidly degrading conditions, Then provide the State and County authorities with the initial notification information by transmitting lines 1-6, at a minimum, on the State Notification Form (SNF) and terminate the call by stating that a new notification form will be provided within 15 minutes.

7. Additional Information or Update
Check "None" (A) or (B) Description and enter additional information, if necessary, or reason for update here. For example:

- Protective Action Recommendations (PARs) change
- An occurrence that would otherwise result in a lower emergency classification, on other unit
- Weather changes affecting public safety
- Radiation level changes
- Loss of off-site power, etc.

/R6

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 3 of 7)

NOTE

If the Class A Model (dose projection model) is being used, a 'State Notification Form Summary Sheet' is available which provides information for items 8-11, 13 and 14. The information is in a format similar to that found on the Florida Nuclear Plant Emergency Notification Form. /R6

8. Weather Data

NOTE

10 meter data should be used.

A. ¶10 Wind direction can be obtained from ERDADS by depressing the "EPIP" key, on the top row of the keyboard. The Met Tower Indicator Panel in the Unit 1 Control Room is an alternate source. If these two sources are not available, refer to Attachment 1, Meteorological Data, in EPIP-09, Off-Site Dose Calculations.

B. If the wind direction is greater than 360° the wind direction is determined by subtracting 360° from the indicated number. Wind direction should be rounded to the nearest whole number.

C. Wind direction is always given as "wind from" (an easterly wind, or wind direction 90°, means that the wind is blowing from east to west).

D. When determining the sectors affected, the adjacent sectors on both sides of the actual downwind sector are included. Three sectors will typically be listed.

E. If the wind is located on the edge of a sector (i.e., 11°, 33°, etc.) an additional (fourth) sector should be added. /R6

F. Enter the wind direction (wind from) in degrees in item "A."

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 4 of 7)

8. (continued)

G. Enter the downwind sectors in item "B."

Wind From	Sectors Affected	Wind From	Sectors Affected	Wind From	Sectors Affected
348-11	HJK	123-146	PQR	236-258	CDE
11-33	JKL	146-168	QRA	258-281	DEF
33-56	KLM	168-191	RAB	281-303	EFG
56-78	LMN	191-213	ABC	303-326	FGH
78-101	MNP	213-236	BCD	326-348	GHJ
101-123	NPQ	There is no "O" sector		There is no "I" sector	

9. Release Status

A. If there are no indications of a release of radioactive material, check box "A" and go to item 11. /R6

A release of radioactive material (during any declared emergency) is defined as:

- Any effluent monitor increase of (approximately) 10 times or one decade above pre-transient values

OR

- Health Physics detecting airborne radioactivity levels in excess of 25% derived air concentration (DAC) outside of plant buildings due to failure of equipment associated with the declared emergency.

B. If a release of radioactive material is occurring, even though it may be less than normal operating limits, check box "B."

C. If a release has occurred but stopped, check box "C." /R6

Dose Assessment personnel in the TSC or EOF will have this information. The TSC Chemistry Supervisor, TSC HP Supervisor or EOF HP Manager should be contacted for the data.

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 5 of 7)

10. Release Significance Category
Do Not Check Any Box in Item 10 if you Checked Box 9 "A" No Release

A. If a release is occurring or has occurred and dose information is not available at the time of notification, check box "A" and follow up as soon as information becomes available.

B. Check box "B" if both noble gas and iodine release rates are less than or equal to the following:

Noble Gas release $\leq$ 3.5 E+5 μCi/sec (3.5 E-1 Ci/sec)
Iodine release $\leq$ 4.6 E+1 μCi/sec (4.6 E-5 Ci/sec)

C. Check box "C" if either noble gas or iodine release rates exceed the values in "B" (above) but forecasted 1 mile doses are less than either 500 mrem TEDE or 1000 mrem Thyroid CDE. These doses are less than the state's Protective Action Guide (PAG) levels.

D. Check box "D" if forecasted 1 mile doses are greater than or equal to either 500 mrem TEDE or 1000 mrem Thyroid CDE. These PAG levels require state and county action.

11. Utility Recommended Protective Actions

A. If there are no Protective Action Recommendations (PARs), check Box "A."

B. If PARs are necessary, check Box "B". Two formats are provided to record PARs. Use the "sector" format and determine appropriate PARs using the guidance in Attachment 2 to this procedure. Copy the PARs into item 11 "C." Indicate PARs using only the words NONE, ALL, ALL REMAINING or by listing the letters of the sectors affected. Protective Action Recommendations shall be approved by the Emergency Coordinator (EC) or the Recovery Manager (RM). The "zone" format is for Crystal River Unit 3 use only.

C. Check the "Yes" box (to consider issuance of potassium iodide (KI) only if:

(1) A General Emergency has been declared

AND

(2) A release of radioactive material is occurring.

/R6

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 6 of 7)

12. Plant Conditions

Answer the three questions "Yes" or "No" by checking the appropriate box.

A. Is the reactor shut down?

B. Is the core adequately cooled?

C. Is the containment intact?

Answer the question regarding the condition of the core as either stable or degrading.

13. Weather Data

NOTE
10 meter data should be used.

A. ¶10 Temperature, wind speed and wind direction can be obtained from ERDADS by depressing the "EPIP" key, on the top row of the keyboard. The Met Tower Indicator Panel in the Unit 1 Control Room is an alternate source. If these two sources are not available, refer to Attachment 7, Meteorological Data, in EPIP-09, Off-site Dose Calculations.

B. Enter wind speed in Miles Per Hour (MPH) in item "A".

C. Stability Class - Enter the stability class as determined by using the figure below. The figure shows the relationship between the Delta T displayed by ERDADS and the stability class.

If Delta-T is	Then Stability Class is
Less than or equal to -1.7	A
-1.6 to -1.5	B
-1.4	C
-1.3 to -0.5	D
-0.4 to +1.4	E
+1.5 to +3.6	F
Greater than +3.6	G

/R6

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 7 of 7)

14. Additional Release Information

This section requires that a release be in progress and completed results of dose assessment be available. Check the "N/A" box if no release is occurring and/or if dose information is not available. Otherwise, provide all information that applies.

A. Enter the noble gas release rate in curies per second.

B. Enter the iodine release rate in curies per second.

C. For an airborne release, enter the date and time started and when terminated, the date and stopped.

D. For a liquid release, enter the date and time started and when terminated, the date and time stopped.

Projected Dose Information - Enter the projected Thyroid Dose (CDE) in mrem for 1 hour (EPIP-09, Manual Dose Calculation Worksheet, Line 5) and the projected Total Dose (TEDE) in mrem for 1 hour (EPIP-09, Manual Dose Calculation Worksheet, Line 16) for the site boundary 2, 5 and 10 miles.

15. Message Received By

Enter the name of the State Warning Point Duty Officer or the individual that receives the notification. Enter the time at the State Warning Point (request it from the Duty Officer) and indicate the date the call is completed.

END OF ATTACHMENT 1A

/R6

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs) /R6
(Page 1 of 5)

NOTE

- Initial notification from the Control Room may utilize PARs based on plant conditions.
- Once dose assessment begins, PARs should be made utilizing all available data including off-site dose projections, plant conditions and field monitoring data.
- **Both plant conditions and off-site doses shall be considered for PARs.**
- The most conservative recommendations should be made.
- If it is anticipated that a threshold for a PAR will be exceeded, it is neither necessary nor desirable to wait until the threshold is exceeded to make that PAR.
- ¶12 Conditions (plant information, dose projections and field monitoring results) are to be continually assessed and PARs expanded, as necessary, to ensure that adequate (most conservative) PARs are issued.
- ¶12 Previously issued PARs, unless found to be less conservative, are to remain in effect until the threat is fully under control and the event is being de-escalated.
- ¶12 Only State and County officials can implement, change and/or terminate protective actions. /R6

1. PAR Flowchart

   A. PARs Based on Plant Conditions

      1. Begin in the upper left hand corner of the chart by answering the General Emergency (GE) question.

      2. Correctly answer the questions until you reach one of the boxes that provides PAR information based on plant conditions.

      3. If there is no release, Then go to the PAR Worksheet and fill-in the PARs based on plant conditions. The sectors affected can be determined by referring to number 8, Weather Data, in Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. /R6

      4. If a release is involved, Then go to Section B, PARs Based on Off-site Dose, below.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 2 of 5)

1. (continued)

NOTE

- If the Class A Model printout, State Notification Form Summary Sheet is available, it should be used to compare dose-based PARs against PARs based on plant conditions.
- Calculated off-site doses should be compared to field monitoring data when determining PARs.

B. PARs Based on Off-site Dose

1. PARs are based on the Total Effective Dose Equivalent (TEDE or total dose) and / or the Committed Dose Equivalent (CDE, thyroid dose). Do NOT use dose rate values.

2. If using the Class A Model, Then in Forecast Mode, print the State Notification Form Summary for computer generated PARs.

a. Go to Section C, PAR Worksheet

3. If using EPIP-09, Off-site Dose Calculations, Then calculate TEDE and CDE in accordance with the procedure.

a. Compare the TEDE dose at 1 mile with the values on the Flowchart. Enter the chart at the appropriate dose level by determining if the dose is between 500 and 999 mrem or between 1000 and 4999 mrem or 5000 mrem or greater.

b. From the selected dose level, move to the right on the chart to the first column, 0-2 miles. The PAR provided corresponds to the calculated TEDE at 1 mile.

c. Enter the PAR in the 0-2 miles block on the TEDE DOSE table below the PAR Flowchart. The sectors affected can be determined by referring to number 8, Weather Data, in Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. /R6

d. Continue to determine the corresponding PAR at 2-5 miles using the calculated 2 mile TEDE, at 5-10 miles using the calculated 5 mile TEDE and the 10 miles plus (To Be Determined (TBD) distance) using the calculated 10 mile TEDE, as necessary.

e. Enter the PAR information in the appropriate blocks of the TEDE DOSE table.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 3 of 5)

1. B. 3. (continued)

f. Follow the same methodology for determining the PARs corresponding to the calculated CDE values beginning with the calculated value at 1 mile.

g. Enter each of the determined PARs in the CDE (Thyroid) DOSE table below the PAR Flowchart.

h. Go to Section C, PAR Worksheet.

C. PAR Worksheet

1. Fill-in the time / date and emergency class.

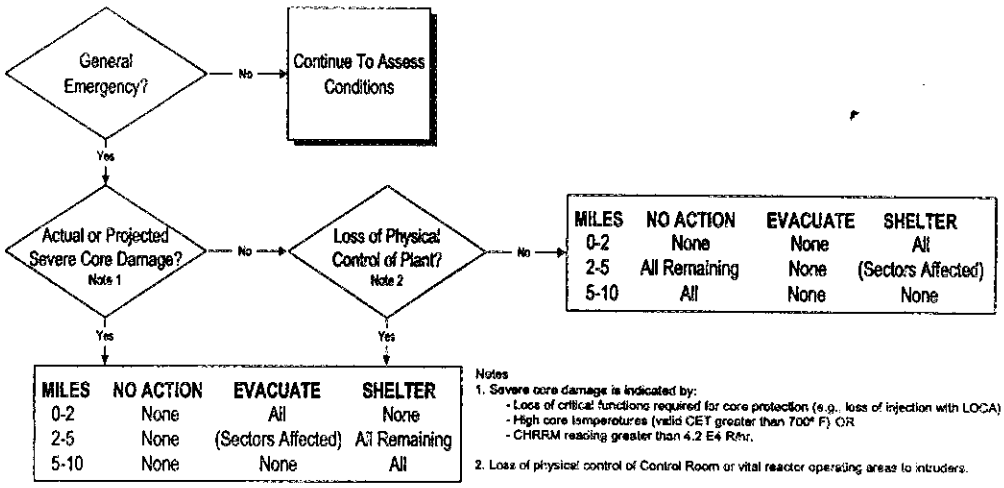
2. In Part A, determine the most conservative PARs by comparing the PARs based on plant conditions against those based on off-site dose. It is important to compare PARs at each distance (0-2, 2-5, 5-10) because the basis of the most conservative PAR could be different at different distances.

3. Enter the most conservative PARs into the table in Part B, Protective Actions Recommended by FPL. Use the word(s) NONE, ALL, ALL REMAINING or list the individual affected sectors by letter.

4. Obtain review and approval.

5. Transfer the approved PARs to the Florida Nuclear Plant Emergency Notification Form.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 4 of 5)



Notes
1. Severe core damage is indicated by:
   - Loss of critical functions required for core protection (e.g., loss of injection with LOCA) OR
   - High core temperatures (valid CET greater than 700° F) OR
   - CHRRM reading greater than 4.2 E4 R/hr.
2. Loss of physical control of Control Room or vital reactor operating areas to intruders.

PARs Based on Off-Site Dose
(For use with manual dose calculation only. Not to be completed when Class A Model is used)

TEDE DOSE	CDE (Thyroid) DOSE		0-2 Miles Use 1 mile value	2-5 Miles Use 2 mile value	5-10 Miles Use 5 mile value (Beyond 10 miles Use this Column and the 10 mile Dose Value)
TEDE 500*-999 mrem	CDE 1000*-4999 mrem	→	Shelter All	Shelter (Sectors Affected)	Shelter (Sectors Affected)
TEDE 1000-4999 mrem	CDE 5000-24999 mrem	→	Evacuate All	Evacuate (Sectors Affected) + Shelter All Remaining	Evacuate (Sectors Affected) + Shelter All Remaining
TEDE 5000 mrem or Greater	CDE 25000 mrem or Greater	→	Evacuate All	Evacuate All	Evacuate (Sectors Affected) + Shelter All Remaining

*Dose below this level requires NO ACTION

TEDE DOSE

Use the following terms in this table: **NONE, ALL, ALL REMAINING or fill in the letters of the sectors affected.**

Miles	NO ACTION	EVACUATE	SHELTER
0-2			
2-5			
5-10			
> 10			

CDE (Thyroid) DOSE

Use the following terms in this table: **NONE, ALL, ALL REMAINING or fill in the letters of the sectors affected.**

Miles	NO ACTION	EVACUATE	SHELTER
0-2			
2-5			
5-10			
> 10			

(P/EP/EPIP-08/APP C-R1)

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 5 of 5)

PAR WORKSHEET

Time / Date ______________________ Emergency Class: ☐ **SAE** ☐ **GE**

A. PAR Comparison

After comparing the possible recommendations from the PARs flowchart, the most conservative PARs are based on: (check one)

☐ PLANT CONDITIONS ☐ OFF-SITE DOSE

B. Protective Actions Recommended by FPL:

Use the following terms in this table: **NONE, ALL, ALL REMAINING Or fill in the letters of the sectors affected.**

	NO ACTION SECTORS	EVACUATE SECTORS	SHELTER SECTORS
0-2 miles			
2-5 miles			
5-10 miles			
10-TBD miles*			

*If necessary, add to State Notification Form.

Control Room

Signature_______________________________________
Emergency Coordinator

Technical Support Center

Signature ___________________________ ___________________________
TSC EC Assistant / Logkeeper TSC HP Supervisor or TSC Chemistry Supervisor

---

Emergency Operations Facility

Signature ___________________________ ___________________________
EOF RM OPS Advisor / Logkeeper EOF HP Manager

END OF ATTACHMENT 2

/R6

ATTACHMENT 3
¶8 NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET
(Page 1 of 2)

NRC FORM 361 (12-2000)

U.S. NUCLEAR REGULATORY COMMISSION OPERATIONS CENTER

REACTOR PLANT EVENT NOTIFICATION WORKSHEET

EN #

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #

EVENT TIME & ZONE	EVENT DATE	POWERMODE BEFORE	POWERMODE AFTER

EVENT CLASSIFICATIONS		1-Hr. Non-Emergency 10 CFR 50.72(b)(1)				
GENERAL EMERGENCY	GEN/AAEC	TS Deviation	ADEV	(v)(A)	Safe S/D Capability	AINA
SITE AREA EMERGENCY	SIT/AAEC	**4-Hr. Non-Emergency 10 CFR 50.72(b)(2)**		(v)(B)	RHR Capability	AINB
ALERT	ALE/AAEC	(i) TS Required S/D	ASHU	(v)(C)	Control of Rad Release	AINC
UNUSUAL EVENT	UNU/AAEC	(iv)(A) ECCS Discharge to RCS	ACCS	(v)(D)	Accident Mitigation	AIND
50.72 NON-EMERGENCY	(see next columns)	(iv)(B) RPS Actuation (scram)	ARPS	(xii)	Offsite Medical	AMED
PHYSICAL SECURITY (73.71)	DDDD	(xi) Offsite Notification	APRE	(xiii)	Loss Comm/Asmt/Resp	ACOM
MATERIAL/EXPOSURE	B???	**8-Hr. Non-Emergency 10 CFR 50.72(b)(3)**		**60-Day Optional 10 CFR 50.73(a)(1)**		
FITNESS FOR DUTY	HFIT	(ii)(A) Degraded Condition	ADEG	Invalid Specified System Actuation		AINA
OTHER UNSPECIFIED REQMT.	(see last column)	(ii)(B) Unanalyzed Condition	AUNA	**Other Unspecified Requirement (Identify)**		
INFORMATION ONLY	NNF	(iv)(A) Specified System Actuation	AESF			NONR
						NONR

DESCRIPTION

Include: Systems affected, actuations and their initiating signals, causes, effect of event on plant, actions taken or planned, etc. (Continue on back)

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT			
STATE(s)			
LOCAL			
OTHER GOV AGENCIES			
MEDIA/PRESS RELEASE			

ANYTHING UNUSUAL OR NOT UNDERSTOOD? ☐ YES (Explain above) ☐ NO

DID ALL SYSTEMS FUNCTION AS REQUIRED? ☐ YES ☐ NO (Explain above)

MODE OF OPERATION UNTIL CORRECTED:	ESTIMATED RESTART DATE:	ADDITIONAL INFO ON BACK ☐ YES ☐ NO

ATTACHMENT 3
¶8 NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET
(Page 2 of 2)

RADIOLOGICAL RELEASES: CHECK OR FILL IN APPLICABLE ITEMS (specific details/explanations should be covered in event description)

LIQUID RELEASE	GASEOUS RELEASE	UNPLANNED RELEASE	PLANNED RELEASE	ONGOING	TERMINATED
MONITORED	UNMONITORED	OFFSITE RELEASE	T.S. EXCEEDED	RM ALARMS	AREAS EVACUATED
PERSONNEL EXPOSED OR CONTAMINATED		OFFSITE PROTECTIVE ACTIONS RECOMMENDED		*State release path in description	

	Release Rate (Ci/sec)	% T.S. LIMIT	HOO GUIDE	Total Activity (Ci)	% T.S. LIMIT	HOO GUIDE
Noble Gas			0.1 Ci/sec			1000 Ci
Iodine			10 uCi/sec			0.01 Ci
Particulate			1 uCi/sec			1 mCi
Liquid (excluding tritium and dissolved noble gases)			10 uCi/min			0.1 Ci
Liquid (tritium)			0.2 Ci/min			5 Ci
Total Activity						

	PLANT STACK	CONDENSER/AIR EJECTOR	MAIN STEAM LINE	SG BLOWDOWN	OTHER
RAD MONITOR READINGS					
ALARM SETPOINTS					
% T.S. LIMIT (if applicable)					

RCS OR SG TUBE LEAKS: CHECK OR FILL IN APPLICABLE ITEMS: (specific details/explanations should be covered in event description)

LOCATION OF THE LEAK (e.g., SG #, valve, pipe, etc.)				
LEAK RATE	UNITS: gpm/gpd	T.S. LIMITS	SUDDEN OR LONG-TERM DEVELOPMENT	
LEAK START DATE	TIME	COOLANT ACTIVITY AND UNITS:	PRIMARY	SECONDARY
LIST OF SAFETY RELATED EQUIPMENT NOT OPERATIONAL				

EVENT DESCRIPTION (Continued from front)

END OF ATTACHMENT 3

¶7 ATTACHMENT 3A

DIRECTIONS FOR COMPLETING THE NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET

(Page 2 of 2)

B. (continued)

> **NOTE**
> No other blocks in the upper half of the form are required.

9. Description - provide a written description of the event.

> **NOTE**
> Check the blocks in the lower portion of the form based on current conditions.

10. Mode of operation until corrected - provided if known.

11. Estimate for restart date - enter "unknown".

12. Additional info on Page 2 - enter yes or no.

C. Reactor Plant Event Notification Worksheet, Page 2

1. Fill in as much of the information on the form as is immediately available - do not create undue delay in making the notification. This information can be gained once the open line of communication is established.

D. Approval

1. Information entered on the worksheet shall be reviewed and approved by the EC or RM (if used in the EOF), prior to transmission.

2. The EC / RM may initial on the worksheet to indicate approval. There is no formal sign-off location on the worksheet.

END OF ATTACHMENT 3A

HPPs

FPL	ST. LUCIE PLANT HEALTH PHYSICS PROCEDURE NON-SAFETY RELATED	Procedure No. HPP-30 Current Revision No. 33 Effective Date 02/11/04

Title:

PERSONNEL MONITORING

Responsible Department: **HEALTH PHYSICS**

REVISION SUMMARY:

Revision 33 - Incorporated PCR 04-0110 to change / clarify wording on forms. (E. Vasilas, 01/30/04)

Revision 32A - Incorporated PCR 03-3499 for CR 03-3467 to change Safety Related to Non-Safety Related. (Bonnie Wooldridge, 12/04/03)

Revision 32 - Incorporated PCR 03-2605 to add instructions regarding EPD dose. (Doc Mercer, 09/18/03)

Revision 31A - Incorporated PCR 03-2321 to add correction factor of 3. (A. B. Sexton, 08/06/03)

Revision 31 - Incorporated PCR 03-1699 to delete Appendix 10, change TLD monitoring period, revise instructions in Appendix 2, delete form HPP-30.6, revise HPP-30.21 & HPP-30.22, and remove requirement for annual whole body count. (B. Johnson, 06/24/03)

Revision 30 - Incorporated PCR 02-2926 for CR 02-2522 to revise forms HPP-30.3, HPP-30.17, HPP-30.21, and HP-30.22. Added form HPP-30.25. Added instructions to Appendix 9. (B.N. Johnson, 03/20/03)

Revision 29 - Changed dose calculation forms out for form used by PTN - standardization issue. (H. Mercer, 11/20/02)

Revision	FRG Review Date	Approved By	Approval Date
0	12/14/93	C. L. Burton Plant General Manager	12/29/93
Revision	**FRG Review Date**	**Approved By**	**Approval Date**
33		N/A Plant General Manager	
		S. Wisla Designated Approver	01/30/04
		N/A Designated Approver (Minor Correction)	

S___OPS	
DATE	
DOCT	PROCEDURE
DOCN	HPP-30
SYS	
COM	COMPLETED
ITM	33

TABLE OF CONTENTS

APPENDIX 1
DETERMINATION OF RADIATION EXPOSURE HISTORY
(Page 1 of 3)

7.0 General

This appendix contains the requirements for obtaining a person's radiation exposure history and their dose limits.

7.1 §1,2 Prior to allowing a person to exceed the following site annual dose limits, a record of current year exposure (NRC Form 4) shall be completed.

1.	TEDE	0.45 rem
2.	Skin	4.5 rem
3.	Extremity	4.5 rem
4.	Lens of eye	1.35 rem
5.	CDE	4.5 rem

1. §1 If radiation exposure for the current calendar year is known to exist and results are not yet available, the form is incomplete.

2. Upon receipt of the missing results and their inclusion into the person's records, the NRC Form 4 shall be considered complete.

NOTE
There is no requirement to re-evaluate separate external dose equivalents and internal committed dose equivalents assessed under 20.1-20.601.

3. §1 A NRC Form 4 for current year exposure shall be considered complete if the following conditions are met:

A. The person presents a signed statement containing documentation of their current year exposure that includes all the information required by the NRC Form 4, or

B. Documentation from the person's most recent employer for work involving radiation exposure that contains all the person's current year periods of exposure and includes all the information required by the NRC Form 4, or

APPENDIX 1
DETERMINATION OF RADIATION EXPOSURE HISTORY
(Page 2 of 3)

7.1 3. (continued)

C. A final exposure report from each licensee where the person received occupational exposure during the current calendar year. The report shall be a NRC Form 5 or equivalent or shall contain all the information required by the NRC Form 5.

D. If it is not possible to obtain records of a person's current year exposure, the person's allowable exposure shall be reduced by 1.25 rem for each quarter in which the individual was engaged in activities that could have resulted in occupational exposure.

1. For purposes of this procedure, the inability to obtain exposure records consists of 2 requests for exposure resulting in no response from the licensee where the exposure occurred or where there exists an unresolvable conflict between the person's statement of dose and the reported dose.

2. By regulations, a licensee is allowed 30 days to respond to a request for exposure records.

7.2 §1 Personnel being concurrently monitored at St. Lucie Plant and another facility(s) are considered to have an incomplete NRC Form 4 and shall have their annual doses at St. Lucie Plant limited to the annual doses in 7.1.

7.3 When personnel being monitored for exposure at PSL are required to visit another nuclear facility, they should:

1. Prior to leaving PSL on the visit, report to the HP Dosimetry Section to receive an exit body count and have the appropriate entries made in their exposure record.

2. When leaving the visited facility to return to PSL, they should:

   A. Request the facility to provide an estimate of the person's exposure while at the facility.

   B. Request that the facility send a copy of the person's final exposure determination to PSL HP Dosimetry Section.

7.4 When a person that is being monitored at PSL receives exposure at another nuclear facility, that person shall report to the HP Dosimetry Section as soon as they return to PSL so that the person's exposure records are appropriately updated.

APPENDIX 1

DETERMINATION OF RADIATION EXPOSURE HISTORY

(Page 3 of 3)

> **CAUTION**
>
> §1 IF THE PERSON IN 7.5 BELOW HAS MORE THAN 450 MREM CURRENT YEAR PSL DOSE, THEY CANNOT BE ALLOWED TO ENTER THE RCA UNTIL A RECORD OF THE DOSE RECEIVED WHILE AT THE VISITED FACILITY HAS BEEN RECEIVED BY THE DOSIMETRY SECTION.

7.5 Personnel returning to PSL from another nuclear facility where they were monitored for radiation exposure shall have their guidelines adjusted to the doses contained in 7.1 above except as provided below:

1. Persons returning with a written estimate of exposure shall have the estimate added to their current year dose. The person's current year dose may require adjustment when the final dose results are received.

2. Persons returning with a final record of dose shall have their current year dose updated.

3. Persons returning and not providing a written estimate or final dose record shall not be allowed to exceed 450 mrem TEDE PSL dose for the current year.

4. §1 Personnel shall have a completed NRC Form 4 for current calendar year dose prior to being authorized the exposure guideline values listed in Table 1.

7.6 §1 Lifetime exposure histories (exposure received before the current year) should be obtained for all FPL personnel having a completed current year NRC Form 4. The steps in 7.1.3 of this appendix should be followed in compiling the exposure histories.

Exposure histories for temporary workers and contractors should be obtained for the current year.

7.7 Completion of lifetime exposure histories on personnel previously monitored at PSL need only to update their Form 4 to account for the exposure received since last monitored at PSL.

7.8 Annually each FPL employee who is currently being monitored at PSL will complete Form HPP-30.21. This form is used to update the individual's exposure history file.

END OF APPENDIX 1

APPENDIX 6
AUTHORIZATION TO EXCEED ADMINISTRATIVE GUIDELINES
(Page 1 of 2)

7.0 Instructions: General

This appendix contains the instructions necessary to authorize persons to exceed administrative exposure guidelines.

7.1 When an individual is expected to exceed his exposure guidelines, a Radiation Exposure Extension Request (HPP-30.12) shall be completed.

> **NOTE**
> A request for this extension should be made at least 8 hours prior to expecting the individual to exceed the administrative exposure guidelines.

7.2 Before an extension is authorized, an investigation into the individual's radiation exposure history shall be conducted.

7.3 §2 Authorization may be granted as long as the exposure limits in 10 CFR 20 are not exceeded.

7.4 §2 Administrative Exposure Guidelines

1. Any individual with non-documented current year exposure shall have their annual exposure guidelines set to the following doses:

   A. TEDE 0.45 rem

   B. Lens of eye 1.35 rem

   C. Skin 4.5 rem

   D. Extremity 4.5 rem

   E. CDE 4.5 rem

> **NOTE**
> Circumstances requiring extending an annual guideline should be documented. The documentation should include justification for the guideline extension and an evaluation of the individual's lifetime dose with respect to the individual's age.

2. §2 At the beginning of each calendar year, each person with a completed current year Form 4 may be allowed to receive up to 1000 mrem TEDE.

APPENDIX 6

AUTHORIZATION TO EXCEED ADMINISTRATIVE GUIDELINES

(Page 2 of 2)

7.4 (continued)

3. §2 An individual with a complete Form 4 for the current year may have their site annual dose guideline extended to 2500 mrem / yr by the Health Physics Department Head and the Plant General Manager.

4. §2 An individual with a complete Form 4 for the current year may have their site annual dose guideline extended to greater than 2500 mrem by the Health Physics Department Head, the Plant General Manager and Site Vice President.

5. An individual with a complete Form 4 for the current year may have their annual dose guideline extended to greater than 3000 mrem / year, by the Health Physics Department Head and Plant General Manager.

6. §2 The Plant General Manager and Health Physics Department Head shall approve and the Site Vice President shall authorize all dose extensions above the annual guideline of 4500 mrem TEDE.

7. §2 After the TLD has been read and if the site annual TEDE is less than 2500 mrem, the individual may be allowed back into the Radiation Controlled Area until his / her annual TEDE (TLD results and MERLIN doses) reaches 2500 mrem.

8. Upon reaching an annual TEDE of 2500 mrem, the individual shall not be allowed to enter the Radiation Controlled Area, unless extended > 2500 mrem / yr as required by step 7.4.4 above.

9. §2 FPL permanent employees should have their lifetime dose (TEDE) restricted to 1XN where N is the employee's age. If an FPL employee has a lifetime dose in excess of 1XN, the employee should have their annual dose guideline set to 1.0 Rem (TEDE).

7.5 The administrative guidelines below should be followed when monitoring shallow (skin and extremity) dose:

1. When a person's annual shallow dose (skin or extremity) reaches 20 rem, the person's TLD should be read to update shallow dose results.

2. §2 When a person's annual shallow dose (skin or extremity) reaches 25 rem site dose or 45 rem all sites, the person shall be restricted from further entry into the RCA for the remainder of the year.

END OF APPENDIX 6

APPENDIX 11
EMERGENCY EXPOSURES
(Page 1 of 1)

7.0 Instructions: General

This appendix contains the instructions necessary for authorizing emergency exposures and incorporating the emergency exposures into a persons exposure records.

7.1 §10 The provisions of this appendix shall apply only to those situations in which exposure is received through the implementation of the St. Lucie Plant Emergency Plan and its implementing procedures.

7.2 §9 Each Emergency Response Person (ERP) exposure guideline is limited to 5.0 rem TEDE and 50 rem CDE unless exposures in excess of these guidelines is authorized by the Emergency Coordinator.

1. Any existing annual dose shall not be included in the ERP exposure guidelines.

2. The exposure guidelines are in effect for the duration of the emergency response as determined by FPL management.

7.3 ERP that have reached their guideline or extended guideline shall be removed from further exposure and have their dose determined.

7.4 §1,2 ERP emergency doses shall be evaluated against the regulatory limits in 10 CFR 20.1201.

1. Emergency doses shall be added to the occupational doses for the year in which the emergency doses are incurred.

2. ERP receiving a combination of occupational and emergency doses in excess of the limits in 20.1201 shall be restricted from any further exposure for the remainder of the year.

3. Any combined annual doses in excess of the limits in 20.1201 shall be accounted for as prescribed in 20.1206(e).

7.5 §1,6,7 Combined doses in excess of 20.1201 shall be reported to the Nuclear Regulatory Commission in accordance with established plant procedures.

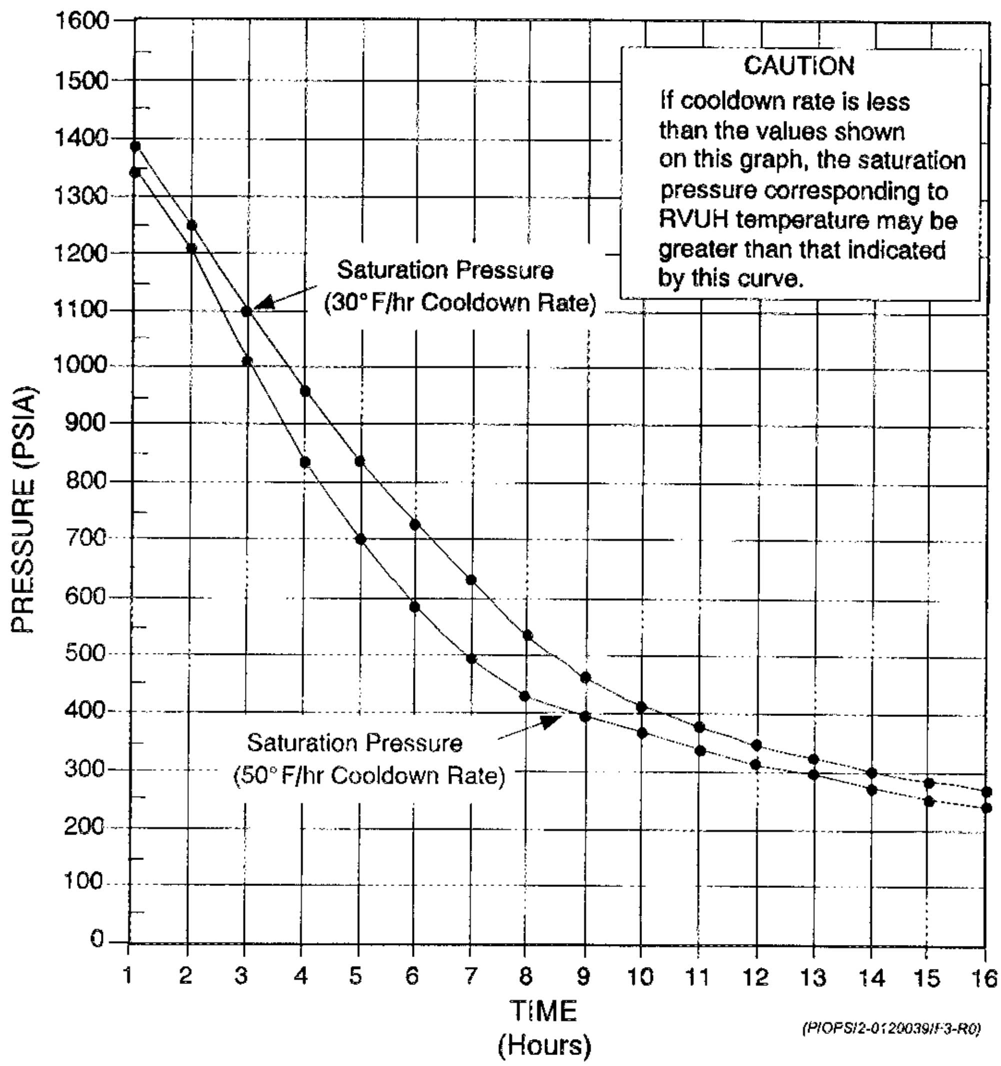
END OF APPENDIX 11

Unit 2 OPS Procedures

FIGURE 3

UPPER HEAD SATURATION PRESSURE VS. TIME

(Page 1 of 1)



Unit 1 Tech Specs

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor driven feedwater pumps, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two motor driven pumps and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a discharge pressure of ≥ 1342 psig on recirculation flow.

2. Verifying that the steam turbine driven pump develops a discharge pressure of ≥ 1342 psig on recirculation flow.*

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* When not in MODES 1, 2 or 3, this surveillance shall be performed within 24 hours after entering MODE 3 and prior to entering MODE 2.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flowpath actuates to its correct position upon receipt of the Auto Start actuation test signal.

2. Verifying that each auxiliary feedwater pump starts automatically as designed upon receipt of the Auto Start actuation test signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 16,450 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action f. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*; restore the diesel generator to OPERABLE status within 14 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

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* If the absence of any common-cause failure cannot be confirmed, this test shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION (continued)

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

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* If the absence of any common-cause failure cannot be confirmed, this test shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION (continued)

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

f. With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units, Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying fuel level in the engine-mounted fuel tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank,

4. Verifying the diesel starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds**. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal**. The diesel generator shall be started for this test by using one of the following signals:

    a) Manual/Local

    b) Simulated loss-of-offsite power by itself.

    c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

    d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 3500 kW in accordance with the manufacturer's recommendations and operates within a load band of 3300 to 3500 kW*** for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water:

1. From the engine-mounted fuel tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2. From the storage tank at least once per 92 days.

** The diesel generator start (10 sec.) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other diesel generator starts for the purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer so that mechanical stress and wear on the diesel generator is minimized.

*** The indicated load band is meant as guidance to avoid routine overloading. Variations in loads in excess of the band due to changing bus loads shall not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

c. By sampling new fuel in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1. By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

   a) API Gravity within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

   b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

   c) A flash point equal to or greater than 125°F, and

   d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-83 and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-83, Method A, or Annex A-2.

e. At least once per 18 months during shutdown by:

1. DELETED

2. Verifying generator capability to reject a load of greater than or equal to 600 hp while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3. Simulating a loss of offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz during this test.

4. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts**** on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady state generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

5. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the auto-sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz during this test.

   c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying the diesel generator operates for at least 24 hours****. During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3960 kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3300 to 3500 kW#. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

7. Verifying that the auto-connected loads do not exceed the 2000-hour rating of 3730 kW.

8. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.

   b) Transfer its load to the offsite power source, and

   c) Be restored to its standby status.

9. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

10. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks to each diesel via the installed cross connection lines.

11. Verifying that the automatic load sequence timers are operable with the interval between each load block within $\pm$ 1 second of its design interval.

f. At least once per ten years or after any modification which could affect diesel generator independence by starting**** the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to approximately 900 rpm in less than or equal to 10 seconds.

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

g. At least once per ten years by:

1. Draining each fuel storage tank, removing the accumulated sediment and cleaning the tank using an appropriate cleaning compound, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to USAS B31.7 Class 3 requirements in accordance with the Inservice Inspection Program.

4.8.1.1.3 Reports – (Not Used)

4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to 215°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 193°F.

b. A setpoint of less than or equal to 530 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than 215°F and less than or equal to 281°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 193°F and less than or equal to 304°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 304°F, MODE 5, and MODE 6 when the head is on the reactor vessel; and the RCS is not vented through greater than a 1.75 square inch vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days; or depressurize and vent the RCS through greater than a 1.75 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, restore at least one PORV to operable status or complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within 24 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 Each PORV shall be demonstrated OPERABLE by:

a. Verifying the PORV isolation valve is open at least once per 72 hours; and

b. Performance of a CHANNEL FUNCTION TEST, but excluding valve operation, at least once per 31 days; and

c. Performance of a CHANNEL CALIBRATION at least once per 18 months.

Unit 2 Tech Specs

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Unless the RCS is depressurized and vented by at least 3.58 square inches, at least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 470 psia and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia.

c. One PORV with a lift setting of less than or equal to 470 psia and with its associated block valve open in conjunction with the use of one SDCRV with a lift setting of less than or equal to 350 psia. This combination may only be used to satisfy LTOP when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

APPLICABILITY: MODES 4#, 5 and 6.

ACTION:

a. With either a PORV or an SDCRV being used for LTOP inoperable, restore at least two overpressure protection devices to OPERABLE status within 7 days or:

   1. Depressurize and vent the RCS with a minimum vent area of 3.58 square inches within the next 8 hours; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.

b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either:

   1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

With cold leg temperature within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. In the event either the PORVs, SDCRVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. In addition to the requirements of the Inservice Testing Program, operating the PORV through one complete cycle of full travel at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

c. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

d. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	Cold Leg Temperature, F° During Cooldown
≤ 21.7	≤ 247	≤ 230

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
≤ 21.7	165	165