SL-2019-3

SRO Written Answer Key

76	C
77	A
78	D
79	A
80	A
81	A
82	A
83	C
84	A
85	C
86	C
87	D
88	A
89	A
90	D
91	B
92	B
93	B
94	C
95	A
96	C
97	D
98	B
99	A
100	C

76.

Given the following conditions:

- Unit 1 is at 100% power
- Engineering reports that Trip Circuit Breaker (TCB) #1 Trip Coil is NON – Qualified / Inoperable

Subsequently:

- An Automatic Rx Trip occurs

Which ONE of the following describes the MINIMUM required actions in accordance with ADM-11.16, Transient Procedure Use and Adherence and TS 3.3.1, Reactor Protective Instrumentation PRIOR to the trip and how the TCB positions are verified AFTER the trip?

The crew is required to OPEN ____(1)____.

TCB Breaker Position is verified by observing ALL ___(2)____ Lights LIT.

A. (1) TCB # 1 ONLY
   (2) GREEN Lights LIT

B. (1) TCB # 1 ONLY
   (2) WHITE Lights LIT

C. (1) TCB # 1 and TCB # 5
   (2) GREEN Lights LIT

D. (1) TCB # 1 and TCB # 5
   (2) WHITE Lights LIT

77.

Given the following conditions:

- Unit 1 is at 100% power
- V2504, RWT to Charging Pump Suction is out of service for maintenance
- The crew is performing a blended makeup to the VCT per 1-NOP-02.24, Boron Concentration Control

Subsequently:

- FCV-2210Y, Boric Acid Flow Control Valve fails closed
- The crew enters 1-AOP-02.01, Boron Concentration Control System Abnormal Operations

Which ONE of the following completes the statements below?

1-OSP-02.07, Boration Flowpath Sources surveillance ____(1)____ MET.

If the crew was required to perform a downpower, in accordance with 1-AOP-02.01, the crew would borate via ____(2)____.

A. (1) is

(2) V2514, Emergency Borate

B. (1) is

(2) V2174, Emergency Boration from BAM Pump Discharge

C. (1) is NOT

(2) V2514, Emergency Borate

D. (1) is NOT

(2) V2174, Emergency Boration from BAM Pump Discharge

78.

Given the following conditions:

- Unit 2 is in Mode 5
- RCS is solid
- RCS Pressure is 150 psia
- RCS temperature is 127 °F
- A cooldown is in progress
- Shutdown Cooling Trains A and B are in service

Subsequently:

- PIC-2201, Letdown Pressure Controller output fails LOW
- RCS Pressure is RISING
- The RCS Hot Leg Suction Isolation Valves have automatically CLOSED

Which ONE of the following describes the PORV lift setpoint for this condition and the status of TS 3.4.9.3, Overpressure Protection Systems?

The PORVs will lift at ____(1)____ psia.

TS 3.4.9.3 ____(2)____ MET.

(REFERENCE PROVIDED)

A. (1) 350
   (2) is

B. (1) 350
   (2) is NOT

C. (1) 490
   (2) is

D. (1) 490
   (2) is NOT

79.

Given the following conditions:

- Unit 2 has tripped from 100% power
- SBCS failed to operate
- The crew is performing 2-EOP-01, SPTAs, step 8, Containment Conditions

Given the plant indications from the Safety Parameter Display System (SPDS), Distributed Control System (DCS), and Radiation Monitoring Control System (RMCS), complete the following.

In accordance with EPIP-01, Classification of Emergencies, a(n) ____(1)____ is the highest emergency level threshold met.

In accordance with EPIP-08, Off-Site Notifications and Protective Action Recommendations, a release ____(2)____ occurring?

(REFERENCE PROVIDED)

A. (1) ALERT

(2) is

B. (1) ALERT

(2) is NOT

C. (1) SITE AREA EMERGENCY

(2) is

D. (1) SITE AREA EMERGENCY

(2) is NOT

80.

Given the following conditions:

- Unit 1 is at 100% power
- 1A and 1B ICW pumps are in service

Subsequently:

- The Nuclear Plant Operator reports a significant ICW leak from SS-21-4A, “A” TCW Heat Exchanger Strainer

Which ONE of the following completes the statements below?

Based on the leak location, ____(1)____ Intake Cooling Water (ICW) header pressure(s) will LOWER.

IF, MV-21-3, “A” ICW Train to TCW HXS, breaker were to trip OPEN prior to full valve closure during leak isolation actions, Entry into Tech Spec LCO 3.7.4.1, Intake Cooling Water System, ____(2)____ required

A. (1) ONLY the A

(2) is

B. (1) ONLY the A

(2) is NOT

C. (1) BOTH the A and the B

(2) is

D. (1) BOTH the A and the B

(2) is NOT

81.

Given the following conditions:

- Unit 1 is at 100% power

Subsequently:

- Instrument Air (IA) pressure is 80 psig and LOWERING
- 1-AOP-18.01, Instrument Air Malfunction is entered

Which ONE of the following completes the statements below?

If IA pressure continues to lower, a Reactor Trip is required at ____(1)____.

AFTER SPTAS are completed and during performance of 1-EOP-02, Reactor Trip Recovery, in accordance with ADM-11.16, Transient Procedure Use and Adherence, 1-AOP-18.01, is used ____(2)____.

A. (1) 60 psig

(2) in parallel with 1-EOP-02

B. (1) 60 psig

(2) after 1-EOP-02 is exited

C. (1) 75 psig

(2) in parallel with 1-EOP-02

D. (1) 75 psig

(2) after 1-EOP-02 is exited

82.

Given the following conditions:

- Unit 2 is at 100% power
- Pressurizer Level Control is selected to Channel “Y”
- Pressurizer Backup Heaters B1 are out of service due to a breaker failure

Subsequently:

- Pressurizer Level Transmitter, LT-1100Y indicates 0%
- Annunciator H-18, PZR Channel Y Level HI/LO is LIT
- 2-AOP-01.10, Pressurizer Pressure and Level, Att. 5, Recovering Power to Pressurizer Heaters has been completed

Which ONE of the following describes the heaters that can be recovered and the applicable Tech Spec requirements?

Based on the given conditions ____(1)____ can be recovered. Tech Spec 3.4.3, Pressurizer, Action (b) ____(2)____ required to be entered.

(REFERENCE PROVIDED)

A. (1) B2 and B3 heaters ONLY

(2) is

B. (1) B2 and B3 heaters ONLY

(2) is NOT

C. (1) B2, B3, B4, B5 and B6 heaters

(2) is

D. (1) B2, B3, B4, B5 and B6 heaters

(2) is NOT

83.

Given the following conditions:

- Unit 2 is shutdown performing a cooldown in accordance with 2-GOP-305, Reactor Plant Cooldown - Hot Standby to Cold Shutdown
- RCS Temperature is 320 °F
- 2B1 and 2B2 RCPs are in operation
- 1 ADV on each S/G is in operation
- 2A and 2B Charging Pumps are running

Subsequently:

- Annunciator L-31, Boron Concentration LOW Channel 1 is LIT

Which ONE of the following completes the statements below?

The Source Range NI has failed ____(1)____.

Given that ONLY 1 Source NI has failed, in accordance with UFSAR, 13.7.2.4, Backup Boron Dilution Detection Sampling, the US ____(2)____ required to direct chemistry to sample the RCS for boron concentration.

A. (1) LOW
   (2) is

B. (1) LOW
   (2) is NOT

C. (1) HIGH
   (2) is

D. (1) HIGH
   (2) is NOT

84.

Given the following conditions:

- Fuel Handling Radiation Monitors indicate on the RMCS:

SA		SB	
GAG-007	DRK BLUE	GAG-008	GREEN
GAG-009	DRK BLUE	GAG-010	GREEN
GAG-011	GREEN	GAG-012	GREEN

Subsequently:

- A Unit 2 spent fuel assembly has been dropped in the Fuel Handling Building

SA		SB	
GAG-007	DRK BLUE	GAG-008	RED
GAG-009	DRK BLUE	GAG-010	YELLOW
GAG-011	RED	GAG-012	RED

Which ONE of the following completes the statements below?

In accordance with TS 3.3.3.1, Radiation Monitoring Instrumentation, the MINIMUM number of OPERABLE Fuel Storage Pool Area Monitors ____(1)____ MET.

After the fuel assembly drops, the Spent Fuel Pool exhaust transfers to ____(2)____ train(s) of the Shield Building Ventilation System.
(REFERENCE PROVIDED)

A. (1) is
   (2) BOTH

B. (1) is
   (2) ONLY ONE

C. (1) is NOT
   (2) BOTH

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

85.

Given the following conditions:

- Unit 1 is experiencing a LOCA
- 1-EOP-03, LOCA is in progress
- The following parameters are observed

Time	0045	0100	0115
CET	430 °F	445 °F	446 °F
RCS Pressure	550 psia	550 psia	550 psia
Pzr Level	0%	0%	0%
ECCS Flow	500 gpm	350 gpm	340 gpm
Containment Temp	174 °F	176 °F	178 °F

Which ONE of the following completes the statement below?

In accordance with ADM-11.16, Transient Procedure Use and Adherence, at time ___(1)____ the crew MUST exit 1-EOP-03 and implement 1-EOP-15, Functional Recovery, Success Path ____(2)____.

REFERENCE PROVIDED

A. (1) 0100

(2) IC-2 – Inventory Control

B. (1) 0100

(2) PC-3 - Saturated Pressure Control

C. (1) 0115

(2) IC-2 – Inventory Control

D. (1) 0115

(2) PC-3 - Saturated Pressure Control

86.

Given the following conditions:

- Unit 2 is at 100% power
- RPS Channel “MA” Linear Range Nuclear Instrument has Failed High

Which ONE of the following describes the RPS Bistables that have tripped and the required actions in accordance with 2-AOP-99.01, Loss of Tech Spec Instrumentation?

Bistables HI POWER, (and) LOC PWR DEN ____(1)____have tripped.

Azimuthal Power Tilt per TS 3.2.4, Azimuthal Power Tilt - Tq ____(2)____ required to be determined at least once per 12 hours.

A. (1) ONLY

(2) is

B. (1) ONLY

(2) is NOT

C. (1) and TM/LO PRESS

(2) is

D. (1) and TM/LO PRESS

(2) is NOT

87.

Given the following conditions:

- Unit 2 is at 100% power
- Annunciator S-9 RAS Channel A/B Actuation is LIT
- LIS-07-2A thru 2D, RWT Level indicate 34 feet
- MV-07-1A, SUCTION FROM RWT TRAIN A is CLOSED
- MV-07-2A, SUCT FROM CNTMT SUMP A TRAIN is OPEN
- "B" Train RAS components were NOT affected
- 2-AOP-69.01, Inadvertent ESFAS Actuation has been entered

Which ONE of the following completes the statement below?

In accordance 2-AOP-69.01, a Reactor Trip ____(1)____ required.

In accordance with LI-AA-102-1001, Regulatory Reporting, an 8 hour report ____(2)____ required due to the Inadvertent ESFAS actuation.

(REFERENCE PROVIDED)

A. (1) is
   (2) is

B. (1) is
   (2) is NOT

C. (1) is NOT
   (2) is

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

88.

Given the following conditions:

00:00:00	Unit 2 tripped due to a LOCA SIAS has actuated CSAS has actuated
03:00:00	2-EOP-03, LOCA, Check if Containment Spray can be terminated, is in progress
03:30:00	RAS actuates Annunciator S-27, 2A CS Pump Running HDR Press Low Alarms PIS-07-3A, CNTMT Spray Header A Press is oscillating between 60 and 90 psig

Which ONE of the following completes the statements below?

At time 03:00:00, in accordance with 2-EOP-03, Containment Pressure MUST be less than a MAXIMUM of ____(1)____ for Containment Spray to be TERMINATED.

At time 03:30:00, in accordance with 2-EOP-03, ____(2)____ must be secured.

A. (1) 3.5
   (2) CS pumps ONLY

B. (1) 3.5
   (2) CS and HPSI pumps

C. (1) 5.0
   (2) CS pumps ONLY

D. (1) 5.0
   (2) CS and HPSI pumps

89.

Given the following conditions:

- Unit 1 is at 100% power
- Annunciator Q-38, MFIV HCV-09-7 N2 Press Low/DC Failure
- HCV-09-7, MFIV, RED indicating light is LIT
- NPO reports PI-09-13 N2 Hdr to 1A MFIV N2 Accum Press reads 288 psig

Which ONE of the following completes the statements below?

In accordance with ADM-11.16, Transient Procedure Use and Adherence, HCV-09-7 ____(1)____ OPERABLE.

IF the N2 pressure LOWERED to 0 psig, the MFIV will FAIL ____(2)____.

A. (1) is
   (2) AS-IS

B. (1) is
   (2) CLOSED

C. (1) is NOT
   (2) AS-IS

D. (1) is NOT
   (2) CLOSED

90.

Which ONE of the following completes the statements below?

Unit 2 Containment **DESIGN** temperature is ____(1)____.

This is based on a ____(2)____ Break Accident inside Containment.

A. (1) 230 °F

(2) Feed Line

B. (1) 230 °F

(2) Steam Line

C. (1) 264 °F

(2) Feed Line

D. (1) 264 °F

(2) Steam Line

91.

Given the following conditions:

00:00:00	Unit 1 has been at 50% for the previous 7 days to repair the 1A MFP
04:00:00	Unit 1 raised power to 90% Group 7 CEAs are at 131 inches withdrawn
07:01:00	Group 7 CEAs are moved for ASI control The following alarms occur: K-26 , CEDS TROUBLE / CONTINOUS GRIPPER VOLTAGE HIGH I&C Reports that Group 7 CEA #1 ACTM TRBL alarm is FAST flashing and has a missing phase from the power switch

Which ONE of the following completes the statements below?

Assuming NO operator action, between time 04:00:00 and 07:00:00, ____(1)____ reactivity will be added to the core due to the change in Xenon concentration.

In accordance with TS 3.1.3.1, Full Length CEA position and ADM-11.16, Transient Procedure Use and Adherence, the affected CEA ____(2)____ required to be declared INOPERABLE.

A. (1) positive
   (2) is

B. (1) positive
   (2) is NOT

C. (1) negative
   (2) is

D. (1) negative
   (2) is NOT

92.

Given the following conditions:

- Unit 1 is at 100% power

Subsequently:

- LIC-9013A, S/G Level Channel fails LOW
- 1-AOP-99.01, Loss of Tech Spec Instrumentation has been entered
- LOW LVL SG (RPS) has been placed in BYPASS

Which ONE of the following describes the required action(s) and the Tech Spec implication for continued operation?

1-AOP-99.01, requires placing ____(1)____ in BYPASS.

In accordance with TS 3.3.1.1, RPS Instrumentation, with the LOW LVL SG (RPS) remaining in BYPASS, Unit operation ____(2)____ be continued until the next cold shutdown.

A. (1) AFAS-1 ONLY

(2) can

B. (1) AFAS-1 ONLY

(2) can NOT

C. (1) AFAS-1 and AFAS-2

(2) can

D. (1) AFAS-1 and AFAS-2

(2) can NOT

93.

Given the following conditions:

- Unit 1 is in a refueling outage
- Core off-load is in progress
- A fuel assembly is being transported from the core to the Fuel Transfer Basket

Subsequently:

- The containment evacuation alarm sounds
- The refueling machine stops, a Programmable Logic Controller (PLC) failure occurs

Which ONE of the following completes the statements below?

The PLC failure ____(1)____ be overridden by placing the Interlock Override switch in the override position to allow placing the fuel in a SAFE CONDITION.

In accordance with Unit 1 UFSAR 13.8.1.11.1, the Refueling Machine shall have a Maximum OVERLOAD cut off limit of less than ____(2)____ pounds.

A. (1) can

   (2) 2500

B. (1) can

   (2) 3000

C. (1) can NOT

   (2) 2500

D. (1) can NOT

   (2) 3000

94.

Given the following conditions:

- Unit 2 is experiencing a SBLOCA and LOOP
- The 2B EDG failed to start
- An ALERT has been declared in accordance with EPIP-01, Classification of Emergencies
- RAS has actuated
- MV-07-2A and MV-07-2B, Containment Sump Isolation Valves failed to OPEN

Which ONE of the following completes the statements below?

The Control Room will dispatch operators to the ____(1)____ to manually OPEN MV-07-2A and MV-07-2B.

If it is determined that a valve wrench is required, at a MINIMUM____(2)____ approval is required, in accordance with OP-AA-100-1000, Conduct of Operations.

A. (1) 19.5 ft Piping Penetration Room

(2) Shift Manager

B. (1) 19.5 ft Piping Penetration Room

(2) Unit Supervisor

C. (1) -0.5 ft Piping Tunnel

(2) Shift Manager

D. (1) -0.5 ft Piping Tunnel

(2) Unit Supervisor

95.

Which ONE of the following is the design bases event for Limiting Condition of Operation 3.1.1.1, Shutdown Margin?

A. Excessive cooldown resulting from a Main Steam Break at end of core life from 0% power conditions.

B. Excessive cooldown resulting from a Main Steam Break at beginning of core life from 100% power conditions.

C. Positive reactivity addition resulting from a Rod Ejection event at end of core life from 100% power conditions.

D. Positive reactivity addition resulting from a Rod Ejection event at beginning of core life from 0% power conditions.

96.

Which ONE of the following completes the statements below?

In accordance with ADM-25.04, Safety Limits and Limiting Safety Settings, the Variable Power Level – High trip setpoint is operator adjustable and can be set no higher than ____(1)____ above indicated thermal power.

This protects the reactor core during rapid positive reactivity excursions which are too rapid to be protected by ____(2)____.

A. (1) 5.61 %
   (2) Pressurizer Pressure High or Thermal Margin/Low Pressure

B. (1) 5.61 %
   (2) Rate of Change of Power – High Trip or Local Power Density

C. (1) 9.61 %
   (2) Pressurizer Pressure High or Thermal Margin/Low Pressure

D. (1) 9.61 %
   (2) Rate of Change of Power – High Trip or Local Power Density

97.

Given the following conditions:

12/1/2017	08:00:00	Unit 1 is at 100% power The 1B Low Pressure Safety Injection (LPSI) pump was taken out of service for preventative maintenance
	08:00:01	Annunciator B-24, Emerg Dg 1A Fuel Stor Tk Level Low alarms
	08:05:00	The 1A Diesel Fuel Oil Storage Tank is at a level corresponding to 18,500 gallons

Which ONE of the following states the **LIMITING** Tech Spec LCO(s) and action statement that applies?

TS 3.5.2, ECCS Subsystems ____________.

(REFERENCE PROVIDED)

A. Restore 1B LPSI pump to operable status no later than 0800 on 12/4/2017 ONLY

B. Restore 1B LPSI pump to operable status no later than 0800 on 12/8/2017 ONLY

C. AND TS 3.8.1.1.b, A.C. Sources; IMMEDIATELY declare the 1A LPSI pump inoperable and enter TS 3.0.3

D. AND TS 3.8.1.1.b, A.C. Sources; restore 1B LPSI pump to operable status within 4 hours; otherwise enter TS 3.0.3

98.

Given the following conditions:

- Unit 1 is at 100% power
- Gaseous Release Permit # 19-12 was issued to authorize this release
- A gaseous release is in progress from the 1A Gas Decay Tank(GDT) in accordance with 1-NOP-06.20, Controlled Gaseous Batch Release to Atmosphere

Subsequently:

- Radiation Monitor Channel #42, Waste Gas Monitor fails HIGH

Which ONE of the following describes the MINIMUM required actions to properly complete the discharge from the 1A GDT, in accordance with 1-NOP-06.20?

A. Restart the release using permit #19-12

B. Issue a new release permit with independent samples and valve lineup verifications

C. Complete an additional independent GDT release rate calculation, then restart the release using permit #19-12

D. Complete an additional independent GDT grab sample, verifying tank activity has not changed, then restart the release using permit #19-12

99.

Given the following conditions:

- Unit 2 is at 100% power
- A Rapid Downpower has just commenced to remove the 2B1 Circulating Water Pump from service
- A Loss of Annunciators occurs on RTGB 201 thru 206, Panels A thru S
- The crew entered 2-AOP-100.03, Partial or Complete Loss of Annunciators

Which ONE of the following completes the statements below?

A backup power supply ____(1)____ available to the Annunciator Logic Cabinet.

In accordance with EPIP-01, Classification of Emergencies, the EC is required to declare an ____(2)____.

(REFERENCE PROVIDED)

A. (1) is

(2) Unusual Event

B. (1) is

(2) Alert

C. (1) is NOT

(2) Unusual Event

D. (1) is NOT

(2) Alert

100.

Given the following conditions:

- A Probable Airborne Threat **SECURITY EVENT** is in progress
- An Alert has been declared in accordance with EPIP-01, Classification of Emergencies
- The Crew has entered 0-AOP-72.01, Response to Security Events

Which ONE of the following completes the statements below?

The PRIMARY method of STATE notification method is using the ____(1)____.

The NRC must be notified within a MAXIMUM of ____(2)____.

A. (1) EMNET

(2) 15 minutes

B. (1) EMNET

(2) 60 minutes

C. (1) Hot Ring Down Phone (HRD)

(2) 15 minutes

D. (1) Hot Ring Down Phone (HRD)

(2) 60 minutes

78S REFERENCE

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 [redacted] 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 [redacted].

c. One PORV with a lift setting of less than or equal to [redacted] 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 [redacted]. 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.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

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. LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

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 in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

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 in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE.

c. Performance of a CHANNEL CALIBRATION on the PORV actuation channel in accordance with the Surveillance Frequency Control Program.

d. Verifying the PORV isolation valve is open in accordance with the Surveillance Frequency Control Program when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open in accordance with the Surveillance Frequency Control Program* when the vent(s) is being used for overpressure protection.

* 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 in accordance with the Surveillance Frequency Control Program.

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
≤ 47	≤ 246	≤ 224

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	Cold Leg Temperature, °F During Heatup	Cold Leg Temperature, °F During Cooldown
≤ 47	80	132

Evaluate each barrier for Loss or Potential Loss and circle the applicable condition

FISSION PRODUCT BARRIER DEGRADATION TABLE (APPLICABILITY: Modes 1, 2, 3, & 4 ONLY)

FUEL CLAD BARRIER – LOSS	FUEL CLAD BARRIER – POTENTIAL LOSS	REACTOR COOLANT SYSTEM BARRIER – LOSS	REACTOR COOLANT SYSTEM BARRIER – POTENTIAL LOSS	PRIMARY CONTAINMENT BARRIER – LOSS	PRIMARY CONTAINMENT BARRIER – POTENTIAL LOSS
1. Safety Function Status		**1. Safety Function Status**		**1. Safety Function Status**	
1. Core Heat Removal Safety function **NOT** met **AND** entry into procedure 1/2 EOP-15	**1.** RCS Heat Removal Safety function **NOT** met **AND** entry into procedure 1/2 EOP-15	Not Applicable **GUIDANCE BOX FOR SAFETY FUNCTION STATUS IN ALL THREE BARRIERS** If directed to perform any step in 1/2 EOP-15, Then entry into 1/2 EOP-15 has been met.	**1.** RCS Pressure Control Safety function **NOT** met **AND** entry into procedure 1/2 EOP-15 **OR** **2.** RCS Heat Removal Safety function **NOT** met and entry into procedure 1/2 EOP-15	Not Applicable **GUIDANCE BOX FOR SAFETY FUNCTION STATUS IN ALL THREE BARRIERS** If safety function cannot be restored within 15 minutes, then that safety function is NOT met for purposes of classification.	**1.** Containment Temperature and Pressure Safety function **NOT** met **AND** entry into procedure 1/2 EOP-15
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
1. Coolant Activity greater than 300 uCi/gm Dose Equivalent I-131 (as determined by procedure CY-SL-108-0004, Guidelines for Collecting Post Accident Samples) **GUIDANCE BOX** See also SU4, Fuel Clad Degradation.	Not Applicable	**1.** RCS leak rate greater than available makeup capacity as indicated by a loss of RCS minimum subcooling **GUIDANCE BOX** • **MINIMUM SUBCOOLING** Determination is made using Figure 1A / 1B in 1/2-EOP-99. • See also SU5, RCS Leakage.	**1.** RCS leak rate indicated greater than 50 gpm with Letdown isolated **GUIDANCE BOX** Isolation of Letdown is to distinguish between RCS leakage and CVCS leakage and is performed when procedurally required.	**1.** A containment pressure rise followed by a rapid unexplained drop in containment pressure. **OR** **2.** Containment pressure **OR** sump level response **NOT** consistent with LOCA conditions	**1.** Containment pressure greater than 44- PSIG and rising **OR** **2.** Containment Hydrogen greater than 4% **OR** **3.** Pressure greater than 10 [5.4] psig **AND** **a.** Less than one FULL TRAIN OF DEPRESSURIZATION EQUIPMENT operating
OR		**OR**		**OR**	
3. Core Exit Thermocouple Readings		**3. Not Applicable**		**3. Core Exit Thermocouple Reading**	
1. Core Exit Thermocouples reading greater than 1200°F **GUIDANCE BOX** At least two (2) Core Exit Thermocouples must exceed the threshold.	**1.** Core Exit Thermocouples reading greater than 700° F **GUIDANCE BOX** At least two (2) Core Exit Thermocouples must exceed the threshold.	Not applicable	Not applicable	**GUIDANCE BOX** • At least two (2) Core Exit Thermocouples must exceed the threshold. • Sensors 4 through 8 **NOT** covered means sensors 4 through 8 inclusive (all). Not applicable	**1.** Core Exit Thermocouples reading greater than 1200°F **AND** **a.** Functional Recovery (1/2 EOP-15) for RCS and Core Heat Removal **NOT** effective within 15 minutes **OR** **2.** Core Exit Thermocouples reading greater than 700° F **AND** **BOTH** of the following apply: • RVLMS indicates Sensors 4 through 8 **NOT** covered **OR** T_{HOT} **AND** REP CET difference greater than 20° F (LOCA **NOT** in progress) **OR** Greater than 22° F superheated on REP CET (LOCA in progress) **AND** • Functional Recovery (1/2 EOP-15) for RCS and Core Heat Removal **NOT** effective within 15 minutes
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**4. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
GUIDANCE BOX Sensors 4 through 8 **NOT** covered means sensors 4 through 8 inclusive (all). Not Applicable	**1.** RVLMS indicates Sensors 4 through 8 **NOT** covered **OR** **2.** T_{HOT} **AND** REP CET difference greater than 20°F (LOCA **NOT** in progress) **OR** **3.** Greater than 22° F superheated on REP CET (LOCA in progress)	**1.** RUPTURED S/G results in a Safety Injection Actuation Signal (SIAS)	Not Applicable	**1.** RUPTURED S/G is also FAULTED outside of containment **OR** **2.** Primary-to-Secondary leakrate greater than 10 gpm **AND** **a.** UNISOLABLE steam release from affected S/G to the environment	Not applicable
OR		**OR**		**OR**	
5. Not Applicable		**5. Not Applicable**		**5. CNTMT Isolation Failure or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	**1.** Failure of all valves in **ANY** one line to close **AND** **a.** Direct downstream pathway to the environment exists after CIS/CIAS	Not Applicable
OR		**OR**		**OR**	
6. Containment Radiation Monitoring		**6. Containment Radiation Monitoring**		**6. Containment Radiation Monitoring**	
1. CHRRM reading greater than 1.4 E+02 R/hr	Not Applicable	**1.** **ANY** CIS monitor reading greater than 1.5 E+03 mR/hr	Not Applicable	Not Applicable	**1.** CHRRM reading greater than 2.7 E+03 R/hr
OR		**OR**		**OR**	
7. Emergency Coordinator Judgment		**7. Emergency Coordinator Judgment**		**7. Emergency Coordinator Judgment**	
1. ANY condition in the opinion of the Emergency Coordinator that indicates Loss of the Fuel Clad Barrier	**1. ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the Fuel Clad Barrier	**1. ANY** condition in the opinion of the Emergency Coordinator that indicate Loss of the RCS Barrier	**1. ANY** condition in the opinion of the Emergency Coordinator that indicate Potential Loss of the RCS Barrier	**1. ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the Containment Barrier	**1. ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the Containment Barrier

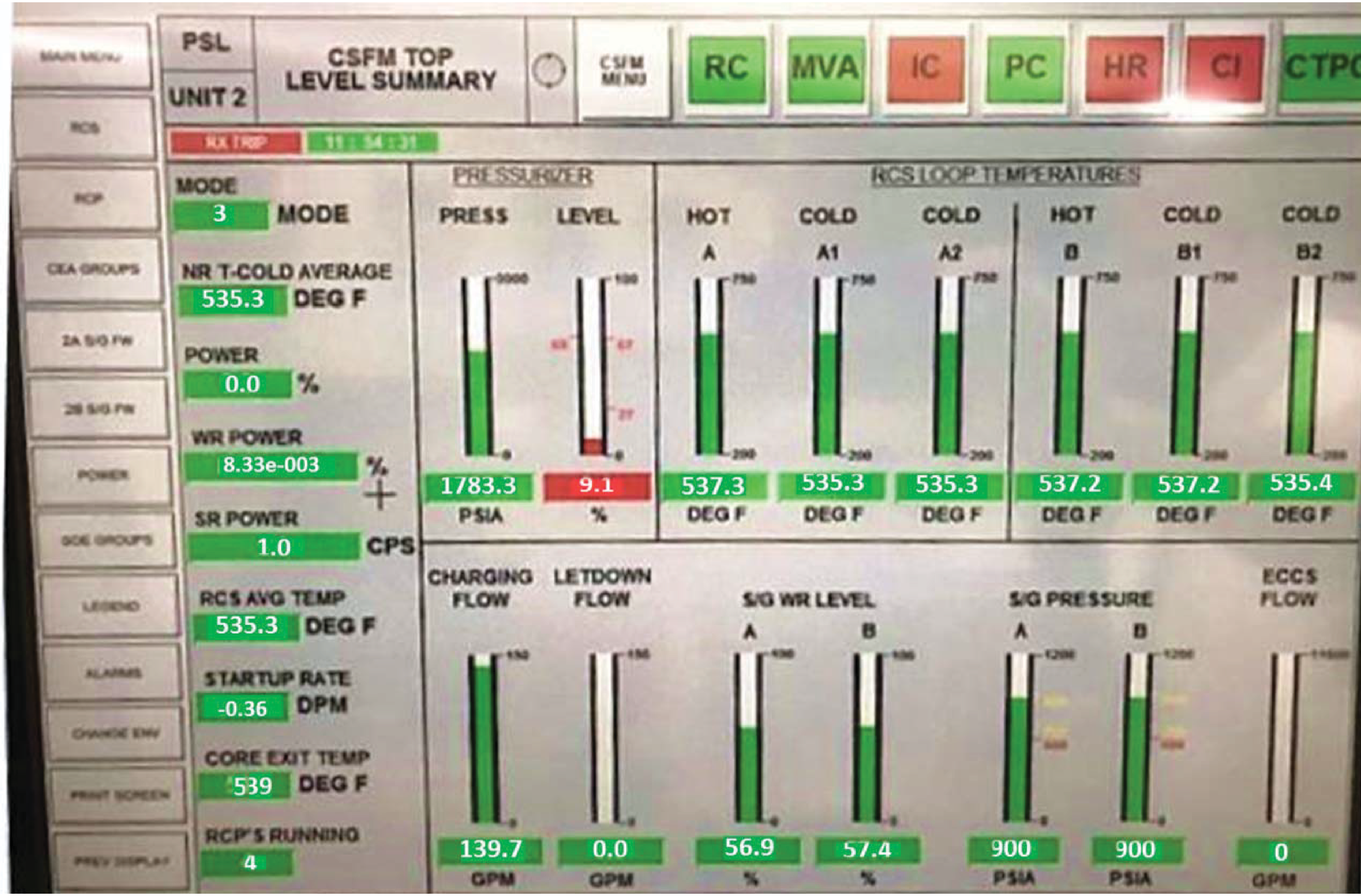
Determine Emergency Classification based on Barrier Status

ANY Loss or **ANY** Potential Loss of **EITHER** Fuel Clad **OR** RCS.
ALERT: FA1

ANY Loss or **ANY** Potential Loss of Containment
UNUSUAL EVENT: FU1

Loss or Potential Loss of **ANY** Two Barriers.
SITE AREA EMERGENCY: FS1

Loss of **ANY** Two Barriers **AND** Loss or Potential Loss of the Third Barrier.
GENERAL EMERGENCY: FG1


PSL
UNIT 2
CSFM TOP
LEVEL SUMMARY
CSFM MENU
RC
MVA
IC
PC
HR
CI
CTPC
MODE
3
MODE
NR T-COLD AVERAGE
535.3
DEG F
POWER
0.0
%
WR POWER
8.33e-003
%
SR POWER
1.0
CPS
RCS AVG TEMP
535.3
DEG F
STARTUP RATE
-0.36
DPM
CORE EXIT TEMP
539
DEG F
RCP'S RUNNING
4
PRESSURIZER
PRESS
LEVEL
1783.3
PSIA
9.1
%
RCS LOOP TEMPERATURES
HOT
A
COLD
A1
COLD
A2
HOT
B
COLD
B1
COLD
B2
537.3
DEG F
535.3
DEG F
535.3
DEG F
537.2
DEG F
537.2
DEG F
535.4
DEG F
CHARGING
FLOW
LETDOWN
FLOW
S/G WR LEVEL
A
B
S/G PRESSURE
A
B
ECCS
FLOW
139.7
GPM
0.0
GPM
56.9
%
57.4
%
900
PSIA
900
PSIA
0
GPM


PSL
UNIT 2
CONTAINMENT ISOLATION SECONDARY RAD TRENDS
11:56 AM
RELATED DISPLAYS
RC
MVA
IC
PC
HR
CI
CTPC
RX TRIP
11 : 54 : 31
00:00:15:00 35
Pause
HIGHEST S/G BLOWDOWN RAD
2.17e-007
UCI/CC
0.0000
CONDENSER AIR EJECTOR RAD
1.39e-002
UCI/CC
0.0139
MAIN STEAM LINE A RAD
0.3
MR/HR
0.2940
MAIN STEAM LINE B RAD
0.4
MR/HR
0.4125
Operator

#82S REFERENCE

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum water level of greater than or equal to 27% indicated level and a maximum water level of less than or equal to 68% indicated level and at least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of the above required pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when second group of required pressurizer heaters intentionally made inoperable.

b. With two groups of required pressurizer heaters inoperable, restore at least one group of required pressurizer heaters to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW in accordance with the Surveillance Frequency Control Program.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality and Ventilation System Isolation Monitor	4	*	$\leq$ 20 mR/hr	$10^{-1} - 10^{4}$ mR/hr	22
b. Containment Isolation	3	****	$\leq$ 90 mR/hr	$1 - 10^{7}$ mR/hr	25
c. Containment Area – Hi Range	1	1, 2, 3 & 4	Not Applicable	$1 - 10^{7}$ R/hr	27
d. Control Room Isolation	1 per intake	ALL MODES	$\leq$ 320 cpm	$10^{-7} - 10^{-2}$ μCi/cc	26
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10^{-7} - 10^{-2}$ μCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{7}$ cpm	23

* With fuel in the storage pool or building.

**** During movement of recently irradiated fuel assemblies within containment.

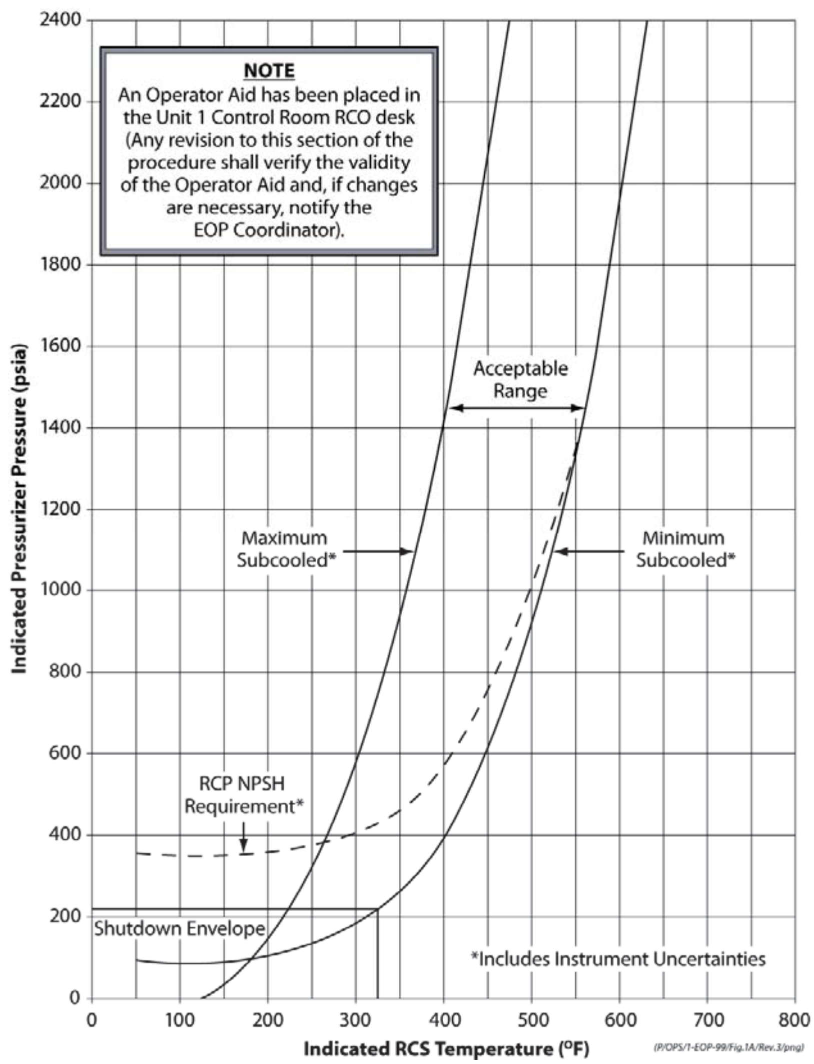
#85S REFERENCE

FIGURE 1A
RCS PRESSURE TEMPERATURE
(Page 1 of 1)

[redacted]

CAUTION

The RCP NPSH curve assumes one pump is operating in each loop. RCP instrumentation should be monitored for seal and pump performance in accordance with 1-EOP-99, Table 13.



RCS Pressure Range	Required QSPDS Subcooled Margin Reading (Rep CET)
2250 psia to 1000 psia	40 to 180°F
1000 psia to 500 psia	50 to 170°F
Less than 500 psia	80 to 160°F

#85S REFERENCE

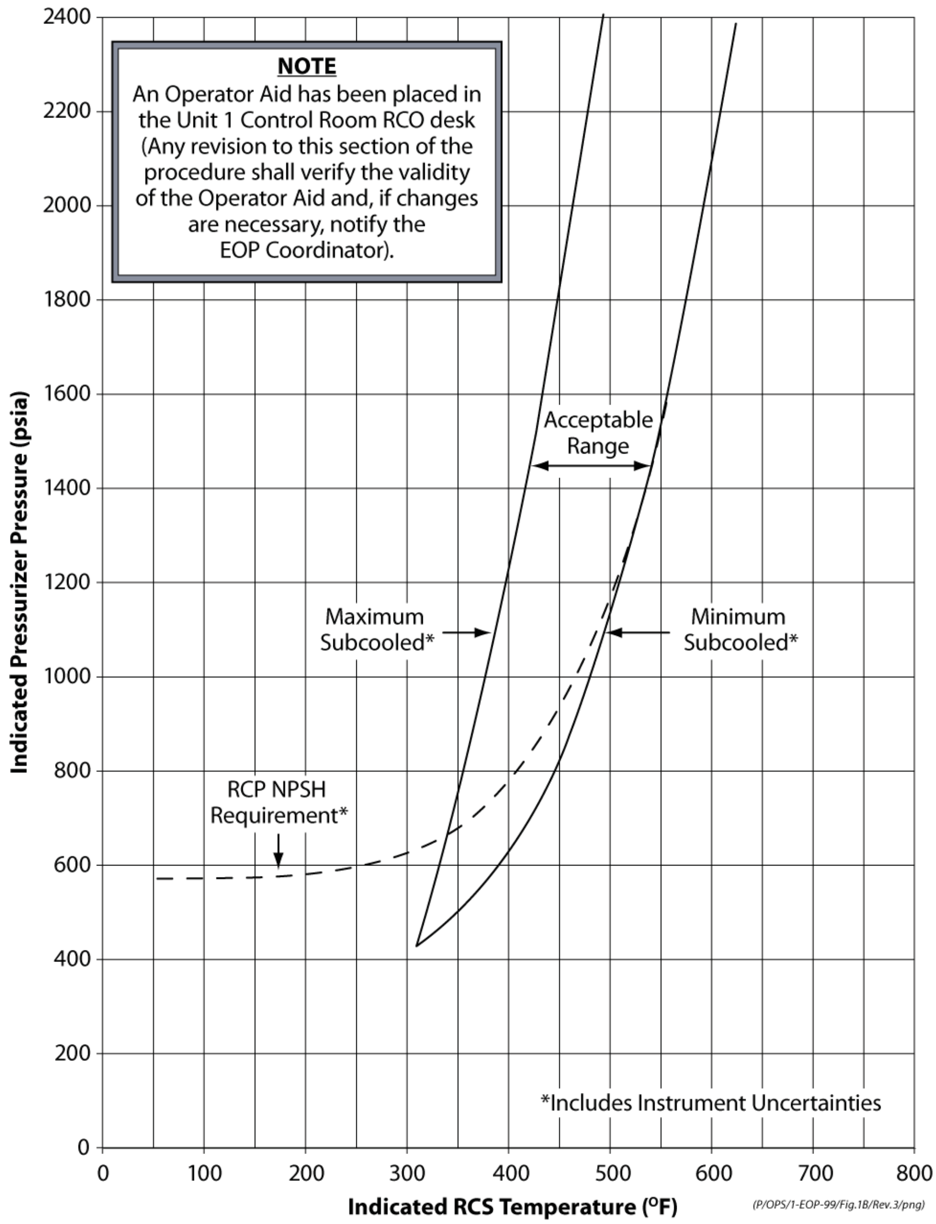
FIGURE 1B

<u>RCS PRESSURE TEMPERATURE</u>

(Page 1 of 1)

[redacted]

> **<u>CAUTION</u>**
>
> The RCP NPSH curve assumes one pump is operating in each loop. RCP instrumentation should be monitored for seal and pump performance in accordance with 1-EOP-99, Table 13.

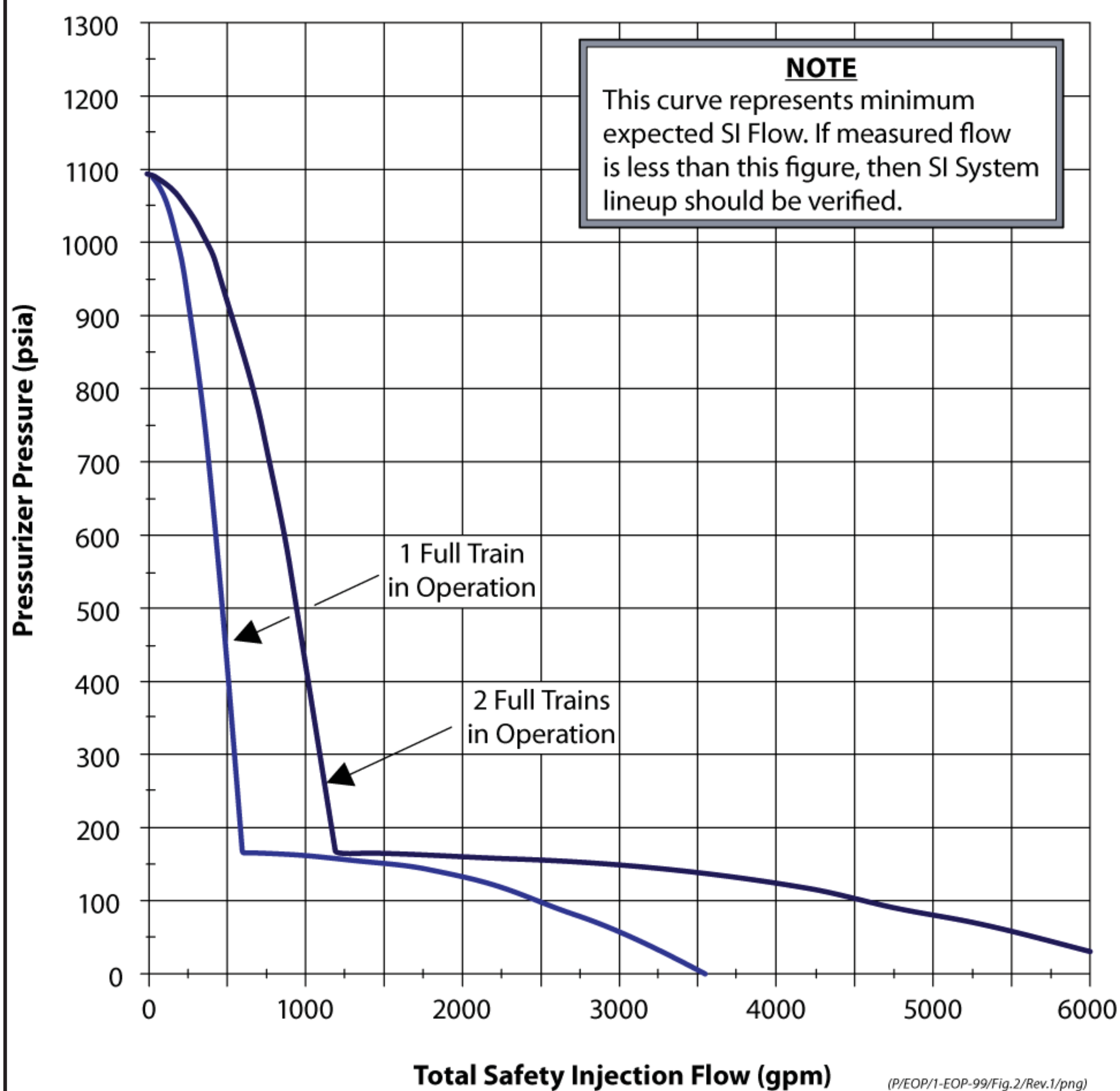


#85S REFERENCE

FIGURE 2

SAFETY INJECTION FLOW VS. RCS PRESSURE

(Page 1 of 1)



ATTACHMENT 1
REPORTABLE EVENTS
(Page 1 of 8)

Declaration of an Emergency Class (See NUREG-1022 Section 3.1.1)	
1 Hour Report § 50.72(a)(1)(i) "The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan."	
Plant Shutdown Required by Technical Specifications (See NUREG-1022 Section 3.2.1)	
4 Hour Report § 50.72(b)(2)(i) "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	**60 Day LER § 50.73(a)(2)(i)(A)** "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."
Operation or Condition Prohibited by Technical Specifications (See NUREG-1022 Section 3.2.2)	
	60 Day LER § 50.73(a)(2)(i)(B) "Any operation or condition which was prohibited by the plant's Technical Specifications except when: (1) The Technical Specification is administrative in nature; (2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event."

ATTACHMENT 1
REPORTABLE EVENTS
(Page 2 of 8)

Deviation from Technical Specifications Authorized under § 50.54(x) (See NUREG-1022 Section 3.2.3)	
1 Hour Report§ 50.72(b)(1) "... any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part."	**60 Day LER § 50.73(a)(2)(i)(C)** "Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part."
Degraded or Unanalyzed Condition (See NUREG-1022 Section 3.2.4)	
8 Hour Report § 50.72(b)(3)(ii) "Any event or condition that results in:	**60 Day LER 50.73(a)(2)(ii)** "Any event or condition that resulted in:
(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety."	(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety."
External Threat or Hampering (See NUREG-1022 Section 3.2.5)	
	60 Day LER § 50.73(a)(2)(iii) "Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant."

ATTACHMENT 1
REPORTABLE EVENTS
(Page 3 of 8)

System Actuation (See NUREG-1022 Section 3.2.6)	
4 Hour Report § 50.72(b)(2)(iv)(A) “Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.” **4 Hour Report § 50.72(b)(2)(iv)(B)** “Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.” **8 Hour Report § 50.72(b)(3)(iv)(A)** “Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) of this section, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.”	**60 Day LER § 50.73(a)(2)(iv)(A)** “Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, except when: (1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or (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. As indicated in 10 CFR 50.73(a)(1), in the case of an invalid actuation reported under 10 CFR 50.73(a)(2)(iv)(A) other than actuation of the RPS when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER.

ATTACHMENT 1
REPORTABLE EVENTS
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8 Hour Report § 50.72(b)(3)(iv)(B) "The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are:	**§ 50.73(a)(2)(iv)(B)** "The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are:
(1) Reactor protection system (RPS) including: reactor scram and reactor trip. [5]	(1) Reactor protection system (RPS) including: reactor scram or reactor trip.
(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).	(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).
(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.	(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.
(4) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.	(4) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.
(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.	(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.
(6) PWR auxiliary or emergency feedwater system.	(6) PWR auxiliary or emergency feedwater system.
(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems	(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.
(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.	(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.
[5] Actuation of the RPS when the reactor is critical is reportable under § 50.72(b)(2)(iv)(B)	(9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

ATTACHMENT 1
REPORTABLE EVENTS
(Page 5 of 8)

Event or Condition that Could Have Prevented Fulfillment of a Safety Function (See NUREG-1022 Section 3.2.7)	
8 Hour Report § 50.72(b)(3)(v) "Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident." **8 Hour Report § 50.72(b)(3)(vi)** "Events covered in paragraph (b)(3)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (b)(3)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function."	**60 Day LER § 50.73(a)(2)(v)** "Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident." **§ 50.73(a)(2)(vi)** "Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function."
Common Cause Inoperability of Independent Trains or Channels (See NUREG-1022 Section 3.2.8)	
	60 Day LER § 50.73(a)(2)(vii) "Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident."

ATTACHMENT 1
REPORTABLE EVENTS
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Radioactive Release (See NUREG-1022 Section 3.2.9)	
	60 Day LER § 50.73(a)(2)(viii)(A) "Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in appendix B to part 20, table 2, column 1." **60 Day LER § 50.73(a)(2)(viii)(B)** "Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases."
Internal Threat or Hampering (See NUREG-1022 Section 3.2.10)	
	60 Day LER § 50.73(a)(2)(x) "Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases."
Transport of a Contaminated Person Offsite (See NUREG-1022 Section 3.2.11)	
8 Hour Report § 50.72(b)(3)(xii) "Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment."	

ATTACHMENT 1
REPORTABLE EVENTS
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News Release or Notification of Other Government Agency **(See NUREG-1022 Section 3.2.12)**	
4 Hour Report § 50.72(b)(2)(xi) "Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials."	
Loss of Emergency Preparedness Capabilities **(See NUREG-1022 Section 3.2.13)**	
8 Hour Report § 50.72(b)(3)(xiii) "Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system)."	

ATTACHMENT 1
REPORTABLE EVENTS
(Page 8 of 8)

Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems (See NUREG-1022 Section 3.2.14)	
	60 Day LER § 50.73(a)(2)(ix)(A) "Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to: (1) Shut down the reactor and maintain it in a safe shutdown condition; (2) Remove residual heat; (3) Control the release of radioactive material; or (4) Mitigate the consequences of an accident." **§ 50.73(a)(2)(ix)(B)** "Events covered in paragraph (ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (ix)(A) of this section if the event results from: (1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or (2) Normal and expected wear or degradation."

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump,

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

d. One OPERABLE charging pump*.

APPLICABILITY: MODES 1, 2 and 3**.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

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* One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

** With pressurizer pressure $\geq$ 1750 psia.

S – SYSTEM MALFUNCTIONS

Recognition Category | General Emergency | Site Area Emergency | Alert | Unusual Event

AC POWER

General Emergency

SG1 Prolonged Loss of All Off-site and All On-Site AC Power to Emergency Busses.

(GB)

Operating Mode Applicability: **1, 2, 3, 4**

EAL Values:

1. Loss of all Off-site **AND** all On-site AC power to A3 4.16 KV **AND** B3 4.16 KV busses.

AND

a. EITHER of the following:

(1) Restoration of at least one emergency bus in less than 4 hours is **NOT** likely

OR

(2) RCS and Core Heat Removal Safety function is **NOT** met.

Site Area Emergency

SS1 Loss of All Off-site and All On-Site AC Power to Emergency Busses for 15 minutes or longer.

(GB)

Operating Mode Applicability: **1, 2, 3, 4**

EAL Values:

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

1. Loss of all Off-site **AND** all On-site AC Power to A3 4.16 KV **AND** B3 4.16 KV busses for 15 minutes or longer

Alert

SA5 AC Power Capability To Emergency Busses Reduced To A Single Power Source For 15 Minutes or Longer Such That Any Additional Single Failure Would Result In Station Blackout.

Operating Mode Applicability: **1, 2, 3, 4**

EAL Value:

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

1. AC power capability to A3 4.16 KV **AND** B3 4.16 KV busses reduced to a single power source for 15 minutes or longer.

AND

a. ANY additional single power source failure will result in a Station Blackout.

Unusual Event

SU1 Loss of All Off-site AC Power to Emergency Busses for 15 Minutes or Longer.

Operating Mode Applicability: **1, 2, 3, 4**

EAL Values:

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

1. Loss of all Off-site AC power to A3 4.16 KV **AND** B3 4.16 KV for 15 minutes or longer.

FAILURE OF RX PROTECTION / CRITICALITY

General Emergency

SG2 Automatic Trip and All Manual Actions Fail to Shutdown the Reactor **AND** Indication of an Extreme Challenge to the Ability to Cool the Core Exists.

Operating Mode Applicability: **1, 2**

EAL Values:

1. Automatic trip failed to shutdown the reactor.

AND

a. **ALL** Manual actions failed to shutdown the reactor as indicated by:

- Reactor power is **NOT** dropping to less than 5% power
- All CEAs are **NOT** inserted

AND

b. **EITHER** of the following exist or have occurred due to continued power generation:

(1) Core Heat Removal Safety Function **NOT** met.

OR

(2) RCS Heat Removal Safety Function **NOT** met.

Site Area Emergency

SS2 Automatic Trip Fails to Shutdown the Reactor **AND** Manual Actions Taken from the Reactor Turbine Generator Board (RTGB) are **NOT** Successful in Shutting Down the Reactor.

Operating Mode Applicability: **1, 2**

EAL Values:

1. An automatic trip failed to shutdown the reactor

AND

a. Manual actions taken at the Reactor Turbine Generator Board (RTGB) **DO NOT** shutdown the reactor as indicated by:

- Reactor power is **NOT** dropping to less than 5% power
- **ALL** full strength CEAs are **NOT** inserted

Alert

SA2 Automatic Trip Fails to Shutdown the Reactor **AND** the Manual Actions Taken from the Reactor Turbine Generator Board (RTGB) are Successful in Shutting Down the Reactor

Operating Mode Applicability: **1, 2**

EAL Values:

1. An Automatic trip failed to shutdown the reactor

AND

a. Manual actions taken at the Reactor Turbine Generator Board (RTGB) successfully shutdown the reactor as indicated by **ALL** of the following:

- Reactor power is dropping to less than 5% power
- Negative start-up rate
- **All** CEAs are inserted or boration in progress

Unusual Event

SU8 Inadvertent Criticality.

Operating Mode Applicability: **3, 4**

EAL Values:

1. UNPLANNED sustained positive start-up rate observed on nuclear instrumentation.

DC POWER

General Emergency

GUIDANCE BOX FOR SG1, SS1

On-site AC power may be provided by the other Unit's Emergency Diesel Generator (EDG) by successful X-tie to either the A3 or B3 4.16 KV bus.

Site Area Emergency

SS3 Loss of All Vital DC Power for 15 Minutes or Longer.

Operating Mode Applicability: **1, 2, 3, 4**

EAL Values:

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

1. Less than 112 VDC on 1[2]A, 1[2]B **AND** 1[2]AB Vital DC busses for 15 minutes or longer.

Unusual Event

DEFINITION BOX

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

S – SYSTEM MALFUNCTIONS

Columns: General Emergency | Site Area Emergency | Alert | Unusual Event | Recognition Category

ANNUNCIATORS

General Emergency

(none)

Site Area Emergency

SS6 Inability to Monitor a Significant Transient in Progress.

(GB)

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

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

1. Loss of greater than approximately 75% of the following for 15 minutes or longer per 1[2]-AOP-100.03:

a. Control Room Safety System annunciation.

OR

b. Control Room Safety System indication associated with the above annunciators.

AND

BOTH of the following apply:

- **ANY** of the following:
  - Electrical load rejection greater than 25% full electrical load
  - Reactor Trip
  - Safety Injection Actuation

AND

- Distributed Control System (DCS) **AND** Qualified Safety Parameter Display System (QSPDS) are unavailable.

Alert

SA4 UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Indicators Unavailable.

(GB)

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

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

1. UNPLANNED loss of greater than approximately 75% of the following for 15 minutes or longer per 1[2]-AOP-100.03:

a. Control Room Safety System annunciation.

OR

b. Control Room Safety System indication associated with the above annunciators.

AND

EITHER of the following apply:

- **ANY** of the following:
  - Electrical load rejection greater than 25% full electrical load
  - Reactor Trip
  - Safety Injection Actuation

OR

- Distributed Control System (DCS) **AND** Qualified Safety Parameter Display System (QSPDS) are unavailable.

Unusual Event

SU3 UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room for 15 Minutes or Longer

(GB)

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

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

1. UNPLANNED loss of greater than approximately 75% of the following for 15 minutes or longer per 1[2]-AOP-100.03:

a. Control Room Safety System annunciation.

OR

b. Control Room Safety System indication associated with the above annunciators.

RCS LEAKAGE

GUIDANCE BOX FOR SS6, SA4, SU3

Safety System indication can not be lost without concurrent loss of Safety System annunciation.

GUIDANCE BOX FOR SU5

See also 2. RCS Leak Rate in the Fission Product Barrier (FPB) Table.

Unusual Event

SU5 RCS Leakage.

(GB)

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

1. Unidentified **OR** pressure boundary leakage greater than 10 gpm.

OR

2. Identified leakage greater than 25 gpm.

FUEL CLAD

DEFINITION BOX

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

RCS LEAK RATE – Comprised of IDENTIFIED and UNIDENTIFIED LEAKAGE as defined by Technical Specifications.

UNIDENTIFIED LEAKAGE – Leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

CONTROLLED LEAKAGE – Seal water flow supplied from the reactor coolant pump seals.

IDENTIFIED LEAKAGE:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system (Primary-to-secondary leakage).

GUIDANCE BOX FOR SU4

See also 2. Primary Coolant Activity in the Fission Product Barrier (FPB) Table.

Unusual Event

SU4 Fuel Clad Degradation.

(GB)

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

1. Reactor Coolant sample activity value indicating fuel clad degradation greater than:

a. 60.0 uCi/gm Dose Equivalent I-131

OR

b. Specific activity greater than 518.9 uCi/gm Dose Equivalent Xe-133

TECH SPECS

Unusual Event

SU2 Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability: 1, 2, 3, 4

EAL Values:

1. Plant is **NOT** brought to required operating mode within Technical Specifications LCO Action Statement Time.

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category
		GUIDANCE FOR SU6 * Commercial phones include installed cell phones in the Control Room, but not personal cell phones.	**SU6** Loss of All On-site or Off-site Communications Capabilities. (GB) **Operating Mode Applicability:** **1, 2, 3, 4** **EAL Values:** **1.** Loss of **ALL** of the following on-site communication methods affecting the ability to perform routine operations: • Plant Page • Plant Radio • Commercial Phones* **OR** **2.** Loss of **ALL** of the following off-site communication methods in **EITHER** box: **State and County Notifications** • Hot Ringdown (HRD) • Commercial phone* • EMnet **OR** **NRC Notifications** • Emergency Notification System (ENS) • Commercial phone*	COMMUNICATIONS S – SYSTEM MALFUNCTIONS