NRC Exam

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: June 28th, 2013	Facility/Unit: **Browns Ferry**
Region: I **II** III IV	Reactor Type: W CE BW **GE**
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

rec'd 6/24/13

ILT 1306 NRC Written Exam Answer Key

Question #	K/A	Answer
1	295001 AK1.03	A
2	295003 AA2.02	D
3	295004 AA1.01	B
4	295005 AA1.01	D
5	295006 AK1.02	A
6	295016 G2.1.32	C
7	295018 AA1.02	D
8	295019 G2.2.39	A
9	295021 AA2.07	D
10	295023 AK3.04	D
11	295024 G2.4.8	C
12	295025 EA2.01	C
13	295026 EA1.01	C
14	295028 EK3.04	B
15	295030 EK2.01	A
16	295031 EK3.03	C
17	295037 EK1.03	C
18	295038 EK2.02	D
19	600000 AK2.01	B
20	700000 AK1.01	C
21	295008 AK1.02	B
22	295014 AA2.03	D
23	295015 AK3.01	C
24	295020 G2.4.4	C
25	295032 EK2.08	B
26	295033 EA1.05	C
27	295036 EA1.01	D
28	203000 A2.16	B
29	205000 A4.07	D
30	206000 G2.2.38	A
31	209001 A3.03	B
32	211000 K1.01	B
33	211000 K3.01	A
34	212000 K4.01	C
35	215003 A3.03	C
36	215004 K2.01	C
37	215005 K2.02	B
38	217000 A1.02	D
39	217000 K5.06	A
40	218000 K4.01	D
41	223002 A2.06	A
42	223002 K4.04	B
43	239002 K6.02	B
44	259002 K5.03	D
45	261000 A1.07	D
46	262001 K2.01	B
47	262001 K5.02	B
48	262002 K3.17	A
49	263000 A1.01	A
50	263000 K6.01	C
51	264000 K3.03	D
52	300000 G2.4.45	D
53	400000 K1.01	C
54	201002 K4.04	B
55	201006 A2.05	A
56	214000 K6.01	B
57	216000 A4.02	A
58	230000 A1.10	C
59	234000 K5.02	A
60	259001 G2.4.6	D
61	271000 K5.11	C
62	272000 K2.03	B
63	288000 K3.05	A
64	290003 A3.01	D
65	290002 K1.14	C
66	G2.1.27	D
67	G2.1.31	D
68	G2.1.36	D
69	G2.2.15	D
70	G2.2.25	A
71	G2.2.39	A
72	G2.3.13	A
73	G2.3.14	A
74	G2.4.1	B
75	G2.4.9	A
76	295003 AA2.05	D
77	295019 AA2.02	B
78	295023 G2.2.44	B
79	295024 G2.4.21	C
80	295037 G2.4.30	D
81	600000 AA2.13	C
82	700000 AA2.05	C
83	295017 AA2.01	A
84	295029 EA2.03	B
85	500000 G2.2.44	C
86	209001 G2.1.31	D
87	211000 A2.05	D
88	215003 A2.01	C
89	218000 A2.05	C
90	239002 G2.2.12	C
91	201003 G2.4.11	A
92	239001 A2.11	B
93	268000 A2.01	D
94	G2.1.35	A
95	G2.1.43	B
96	G2.2.18	A
97	G2.2.3	C
98	G2.3.4	A
99	G2.4.18	A
100	G2.4.19	B

QUESTION 76

At 0900 on April 27th, offsite power was lost resulting in the following conditions:

- All 500 KV lines are DE-ENERGIZED
- Athens 161KV line is DE-ENERGIZED
- Trinity 161KV line is ENERGIZED
- B and D Emergency Diesel Generators failed to start and can NOT be manually started

At 1000, power has been restored to the Browns Ferry transmission yard via 500KV lines (Limestone and Union).

Which ONE of the following is the LATEST date & time Units 1 and 2 are required to be in Mode 4 in accordance with Technical Specification 3.8.1, AC Sources- Operating?

[REFERENCE PROVIDED]

A. April 27th at 2200

B. April 27th at 2300

C. April 28th at 2200

D April 28th at 2300

QUESTION 77

Given the following conditions:

- Unit 3 is operating at 40% power when a loss of drywell control air (DWCA) occurs
- DRYWELL CONTROL AIR PRESS LOW (Panel 9-3E, Window 35) is in Alarm
- MAIN STEAM RELIEF VLV ACCUM PRESS LOW (Panel 9-3D, Window 18) is in Alarm
- DWCA has been cross tied to the Containment Atmospheric Dilution (CAD) System in accordance with 3-AOI-32A-1, Loss of Drywell Control Air

Which ONE of the following completes both statements below?

In accordance with the Tech Spec Bases for SR 3.5.1.3, the minimum required pneumatic pressure for the ADS valves to remain operable is __(1)__ psig.

The CAD subsystem __(2)__ required to be declared inoperable when crosstied with the DWCA System.

A. (1) 81
(2) is

B. (1) 81
(2) is NOT

C. (1) 87
(2) is

D (1) 87
(2) is NOT

QUESTION 78

Given the following conditions:

- Unit 2 is in a Refueling Outage
- Movement of irradiated fuel is in progress
- Refueling personnel report that a grapple failure has caused a fuel bundle to drop, resulting in the following alarms/indications:
  - FUEL POOL FLOOR AREA RADIATION HIGH, (2-9-3A, Window 1)
  - 2-RM-90-1A, Fuel Pool Area Radiation Monitor is reading 1000 mrem/hr

Which one of the following completes the statements below?

Based on the above conditions, the required immediate actions in accordance with 2-AOI-79-1, Fuel Damage During Refueling, for the SRO is to direct evacuation of non essential personnel from the __(1)__.

In accordance with EPIP-1, Emergency Plan Implementing Procedure, the HIGHEST required emergency action level classification for these conditions is a (an) __(2)__.

[REFERENCE PROVIDED]

A. (1) Refuel Floor ONLY
   (2) Unusual Event

B. (1) Refuel Floor ONLY
   (2) Alert

C. (1) Drywell AND Refuel Floor
   (2) Unusual Event

D. (1) Drywell AND Refuel Floor
   (2) Alert

QUESTION 79

Which ONE of the following completes both statements below?

In accordance with EOI-2, Step PC/P-13, __(1)__ is required to be implemented to control suppression chamber pressure less than 55 psig.

In accordance with EPIP-1, Emergency Classification Technical Basis, Section 2.1-G (general Emergency: Primary containment pressure at 55 psig), at this point, __(2)__ is (are) potentially threatened due to direction in the EOIs to Spray Primary Containment.

A. (1) Appendix 12, Primary Containment Venting
   (2) fuel cladding integrity

B. (1) Appendix 12, Primary Containment Venting
   (2) offsite release limits

C. (1) Appendix 13, Emergency Venting Primary Containment
   (2) fuel cladding integrity

D. (1) Appendix 13, Emergency Venting Primary Containment
   (2) offsite release limits

QUESTION 80

Given the following conditions:

- Unit 2 was operating at 100% power
- At time 0805 a scram occurred due to a loss of both RPS Bus A and RPS Bus B
- **NO** control rods initially inserted
- Following the manual scram and ARI, several rods failed to fully insert

At 0815, the following conditions exist:

- Reactor power is UNKNOWN
- Reactor pressure is being maintained 800 to 1000 psig with two (2) SRVs OPEN and a third being manually cycled
- Reactor water level is (-)75 inches and steady, being maintained using HPCI
- Suppression pool temperature is 136°F and rising

At 0817, the Shift Manager, as the Site Emergency Director, made an Emergency Plan event declaration in accordance with EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE

Which ONE of the following completes the statement below?

At 0817, the highest required classification is __(1)__ and the State of Alabama is required to be notified no later than __(2)__.

[REFERENCE PROVIDED]

A. (1) General Emergency
(2) 0832

B. (1) General Emergency
(2) 0835

C. (1) Site Area Emergency
(2) 0835

D. (1) Site Area Emergency
(2) 0832

QUESTION 81

Given the following conditions:

- All three units are operating at 100% power
- A fire is reported in the 4KV Shutdown Board "C"
- RHR Pump 2B automatically started and then tripped with no operator action

Note:

- 1(3)-AOI-100-1, Manual Scram
- 0-SSI-9, Unit 2 Reactor Building Fire 4KV Electrical Board Room 2A

Which ONE of the following completes the statements below?

The Shift Manager has entered the SSIs __(1)__, and __(2)__.

A. (1) solely based on the location of the fire
(2) all three Units will implement 0-SSI-9

B. (1) due to RHR Pump 2B starting and tripping
(2) Units 1 and 3 will implement 1(3)-AOI-100-1, and Unit 2 will implement 0-SSI-9

C. (1) due to RHR Pump 2B starting and tripping
(2) all three Units will implement 0-SSI-9

D. (1) solely based on the location of the fire
(2) Units 1 and 3 will implement 1(3)-AOI-100-1, and Unit 2 will implement 0-SSI-9

QUESTION 82

Given the following conditions:

- Unit 1 is at 90% power
- Unit 2 is in MODE 5 performing a refueling outage
- Unit 3 is at 100% power
- Severe weather in the area causes grid instabilities

The following conditions exist on Unit 1:

- Incoming Mvars are 150 MVAR
- Grid voltage is 505 kV on the 500kV bus
- Grid voltage is 161kV on the 161kV bus
- Grid frequency is fluctuating from 59.97Hz to 60.03Hz

The grid conditions are RED for the 500kV system and YELLOW for the 161kV system.

Which ONE of the following completes the statements below?

The required action per 0-AOI-57-1E, Grid Instability, is to __(1)__.

In accordance with TRO-TO-SOP-30.128, Browns Ferry Nuclear Plant (BFN) Grid Operating Guide, the 161KV circuits __(2)__.

A. (1) RAISE reactive load to restore system voltage
(2) must be declared inoperable

B. (1) RAISE turbine load to restore system frequency
(2) remain operable

C. (1) RAISE reactive load to restore system voltage
(2) remain operable

D. (1) RAISE turbine load to restore system frequency
(2) must be declared inoperable

QUESTION 83

Given the following conditions:

- Unit 2 was operating at 90%
- An unisolable Main Steam Line break in the Steam Tunnel occurred
- A manual reactor scram was inserted

At time T=0, the following conditions exist:

- Stack Noble gas (WRGERMS) release rate is $6.1 X 10^{10}$ µCi/sec
- Actual Site Boundary Dose rate is 950 mR/hr gamma
- The projected Iodine-131 Site Boundary measurement is $3.8 X 10^{-6}$ µCi/cm^3

Which ONE of the following completes the statements below?

At time T=0, WRGERMS data __(1)__ be used to make an emergency declaration.

The highest required emergency classification for this event is __(2)__.

[REFERENCE PROVIDED]

A. (1) should NOT
(2) Site Area Emergency

B. (1) should NOT
(2) General Emergency

C. (1) should
(2) Site Area Emergency

D. (1) should
(2) General Emergency

QUESTION 84

Unit 2 was operating at 100% power when a LOCA occurred.
Current conditions are as follows:

- RPV water level is (-)135 inches and steady
- RPV pressure is 275 psig and steady
- Core Spray Loop I is injecting at 7000 gpm, and is the ONLY makeup source
- Drywell pressure is 32 psig
- Suppression Chamber Pressure is 30 psig
- Suppression Pool Water level is 19.5 feet and slowly rising

Based on these conditions, which ONE of the following procedures is required to be entered by the Unit Supervisor?

A. 2-EOI Appendix 12, Primary Containment Venting

B. EOI C-2, RPV-Emergency Depressurization

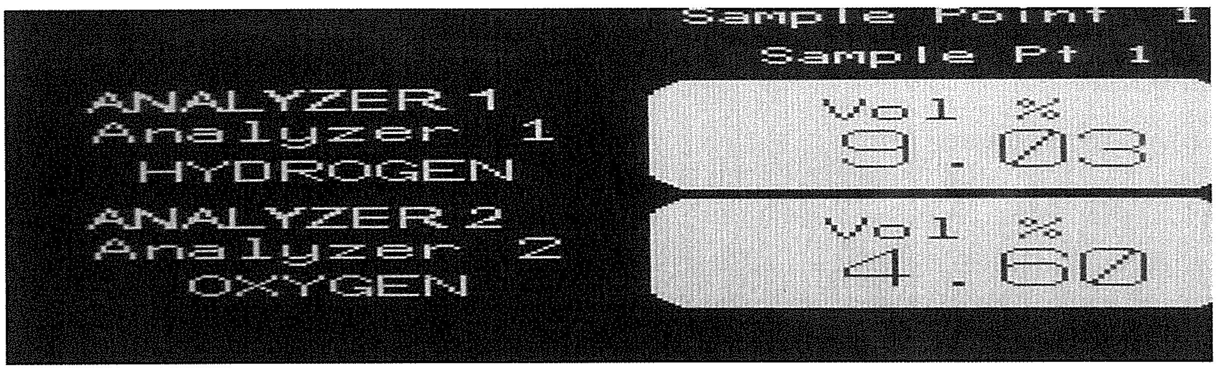
C. 2-EOI Appendix 13, Emergency Venting Primary Containment

D. EOI C-1, Alternate Level Control

QUESTION 85

Given the following conditions:

- A loss of coolant accident (LOCA) has occurred on Unit 1
- Suppression Pool level is 16 feet
- H_2/O_2 concentrations are as indicated below:



Which ONE of the following completes the statements below?

In accordance with 1-EOI-Appendix-19, H_2/O_2 Analyzer Operation, readings from 1-XR-76-110 H_2/O_2 Concentration Recorder (Panel 1-9-54) or from 1-MON-76-110, H_2/O_2 Analyzer (Panel 1-9-55) __(1)__.

Based on the current H_2/O_2 readings and in accordance with 1-EOI-2, PC/H leg, enter__(2)__.

A. (1) are valid as soon as the analyzer is placed in service
   (2) 1-EOI-Appendix 14A, N2 MAKEUP TO PRIMARY CONTAINMENT

B. (1) are valid as soon as the analyzer is placed in service
   (2) 1-EOI-Appendix 14B, CAD OPERATION

C. (1) are NOT valid until ten minutes after the analyzer is placed in service
   (2) 1-EOI Appendix 14A, N2 MAKEUP TO PRIMARY CONTAINMENT

D. (1) are NOT valid until ten minutes after the analyzer is placed in service
   (2) 1-EOI-Appendix 14B, CAD OPERATION

QUESTION 86

Given the following conditions:

- Unit 2 is operating at 100% Reactor power
- Core Spray Loop II has been inoperable since June 15 at 0800

The following Core Spray discharge pipe pressures were reported on June 18 at 0830:

- 2-PI-75-20, CS Loop I Discharge Pressure, 43 psig
- 2-PI-75-48, CS Loop II Discharge Pressure, 0 psig

Assuming no further operator action, which ONE of the following identifies the EARLIEST time that Unit 2 is required to be in Mode 3, in accordance with Tech Specs?

[REFERENCE PROVIDED]

A. June 18 at 2130

B. June 18 at 2230

C. June 22 at 0800

D. June 22 at 2000

QUESTION 87

Given the following conditions:

- At 0900 on March 26th Unit 1 scrammed from rated conditions and all rods fully inserted

At 1300:

- RPV Pressure is 500 psig
- Annunciator 9-5B window #13, SLC TEMP ABNORMAL alarms
- The AUO reports that local tank temperature is 53° F and the breakers for the heaters are tripped
- SLC storage tank boron concentration is 12.0%
- The SLC storage tank temperature continues to drop while troubleshooting the tripped breakers

Which ONE of the following completes the statements below?

At __(1)__ reactor coolant temperature must be less than 212 degrees.

In accordance with Tech Spec Bases 3.1.7, the reason why SLC is required to remain operable in Mode 3 is to ensure __(2)__.

[REFERENCE PROVIDED]

A. (1) 2100 on March 27th
(2) offsite doses remain within 10CFR50.67 limits following a LOCA

B. (1) 0900 on March 28th
(2) shutdown capability exists for the subsequent plant cooldown

C. (1) 2100 on March 27th
(2) shutdown capability exists for the subsequent plant cooldown

D. (1) 0900 on March 28th
(2) offsite doses remain within 10CFR50.67 limits following a LOCA

QUESTION 88

Given the following conditions:

- Unit 1 is in MODE 2 withdrawing control rods for a startup
- IRM D has failed and has been bypassed on Panel 9-5
- All IRM DOWNSCALE alarms have cleared on the operable IRMs
- All IRMs are on range 1
- SRMs are inserted and indicate between 9×10^3 cps and 5×10^4 cps and are trending higher

Subsequently:

- IRM H fails downscale due to a degraded power supply

Which ONE of the following completes the statements below?

The shortest required completion time for a Tech Spec or Technical Requirements Manual, LCO Required Action is __(1)__ hour(s).

Source Range Monitors (SRMs) __(2)__ be withdrawn at this time.

[REFERENCE PROVIDED]

A. (1) one
(2) can NOT

B. (1) one
(2) can

C. (1) twelve
(2) can NOT

D. (1) twelve
(2) can

QUESTION 89

Given the following conditions:

- Unit 3 is operating at 100% power
- A loss of 250VDC RMOV Board 3B occurs

Which ONE of the following identifies the required action statement(s), if any, in accordance with Technical Specification 3.5.1, ECCS - Operating?

[REFERENCE PROVIDED]

A. Action statements G.1 and G.2 are required to be entered

B. Action statement E.1 is required to be entered

C. No action statement in LCO 3.5.1 is required to be entered because LCO 3.0.6 applies

D. Action statement H.1 is required to be entered

QUESTION 90

Which ONE of the following completes the statements below in accordance with 2-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test, acceptance criteria?

Each relief valve shall be manually opened and confirmed OPEN by acoustic monitors __(1)__thermocouples.

The green CLOSED valve indicating light must be confirmed by no acoustic monitor response __(2)__ no steam flow indicated on thermocouples downstream of each relief valve.

A. (1) OR
(2) AND

B. (1) AND
(2) AND

C. (1) OR
(2) OR

D. (1) AND
(2) OR

QUESTION 91

Unit 2 has entered 2-AOI-85-4, Loss of RPIS due problems with control rod 10-27 position indication.

Which ONE of the following completes the statement below?

In accordance with TRM TR 3.3.5, Surveillance Instrumentation, the __(1)__ and __(2)__ are considered redundant to each other for the parameter of control rod motion.

A. (1) control rod position indicators
   (2) neutron monitoring instruments

B. (1) control rod position indicators
   (2) rod drift alarm feature

C. (1) rod drift alarm feature
   (2) neutron monitoring instruments

D. (1) RPIS INOPERABLE annunciator (Panel 2-9-5A Window 35)
   (2) rod drift alarm feature

QUESTION 92

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

- Main steam line flow increases rapidly
- All Main Steam Line Isolation Valves close, except MSIV LINE A INBOARD, 3-FCV-1-14A, which indicates intermediate.
- Main Steam Line Leak Detection High (Panel 9-3D-Window 24) is in alarm and TIS-1-60A reached 240° F and is slowly lowering

Which ONE of the following completes the statements below?

In accordance with the bases for Tech Spec 3.6.1.3, PCIVs, the reason why an additional 4 hours is allowed in the required completion time for Condition A (due to an inoperable MSIV) is to __(1)__.

The TSC is __(2)__ to be staffed.

[REFERENCE PROVIDED]

A. (1) restore the MSIV to Operable before having to reduce power or shutdown the unit
(2) required

B. (1) restore the MSIV to Operable before having to reduce power or shutdown the unit
(2) NOT required

C. (1) implement administrative controls for the Operable in-series valve
(2) required

D. (1) implement administrative controls for the Operable in-series valve
(2) NOT required

QUESTION 93

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

- Annunciator CONDENSATE DEMIN ABNORMAL (9-6B, W6) is received
- Hotwell Level is lowering
- Reactor vessel water conductivity is slowly rising

Further investigation reveals that DRAIN VLV (U), 3-FCV-002-0213J for the 3J Condensate Demineralizer has failed OPEN.

Which ONE of the following completes both statements below?

The failure of the Condensate Demineralizer "U" valve will cause a __(1)__.

In accordance with 3-AOI-2-1, Reactor Coolant High Conductivity, if reactor vessel water conductivity is rising towards 10 μmhos, then __(2)__ is required to be implemented.

NOTE:	3-AOI-100-1	Reactor Scram
	3-GOI-100-12A	Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations

A. (1) Backwash receiver tank high level and overflow to the backwash receiver pit sump
(2) 3-AOI-100-1

B. (1) Phase separator high level and overflow to the waste collector tank
(2) 3-GOI-100-12A

C. (1) Phase separator high level and overflow to the waste collector tank
(2) 3-AOI-100-1

D. (1) Backwash receiver tank high level and overflow to the backwash receiver pit sump
(2) 3-GOI-100-12A

QUESTION 94

In accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling, which ONE of the following identifies where the **OFFICIAL** Fuel Assembly Transfer Form (FATF) is required to be located during fuel handling?

A. The Fuel Handling Supervisor's desk

B. At the refuel floor tag board

C. On the refuel platform

D. In the Control Room on the Unit Supervisor's desk

QUESTION 95

Unit 1 is operating at 100% power when the following occurs:

- An inadvertent closure of 1-FCV-005-0005, HTR A1 EXTR ISOL VLV
- Core thermal power exceeded 3458 MWth for an 8 hour period (8 hr average)
- NO operator actions have been taken per 1-AOI-6-1A, High Pressure Feedwater Heater String/Extraction Steam Isolation

Which ONE of the following completes the statements below?

Reactor power will rise and generator output will __(1)__ slightly.

Per NPG-SPP-03.5, Regulatory Reporting Requirements, the Shift Manager must make a report to the NRC within __(2)__.

[REFERENCE PROVIDED]

A. (1) rise
(2) 4 hours

B. (1) rise
(2) 24 hours

C. (1) lower
(2) 4 hours

D. (1) lower
(2) 24 hours

QUESTION 96

Which ONE of the following completes the statements below?

Tech Spec LCO 3.0.4(b) allows entry into a mode with the LCO NOT met ONLY if __(1)__.

In accordance with NPG-SPP-09.11.2, Risk Assessment Methods for Tech Specs, the Tech Spec 3.0.4(b) provision should ONLY be used when __(2)__ once the mode is entered.

A. (1) a risk assessment is performed
   (2) there is reasonable likelihood that the inoperable equipment will be made operable within the applicable completion time

B. (1) the associated actions to be entered permit continued operation in the higher mode for an unlimited period of time
   (2) the risk management actions do not prevent the completion of other Tech Spec required actions

C. (1) a risk assessment is performed
   (2) the risk management actions do not prevent the completion of other Tech Spec required actions

D. (1) the associated actions to be entered permit continued operation in the higher mode for an unlimited period of time
   (2) there is reasonable likelihood that the inoperable equipment will be made operable within the applicable completion time

QUESTION 97

Which ONE of the following completes the statements below?

Consider each statement separately.

An Emergency Paging System (EPS) touch screen CRT __(1)__ located in the Unit 3 Control Room.

In accordance with the Technical Bases for EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, the reason the threshold value for the General Emergency classification on drywell radiation is different for the 2-RE-90-272A rad monitor on Unit 2 (when compared to the other units) is because of __(2)__.

TABLE 2.3-G1 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	90091	2-RE-90-272A	68405	3-RE-90-272A	90091

A. (1) is
   (2) fuel loading pattern

B. (1) is NOT
   (2) fuel loading pattern

C. (1) is NOT
   (2) detector geometry and relative shielding

D. (1) is
   (2) detector geometry and relative shielding

QUESTION 98

Which ONE of the following completes the statements below in accordance with EPIP-15, Emergency Exposures?

The __(1)__ shall provide authorization for all emergency radiation doses that may exceed 10 CFR 20.1201 entitled “Occupational Dose Limits for Adults”.

Potassium Iodide (KI) should be issued if a projected dose to the thyroid is expected to exceed a MINIMUM of __(2)__ REM during emergency conditions.

A. (1) Site Emergency Director
   (2) 10

B. (1) Radiation Protection Manager
   (2) 10

C. (1) Site Emergency Director
   (2) 25

D. (1) Radiation Protection Manager
   (2) 25

QUESTION 99

Which ONE of the following completes the both statements regarding, C-5, Level/Power Control, below?

In accordance with 0-EOIPM SECTION 0-V-K, CONTIGENCY #5, LEVEL/POWER CONTROL BASES, the lower limit of the RPV water level control band is Minimum Steam Cooling RPV Water Level (MSCRWL), WHICH is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding __(1)__.

In C-5, Level/Power Control, if RPV level cannot be restored and maintained above __(2)__, then SAMG entry is required.

A. (1) 1500
   (2) Minimum Steam Cooling RPV Water Level

B. (1) 1800
   (2) Minimum Zero-Injection RPV Water Level

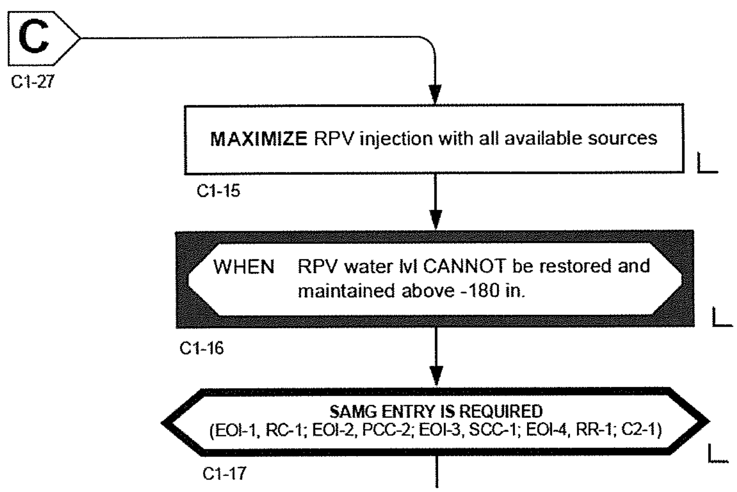
C. (1) 1800
   (2) Minimum Steam Cooling RPV Water Level

D. (1) 1500
   (2) Minimum Zero-Injection RPV Water Level

QUESTION 100

During the implementation of EOI's, the Unit Supervisor reaches the step C1-17 shown below.

For step C1-17, which ONE of the following identifies 1) the generic name of this EOI symbol **AND** 2) the point in C-1 at which the exit arrow "C" originated?



A. Action Step Symbol; in the Primary Containment Flooding portion of C-1

B. Signal Step Symbol; in the Primary Containment Flooding portion of C-1

C. Action Step Symbol; in the Steam Cooling portion of C-1

D. Signal Step Symbol; in the Steam Cooling portion of C-1

SRO Reference Table of Contents

13. EOI Curve 2 NPSH Limits (RO)

29. 2-SR-3.4.9.1 (1) Table 2 (RO)

76. Unit 2 Tech Spec 3.8.1

78. EPIP-1

80. EPIP-1

83. EPIP-1

86. Unit 2 TS 3.5.1

87. Unit 1 Tech Spec 3.1.7

88. Unit 1 TS 3.3.1.1 pages 3.3-1 through 3.3-6 only, TRM 3.3.4 pages 30-36 only

89. Unit 3 Tech Spec 3.5.1

92. Unit 3 Tech Spec 3.6.1.3, EPIP-1

95. NPG-SPP-03.5, Regulatory Reporting Requirements, Appendix A

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

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

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

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

(continued)

ACTIONS

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

(continued)

ACTIONS

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

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

(continued)

6.0 EVENT CLASSIFICATION INDEX

SECTION 1.0	REACTOR	1.1 WATER LEVEL 1.2 SCRAM FAILURE 1.3 REACTOR COOLANT ACTIVITY 1.4 MSL/OFFGAS RADIATION 1.5 LOSS OF DECAY HEAT REMOVAL
SECTION 2.0	PRIMARY CONTAINMENT	2.1 PRIMARY CONTAINMENT PRESSURE 2.2 PRIMARY CONTAINMENT HYDROGEN 2.3 DRYWELL RADIATION 2.4 DRYWELL INTERNAL LEAKAGE 2.5 LOSS OF PRIMARY CONTAINMENT
SECTION 3.0	SECONDARY CONTAINMENT	3.1 SECONDARY CONTAINMENT TEMPERATURE 3.2 SECONDARY CONTAINMENT RADIATION
SECTION 4.0	RADIOACTIVITY RELEASES	4.1 GASEOUS EFFLUENT 4.2 MAIN STEAM LINE BREAK 4.3 LIQUID EFFLUENT
SECTION 5.0	LOSS OF POWER	5.1 LOSS OF AC POWER 5.2 LOSS OF 250V DC POWER
SECTION 6.0	HAZARDS	6.1 RADIOLOGICAL 6.2 CONTROL ROOM EVACUATION 6.3 TURBINE FAILURE 6.4 FIRE/EXPLOSION 6.5 TOXIC GASES 6.6 FLAMMABLE GASES 6.7 SECURITY 6.8 VEHICLE CRASH 6.9 SPENT FUEL STORAGE
SECTION 7.0	NATURAL EVENTS	7.1 EARTHQUAKE 7.2 TORNADO/HIGH WINDS 7.3 FLOOD
SECTION 8.0	EMERGENCY DIRECTOR JUDGMENT	8.1 TECHNICAL SPECIFICATIONS 8.2 LOSS OF COMMUNICATION 8.3 LOSS OF ASSESSMENT CAPABILITY 8.4 OTHER

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REACTOR
1.0

NOTES

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

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

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

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

CURVES/TABLES:

TABLE 1.1 - G2
MINIMUM ALTERNATE RPV FLOODING PRESS (MARFP)

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

WATER LEVEL

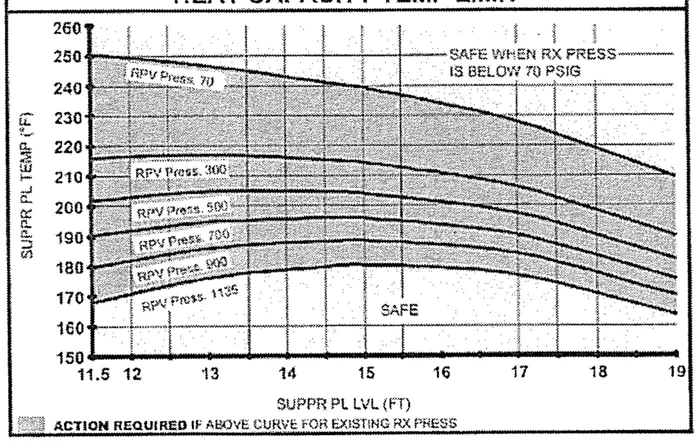
Description	Description	
1.1-U1 | NOTE	**1.1-U2**	
Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION: Mode 5	Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain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

NOTES

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

CURVES/TABLES:

CURVE 1.2-G
HEAT CAPACITY TEMP LIMIT

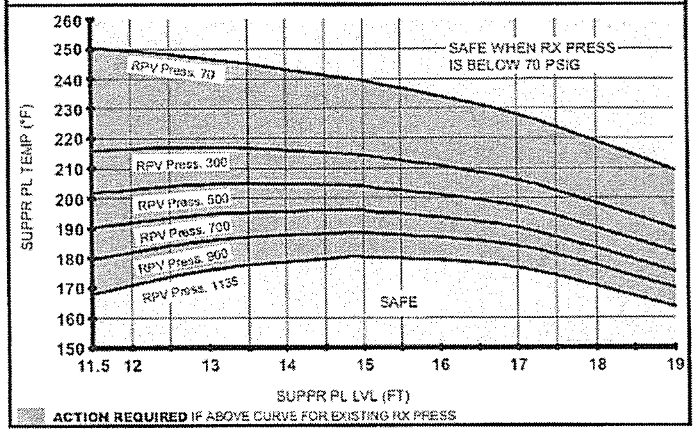


SCRAM FAILURE	REACTOR COOLANT ACTIVITY	
Description	**Description**	
	1.3-U Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample. OPERATING CONDITION ALL	**UNUSUAL EVENT**
1.2-A **NOTE** Failure of RPS automatic scram functions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

NOTES

CURVES/TABLES:

CURVE 1.5-S
HEAT CAPACITY TEMP LIMIT



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

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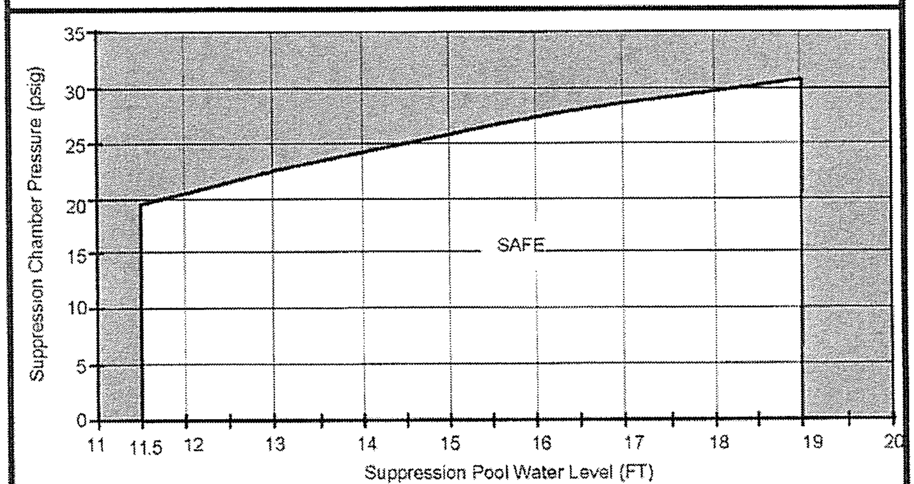
PRIMARY CONTAINMENT 2.0

NOTES

CURVES/TABLES:

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

CURVE 2.1-S
PRESS SUPPR PRESS



	PRIMARY CONTAINMENT PRESSURE Description	PRIMARY CONTAINMENT HYDROGEN Description
UNUSUAL EVENT		
ALERT	**2.1-A** **TABLE** Drywell pressure at or above 2.45 psig **AND** Indication of Primary System leakage into Primary Containment. Refer to Table 2.1-A. OPERATING CONDITION: Mode 1 or 2 or 3	
SITE EMERGENCY	**2.1-S** **CURVE** Suppression Chamber pressure can NOT be maintained in the safe area of Curve 2.1-S. OPERATING CONDITION: Mode 1 or 2 or 3	**2.2-S** Drywell or Suppression Chamber hydrogen concentration at or above 4% **AND** Drywell or Suppression Chamber oxygen concentration at or above 5%. OPERATING CONDITION: Mode 1 or 2 or 3
GENERAL EMERGENCY	**2.1-G** Suppression Chamber pressure can NOT be maintained below 55 psig. OPERATING CONDITION: Mode 1 or 2 or 3	**2.2-G** Drywell or Suppression Chamber hydrogen concentration at or above 6% **AND** Drywell or Suppression Chamber oxygen concentration at or above 5%. OPERATING CONDITION: Mode 1 or 2 or 3

NOTES

CURVES/TABLES:

TABLE 2.3-A/2.3-S2 DRYWELL RADIATION LEVELS WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	196	2-RE-90-272A	642	3-RE-90-272A	196
1-RE-90-273A	297	2-RE-90-273A	297	3-RE-90-273A	297

TABLE 2.3-S1/2.3-G2 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	2981	2-RE-90-272A	2263	3-RE-90-272A	2981
1-RE-90-273A	2960	2-RE-90-273A	2960	3-RE-90-273A	2960

TABLE 2.3-G1 DRYWELL RADIATION LEVELS WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	R/HR	RAD MONITOR	R/HR	RAD MONITOR	R/HR
1-RE-90-272A	90091	2-RE-90-272A	68405	3-RE-90-272A	90091
1-RE-90-273A	89450	2-RE-90-273A	89450	3-RE-90-273A	89450

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

DRYWELL RADIATION

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

NOTES

CURVES/TABLES:

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

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

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SECONDARY CONTAINMENT 3.0

NOTES

CURVES/TABLES:

TABLE 3.1 MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS				
AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE ^{0}F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

TABLE 3.1-G/3.2-G INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	≥ 196 R/HR
1-RE-90-273A	≥ 297 R/HR	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR
Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131	

SECONDARY CONTAINMENT TEMPERATURE

Description

UNUSUAL EVENT

ALERT

SITE EMERGENCY

3.1-S			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1.

OPERATING CONDITION:
Mode 1 or 2 or 3

GENERAL EMERGENCY

3.1-G			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment.

OPERATING CONDITION
Mode 1 or 2 or 3

NOTES

CURVES/TABLES:

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

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

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

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

SECONDARY CONTAINMENT RADIATION

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

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RADIOACTIVITY RELEASES 4.0

NOTES

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

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

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

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

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

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

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

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

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

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

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

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

CURVES/TABLES:

Table 4.1-U
RELEASE LIMITS FOR UNUSUAL EVENT

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

Table 4.1-A
RELEASE LIMITS FOR ALERT

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

Table 4.1-S
RELEASE LIMITS FOR SITE AREA EMERGENCY

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

Table 4.1-G
RELEASE LIMITS FOR GENERAL EMERGENCY

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

GASEOUS EFFLUENT

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

NOTES

CURVES/TABLES:

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

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LOSS OF POWER
5.0

NOTES

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

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

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

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

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

CURVES/TABLES:

Table 5.1
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

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

LOSS OF AC POWER

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

NOTES

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

CURVES/TABLES:

Table 5.2-U UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, AND D
UNIT 2	A, B, C, AND D
UNIT 3	3A, 3B, 3C, AND 3D

Table 5.2-S CRITICAL DC POWER AND ESSENTIAL SYSTEMS		
COMBINATION	LOSS OF CRITICAL 250V DC POWER (Unit Specific Unless Otherwise Noted)	POTENTIALLY RESULTS IN
I	Control Power for 4KV Unit Boards A, B, and C **AND** Control Power for 480V Unit Boards A and B **AND** Power for Panel 9-9 Cabinet 1	Loss of Main Condenser **AND** Loss of Both EHC Pumps **AND** Loss of All Reactor Feed Pumps
II	Power for 250V DC RMOV Board A	Loss of HPCI
III	Power for 250V DC RMOV Board C	Loss of RCIC
IV	Power for 250V DC RMOV Boards A, B, and C **AND** Control Power for 4KV Shutdown Boards A, B, C, and D (4KV Shutdown Boards 3A, 3B, 3C, and 3D for Unit 3)	Less than 4 MSRVs **AND** Loss of All RHR Pumps And Core Spray Pumps

LOSS OF 250V DC POWER

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

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HAZARDS
6.0

NOTES

CURVES/TABLES:

RADIOLOGICAL

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

NOTES

CURVES/TABLES:

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

NOTES

CURVES/TABLES:

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

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

FIRE / EXPLOSION

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

NOTES

CURVES/TABLES:

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

TOXIC GASES

Description	
6.5-U | | | **TABLE** | EITHER of the following conditions exists: • Normal operations impeded due to access restrictions caused by toxic gas concentrations within any building or structure listed in Table 6.5/6.6. • Confirmed report by local, county, or state officials that a large offsite toxic gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above the Permissible Exposure Limit (PEL) causing an evacuation of any site personnel. OPERATING CONDITION: ALL	**UNUSUAL EVENT**
6.5-A | | | **TABLE** | ALL of the following conditions exist: • Plant personnel report toxic gas within any building or structure listed in Table 6.5/6.6. • Plant personnel report severe adverse health reactions due to toxic gas (i.e., burning eyes, throat, or dizziness), or sampling results by Fire Protection or Industrial Safety personnel indicate levels above the Permissible Exposure Limit (PEL). • Determination by the Site Emergency Director that plant personnel would be unable to perform actions necessary to establish and maintain cold shutdown conditions while utilizing appropriate personnel protective equipment. OPERATING CONDITION: ALL	**ALERT**
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

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

FLAMMABLE GASES

Description

UNUSUAL EVENT

6.6-U			TABLE	

EITHER of the following conditions exists:

- Release of flammable gas within the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation.
- Confirmed report by local, county, or state officials that a large offsite flammable gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL).

OPERATING CONDITION:
ALL

ALERT

6.6-A			TABLE	

Release of flammable gases within any building or structure listed in Table 6.5/6.6 in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation.

OPERATING CONDITION:
ALL

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

SECURITY

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

NOTES

CURVES/TABLES:

VEHICLE CRASH

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

NOTES

CURVES/TABLES:

SPENT FUEL STORAGE

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

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NATURAL EVENTS
7.0

NOTES

CURVES/TABLES:

EARTHQUAKE

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

NOTES

CURVES/TABLES:

TORNADO / HIGH WINDS

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

NOTES

CURVES/TABLES:

FLOOD

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

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EMERGENCY DIRECTOR JUDGMENT 8.0

NOTES

CURVES/TABLES:

TECHNICAL SPECIFICATIONS

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

NOTES

CURVES/TABLES:

Table 8.2-U
LOSS OF COMMUNICATIONS

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

LOSS OF COMMUNICATION

Description

8.2-U			TABLE		UNUSUAL EVENT

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

OR

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

OPERATING CONDITOIN:
ALL

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

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

CURVES/TABLES:

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

LOSS OF ASSESSMENT CAPABILITY

Description

8.3-U				

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

AND

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

AND

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

OPERATING CONDITION:
MODE 1, or 2, or 3

UNUSUAL EVENT

8.3-A		**NOTE**		

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

AND

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

AND

EITHER of the following conditions exists:

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

OPERATING CONDITION:
MODE 1, or 2, or 3

ALERT

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

Loss of most or all annunciators associated with safety systems

AND

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

AND

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

AND

A significant transient is in progress.

OPERATING CONDITION:
MODE 1, or 2, or 3

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

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

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

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

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

CURVES/TABLES:

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

OTHER

Description

8.4-U		NOTE	TABLE	

Events are in process or have occurred which indicate a potential degradation in the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Refer to Table 8.4-U for examples.

OR

Any loss or any potential loss of containment.

OPERATING CONDITION:
ALL

UNUSUAL EVENT

8.4-A		NOTE		

Events are in process or have occurred which involve an actual or potential substantial degradation in the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

OR

Any loss or potential loss of fuel cladding or RCS pressure boundary.

OPERATING CONDITION:
ALL

ALERT

8.4-S		NOTE		

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts (1) toward site personnel or equipment that could lead to the likely failure thereof or, (2) prevent effective access to equipment needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

OR

Any loss or potential loss of both fuel cladding and RCS pressure boundary.

OR

Potential loss of either fuel cladding or RCS pressure boundary and loss of any additional barrier.

OPERATING CONDITION:
ALL

SITE EMERGENCY

8.4-G		NOTE		

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

OR

Loss of any two barriers and potential loss of third barrier.

OPERATING CONDITION:
ALL

GENERAL EMERGENCY

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

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

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	<u>OR</u>		
	A.2	-------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -------------------------------- Place associated trip system in trip.	12 hours
B. -------------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -------------------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	<u>OR</u>		
	B.2	Place one trip system in trip.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
J. Required Action and associated Completion Time of Condition I not met.	J.1	Be in Mode 2.	4 hours

SURVEILLANCE REQUIREMENTS

---NOTES---
1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	-------------------------NOTE------------------------ Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------------NOTES----------------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	-------------------------NOTE------------------------- Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is $\geq$ 30% RTP.	24 months
SR 3.3.1.1.16	-------------------------NOTE------------------------- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.17	Verify OPRM is not bypassed when APRM Simulated Thermal Power is $\geq$ 25% and recirculation drive flow is < 60% of rated recirculation drive flow.	24 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

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

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

(c) [0.66 W + 66% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

TR 3.3 INSTRUMENTATION

TR 3.3.4 Control Rod Block Instrumentation

LCO 3.3.4 The control rod block instrumentation for each Function in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.4-1

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.4-1 for the Function.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.4-1.	B.1 Place at least one inoperable channel in the tripped condition.	1 hour
C. As required by Required Action A.1 and referenced in Table 3.3.4-1.	C.1 Place the channel in the tripped condition. OR C.2 Impose administrative controls to prevent control rod withdrawal.	Immediately Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.4-1.	-----------------NOTE------------------ Inoperable Rod Block Logic may also affect Technical Specifications LCO 3.3.2.1, LCO 3.9.1, and LCO 3.9.2. ---	
	D.1 Impose administrative controls to prevent control rod withdrawal.	Immediately

--NOTE--
Refer to Table 3.3.4-1 to determine which TSRs apply for each Function.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.4.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.4.2	------------------------NOTE------------------------ For APRM Function 1.b, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
TSR 3.3.4.3	------------------------NOTE------------------------ 1. For IRM Functions, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For SRM Functions, not required to be performed until 12 hours after IRMs on Range 2 or below. --- Perform CHANNEL FUNCTIONAL TEST.	Once within 7 days prior to startup <u>AND</u> 31 days thereafter

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.4.4	Perform CHANNEL FUNCTIONAL TEST.	Once within 7 days prior to startup AND Once per OPERATING CYCLE thereafter
TSR 3.3.4.5	Perform CHANNEL FUNCTIONAL TEST.	92 days
TSR 3.3.4.6	------------------------NOTE------------------------ 1. For IRM and SRM Functions, neutron detectors are excluded. 2. For IRM Functions, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For SRM Functions, not required to be performed until 12 hours after IRMs on Range 2 or below. --- Perform CHANNEL CALIBRATION.	92 days
TSR 3.3.4.7	------------------------NOTE------------------------ For APRM Functions, neutron detectors are excluded. --- Perform CHANNEL CALIBRATION.	Once per OPERATING CYCLE

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.4.8	Perform CHANNEL CALIBRATION.	24 months
TSR 3.3.4.9	Perform CHANNEL FUNCTIONAL TEST.	During OPERATING CYCLE

Table 3.3.4-1 (page 1 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP FUNCTION (a)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	TECHNICAL SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Average Power Range Monitors					
a. APRM Upscale (Flow Bias)	1	3	B	TSR 3.3.4.1 TSR 3.3.4.2 TSR 3.3.4.7	(b)
b. APRM Upscale (Startup) (c)	2	3	B	TSR 3.3.4.1 TSR 3.3.4.2 TSR 3.3.4.7	≤ 12%
c. APRM Downscale (d)	1	3	B	TSR 3.3.4.1 TSR 3.3.4.2 TSR 3.3.4.7	≥ 3%
d. APRM Inoperative	1, 2	3	B	TSR 3.3.4.1 TSR 3.3.4.2	(e)
2. Intermediate Range Monitors					
a. IRM Upscale (c)	2	6	B	TSR 3.3.4.1 TSR 3.3.4.3 TSR 3.3.4.6	≤ 108/125 of full scale
b. IRM Downscale (c) (f)	2	6	B	TSR 3.3.4.1 TSR 3.3.4.3 TSR 3.3.4.6	≥ 5/125 of full scale

(continued)

(a) During repair or calibration of equipment, not more than one SRM or APRM channel nor more than two IRM channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements.
(b) The APRM Rod Block Allowable Value shall be less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.
(c) This function is bypassed when the MODE switch is placed in the "Run" position.
(d) This function is only active when the MODE switch is in the "Run" position.
(e) The inoperative trips for the APRMs are produced by the following functions:
1. Local APRM chassis MODE switch not in operate.
2. Less than the required minimum number of LPRM inputs, both total and per axial level.
3. APRM module unplugged.
4. Self-test detected critical fault.
(f) IRM downscale is bypassed when it is on its lowest range.

Table 3.3.4-1 (page 2 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP FUNCTION (a)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	TECHNICAL SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Intermediate Range Monitors (continued)					
c. IRM Detector Not in Startup Position (c)	2	6	B	TSR 3.3.4.4 TSR 3.3.4.7	(g)
d. IRM Inoperative (c)	2	6	B	TSR 3.3.4.3	(h)
3. Source Range Monitor					
a. SRM Upscale (c)	2 (i)	3 (j)	B	TSR 3.3.4.1 TSR 3.3.4.3 TSR 3.3.4.6	$\leq 1X10^5$ counts/sec.
b. SRM Downscale (c) (k)	2 (i)	3 (j)	B	TSR 3.3.4.1 TSR 3.3.4.3 TSR 3.3.4.6	≥ 3 counts/sec.
c. SRM Detector not in Startup Position (c) (k)	2 (i)	3 (j)	B	TSR 3.3.4.4 TSR 3.3.4.7	(g)
d. SRM Inoperative (c)	2 (i)	3 (j)	B	TSR 3.3.4.3	(h)

(continued)

(a) During repair or calibration of equipment, not more than one SRM or APRM channel nor more than two IRM channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements.
(c) This function is bypassed when the MODE switch is placed in the "Run" position.
(g) Detector traverse is adjusted to 114 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.
(h) The inoperative trips for the SRMs and IRMs are produced by the following functions:
1. Local "operate-calibrate" switch not in operate.
2. Power supply voltage low.
3. Circuit boards not in circuit.
(i) With IRMs on Range 2 or below.
(j) IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions. IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.
(k) SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2. SRM detector not in startup position is bypassed when the count rate is ≥ 100 CPS or the above condition is satisfied.

Table 3.3.4-1 (page 3 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP FUNCTION (a)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	TECHNICAL SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Scram Discharge Volume Water Level					
a. High Water Level in West Scram Discharge Tank (LS-85-45L)	1,2	1 (l)	C	TSR 3.3.4.5 TSR 3.3.4.8	≤ 25 gal.
b. High Water Level in East Scram Discharge Tank (LS-85-45M)	1,2	1 (l)	C	TSR 3.3.4.5 TSR 3.3.4.8	≤ 25 gal.
5. Rod Block Logic	1,2	1	D	TSR 3.3.4.9	N/A

(a) During repair or calibration of equipment, not more than one SRM or APRM channel nor more than two IRM channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements.
(l) This function may be bypassed when the Reactor MODE Switch is in the "Shutdown" or "Refuel" position.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

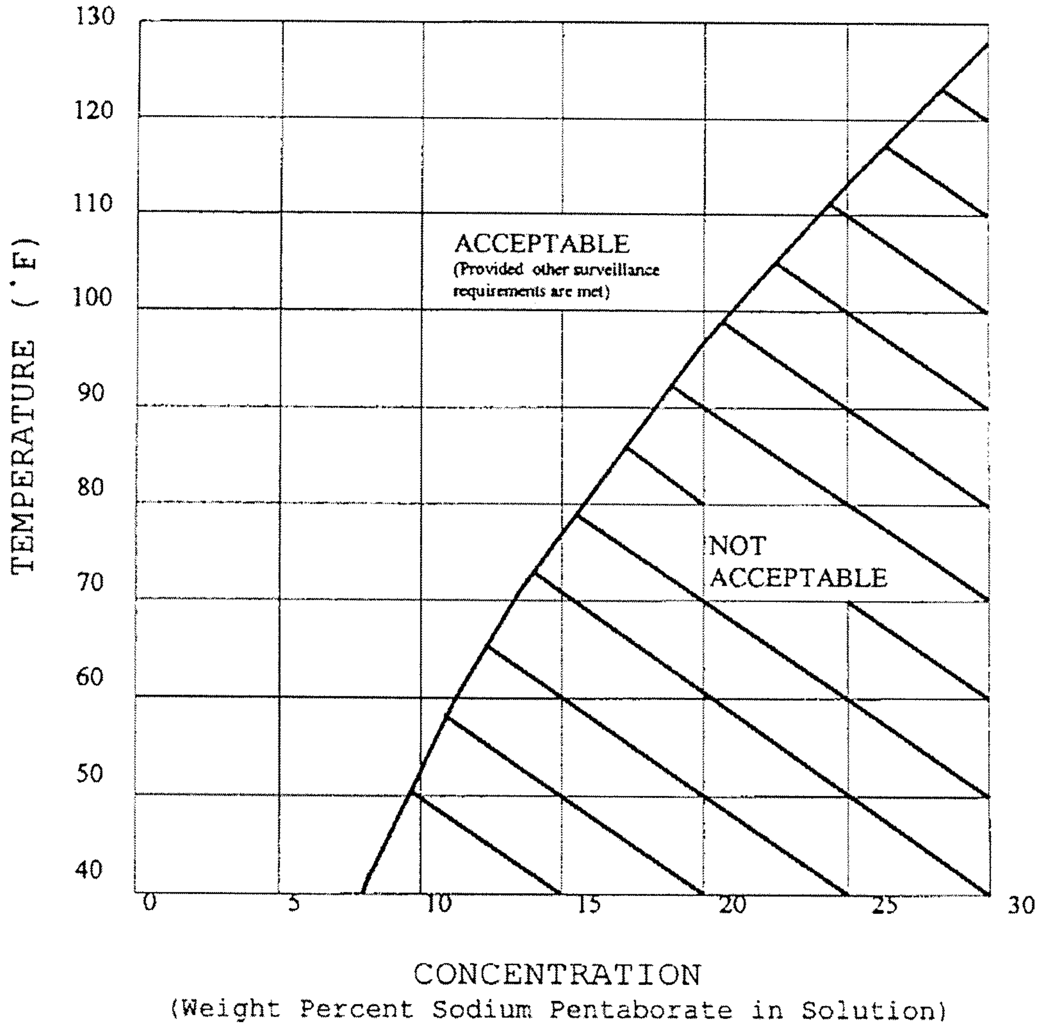

Figure 3.1.7-1
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days[1]

(continued)

[1] - This Completion Time may be extended to 14 days on a one-time basis. This temporary approval expires June 1, 2005.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	Immediately 14 days
D. HPCI System inoperable. AND Condition A entered.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. AND Condition A entered.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1 AND G.2	Be in MODE 3. Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. <u>OR</u> One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	Immediately 14 days
D. HPCI System inoperable. AND Condition A entered.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. AND Condition A entered.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1 AND G.2	Be in MODE 3. Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

---NOTES---

1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. --------------------------------- One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours except for main steam line <u>AND</u> 8 hours for main steam line
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. ----------------------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. ---------------------------------- One or more penetration flow paths with two PCIVs inoperable except due to MSIV leakage not within limits.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. --------------NOTE------------ Only applicable to penetration flow paths with only one PCIV. ---------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours except for excess flow check valves (EFCVs) <u>AND</u> 12 hours for EFCVs
	<u>AND</u>		
	C.2	--------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. ---------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with MSIV leakage not within limits.	D.1	Restore leakage rate to within limit.	4 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1	Be in MODE 3.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	F.1	Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	OR		
	F.2	--------------NOTE------------ Only applicable for inoperable RHR Shutdown Cooling Valves. ----------------------------------- Initiate action to restore valve(s) to OPERABLE status.	Immediately

Appendix A
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Reporting of Events or Conditions Affecting
Licensed Nuclear Power Plants

1.0 PURPOSE

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

2.0 SCOPE

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

NOTES

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

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

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

Appendix A
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Reporting of Events or Conditions Affecting
Licensed Nuclear Power Plants

3.0 REQUIREMENTS

NOTES

1) Internal management notification requirements for plant events are found in Appendix D. The Operations Shift Manager is responsible for notifying Site Operations Management and the Duty Plant Manager. The Duty Plant Manager is responsible for making the remaining internal management notifications.

2) NRC NUREG-1022, Supplements and subsequent revisions should be used as guidance for determining reportability of plant events pursuant to §50.72 and §50.73. A text searchable copy of NUREG-1022 is maintained on the TVA NPG Nuclear Licensing Webpage at address http://tvanweb.cha.tva.gov/licensing/Pages/NRC-Industry_Guidance_Documents.htm.

3.1 Immediate Notification - NRC

TVA is required by §50.72 to notify NRC immediately if certain types of events occur. This appendix contains the types of events and the allotted time in which NRC must be notified. (Refer to Form NPG-SPP-03.5-1 or NRC Form 361). Operations is responsible for making the reportability determinations for §50.72 and §50.73 reports. For any event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event shall the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §50.72.

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

NOTE

The NRC Event Notification Worksheet may be used in preparing for notifying the NRC. This Worksheet may be obtained directly from the NRC website (www.nrc.gov) under Form 361, or TVA NPG Form NPG-SPP-03.5-1 may be used.

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

B. The following criteria require 1-hour notification:

1. (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been violated.

Reporting of Events or Conditions Affecting
Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

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

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

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

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

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

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

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

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

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

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

Appendix A
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Reporting of Events or Conditions Affecting
Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

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

C. The following criteria require 4-hour notification:

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

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

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

NOTES

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

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

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

D. The following criteria require 8-hour notification:

NOTE

The non-emergency events specified below are only reportable if they occurred within three years of the date of discovery.

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

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Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

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

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

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

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

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

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

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

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

      (5) BWR reactor core isolation cooling system.

      (6) PWR auxiliary or emergency feedwater system.

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

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

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Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

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

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

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

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

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

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

E. Follow-up Notification (§50.72(c))

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

1. Immediately report:

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

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3.1 Immediate Notification - NRC (continued)

not been previously made; or

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

(iii) A termination of the Emergency Class.

(1) Immediately report:

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

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

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

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

3.2 Twenty-Four Hour Notification - NRC

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

3.3 Two-Day Notification - NRC

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

3.4 Sixty-Day Verbal Report

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

A. Verbal Report Required Content:

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3.4 Sixty-Day Verbal Report (continued)

If the verbal notification option is selected (NUREG 1022, Revision 2, Section 3.2.6., System Actuation), instead of an LER, the verbal report:

1. Is not considered an LER.
2. Should identify that the report is being made under §50.73(a)(2)(iv)(A).
3. Should provide the following information:
   a. The specific train(s) and system(s) that were actuated.
   b. Whether each train actuation was complete or partial.
   c. Whether or not the system started and functioned successfully.

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

B. Verbal Report Development and Review

Licensing will:

1. Develop (with input from responsible organization) the response (i.e., report summary) to address the required input.
2. Ensure that the reporting details are approved by site vice president or his designee prior to making the verbal report.

C. Telephone Report Timeliness

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

3.5 Written Report - NRC

A. A report on a Safety Limit Violation shall be submitted to the NRC, the NSRB, and the Site Vice President if required by Technical Specifications.

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

C. Reporting Radiation Injuries

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3.5 Written Report - NRC (continued)

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

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

   b. In the course of transportation; or

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

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

D. Licensee Event Reports

A written report shall be prepared in accordance with §50.73(a)(i) for items in the 60-day report criteria or Technical Specifications. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the USNRC, Document Control Desk, Washington, DC 20555. NUREG 1022, Revision 2, contains the instructions for completion of the LER form. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports (or optional telephone reports [refer to Appendix A, Section 3.4]) required by §50.73.

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

E. Report Criteria

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

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

   a. The Technical Specification is administrative in nature;

3.5 Written Report - NRC (continued)

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

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

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

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

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

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

7. §50.73(a)(2)(iv)(A) - Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) [see list in Section 3.5E.8 below], except when

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

b. The actuation was invalid and

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

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

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3.5 Written Report - NRC (continued)

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

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

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

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

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

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

   e. BWR reactor core isolation cooling system.

   f. PWR auxiliary or emergency feedwater system.

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

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

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

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

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

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3.5 Written Report - NRC (continued)

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

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

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

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

    (B) Remove residual heat;

    (C) Control the release of radioactive material; or

    (D) Mitigate the consequences of an accident.

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

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

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3.5 Written Report - NRC (continued)

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

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

    b. Remove residual heat;

    c. Control the release of radioactive material; or

    d. Mitigate the consequences of an accident.

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

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

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

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

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

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

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3.5 Written Report - NRC (continued)

(1) Excessive bird impaction events;

(2) Onsite plant or animal disease outbreaks;

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

(4) Fish kills near the plant site;

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

WBN only

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

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

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

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

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

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

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