U.S. Nuclea	r Regulatory	Commission
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Site-Specific SRO Written Examination

Applicant Information		
Name:		
Date:	Facility/Unit: Browns Ferry	
Region: II	Reactor Type: 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.		
RO/SRO-Only/Total Examination Values	<u>75</u> / <u>25</u> / <u>100</u> Points	
Applicant's Scores	/ / Points	
Applicant's Grade	/ / Percent	

Test	15-01	NKC	EXAM	KEY
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LXR.TESTTM **Response Form** LXR-20020 Side 1

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A Mark responses darkly and fill completely. *> Erase unwanted marks clearly

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Signature:

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A If Side 2 is used, fill in ID on both sides

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SRO Exam Q# 76-100

QUESTION ID	NUREG K/A CROSS REF						
1	295001 AA2.02	26	295033 EK2.01	51	264000 A1.01	76	295006 G2.4.30
2	295003 AA1.02	27	295035 EA1.01	52	300000 A4.01	77	295021 AA2.02
3	295004 AK1.05	28	203000 G2.2.38	53	400000 K3.01	78	295023 G2.4.21
4	295005 G2.4.21	29	205000 K6.04	54	201006 A2.05	79	295026 EA2.03
5	295006 AK2.06	30	206000 K2.04	55	202002 A1.07	80	295028 EA2.05
6	295016 G2.4.34	31	209001 A4.03	56	215001 K1.02	81	295037 EA2.03
7	295018 AK1.01	32	209001 K6.04	57	223001 K6.09	82	295038 G2.4.41
8	295019 AK3.02	33	211000 A2.02	58	226001 K3.02	83	295015 AA2.01
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11	295024 EK2.03	36	215003 K4.04	61	239001 K4.05	86	203000 G2.2.37
12	295025 G2.2.44	37	215004 A1.05	62	256000 G2.2.4	87	215004 G2.4.47
13	295026 EK3.04	38	215004 K5.03	63	271000 K3.02	88	215005 A2.04
14	295028 EK3.05	39	215005 K4.01	64	290002 K5.07	89	218000 A2.02
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16	295031 EA1.05	41	218000 K5.01	66	2.1.19	91	201001 A2.08
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18	295038 EK2.02	43	239002 A3.02	68	2.2.17	93	272000 A2.02
19	600000 AA2.02	44	239002 K3.03	69	2.2.2	94	2.1.34
20	700000 AK3.02	45	259002 A2.04	70	2.2.40	95	2.1.7
21	295002 G2.4.11	46	259002 K3.03	71	2.3.11	96	2.2.19
22	295009 AA1.04	47	261000 A1.02	72	2.3.15	97	2.2.21
23	295017 AK1.02	48	262001 K2.01	73	2.3.7	98	2.3.14
24	295020 AK3.01	49	262002 A4.01	74	2.4.29	99	2.4.11
25	295029 G2.1.7	50	263000 K1.01	75	2.4.5	100	2.4.29

Night shift turned over all three units operating at 100 percent power. Unit 1 scrammed at 1000 due to a Generator Load Reject.

In accordance with NPG-SPP-03.5 which one of the following completes the statements below?

The <u>latest</u> time that the NRC is required to be notified of this event is (1), and (2) is responsible for making this NRC notification.

- A. (1) 1400 (2)Site Licensing
- B. (1) 1800(2) Site Licensing
- C. (1) 1400 (2) Operations
- D. (1) 1800 (2) Operations

Unit 3 is shut down due to an LCO 3.0.3 entry condition.

- Mode 4 was entered
- 3D RHR pump is in Shutdown Cooling
- Loop 1 RHR Shutdown Cooling is Not available
- RWCU is in service
- Drywell Equipment Hatch is open

Subsequently plant equipment problems result in the following conditions:

- RHR Pump 3D flow lowers to 3000 gpm and cannot be raised
- 3-TR-56-4 Point 7, 3-TE-56-8 Reactor Vessel Drain to RWCU indicates 218° F
- Reactor Steam Dome Pressure is 1.5 psig

Which ONE of the following completes both statements below?

At this flowrate, RHR Shutdown Cooling __ (1) __ considered in-service. In accordance with EPIP-1, Emergency Plan Implementing Procedure, __ (2)___.

- A. (1) is(2) No EAL is exceeded
- B. (1) is(2) an Alert EAL is met
- C. (1) is Not (2) No EAL is exceeded
- D. (1) is Not(2) an Alert EAL is met

Unit 1, 2, and 3 are currently at 100% power.

While fuel movement was in progress in Unit 1 Spent Fuel Pool, several fuel bundles were damaged.

Subsequently:

- The Field Assessment Team reported that the Site Boundary Radiation Reading reached 10 MREM/HR at 0840, and is now 12 MREM/HR and rising slowly at 0900.
- Radiation levels on the refuel floor elevation 664 and the Recirc MG set area elevation 639 are above the Max Safe value.

Which one of the following completes both statements below?

Entry into 0-EOI 4, Radioactive Release Control, __ (1) __ required.

Entry into 1-C-2, Emergency RPV Depressurization, __ (2) __ required.

- A. (1) is (2) is Not
- B. (1) is (2) is
- C. (1) is Not (2) is Not
- D. (1) is Not (2) is

Unit 1 was at 100% power when one Safety Relief Valve failed open and was unable to be closed. The Reactor Mode Switch was placed in SHUTDOWN.

The following conditions exist:

- Reactor power 20% and lowering due to SLC injection
- Reactor pressure 900 psig and stable
- Suppression pool temperature 190 °F and rising
- Suppression pool level 16.0 ft. and slowly rising

Which ONE of the following identifies the <u>Minimum</u> required action in accordance with 1-EOI-1, RPV Control and 1-EOI-2, Primary Containment Control?

- A. Lower reactor pressure, must maintain ≤ 100 °F/Hr
- B. Emergency depressurize (ED) the RPV using the safety relief valves
- C. Rapidly depressurize the RPV with the main turbine bypass valves
- D. Do Not ED, lower reactor pressure, OK to exceed 100 °F/Hr

An ATWS and a LOCA have occurred on Unit 2, resulting in the following plant conditions:

- Suppression Chamber Pressure is 53 psig and rising 1 psig every 2 minutes
- Drywell Temperature is 325°F and rising 1°F every 5 minutes
- Suppression Pool Level is 18 feet

Which ONE of the following identifies the required procedure(s) in accordance with 2-EOI-2, Primary Containment Control?

- A. EOI appendix 12 Primary Containment Venting and EOI appendix 17B RHR System Operation Drywell Sprays
- B. EOI appendix 17B RHR System Operation Drywell Sprays ONLY
- C. EOI appendix 13 Emergency Venting Primary Containment ONLY
- D. EOI appendix 17B RHR System Operation Drywell Sprays and EOI appendix 13 Emergency Venting Primary Containment

An ATWS has occurred on Unit 2.

- RPV water level was lowered in accordance with 2-C-5, Level / Power Control
- SLC is injecting, SLC Tank level is 80%
- MSRVs are being cycled for pressure control

Subsequently the following indications and alarms are reported:

- RPV water level is (-) 35 inches
- Suppression Pool Temperature is 106 °F
- SLC tank level is 63%
- RPV water level is being restored to (+) 2 to (+) 51 inches
- SRM PERIOD, (2-9-5A, Window 20)
- APRM downscale lights extinguished

Which ONE of the following describes the action(s) that is(are) required in accordance with 2-C-5, Level / Power Control?

- A. Continue to raise RPV water level, but at a slower rate.
- B. Stop raising RPV water level.Maintain RPV water level with (-) 35 inches as the upper limit.
- C. Stop and prevent all injection into the RPV (except RCIC, CRD & SLC). Re-inject when RPV water level drops below (-) 50 inches.
- D. Stop and prevent all injection into the RPV (except RCIC, CRD & SLC). Re-inject when reactor power is < 5%.

Unit 2 is operating at 96% power when the following occurred:

- 2A Recirculation Pump speed started to rise rapidly
- The US directed tripping the 2A Recirc pump
- A reactor scram was inserted due to the power excursion

Subsequently

- 2-TIS-1-60C temperature is rising rapidly
- Numerous Turbine Building radiation alarms are received
- The US then directs closing the MSIVs

The following conditions currently exist:

- All control rods are FULL IN
- Reactor pressure is 780 psig and lowering
- 2-TIS-1-60C Main Steam Tunnel Temperature indicates 320 °F and rising slowly
- B Main Steam Line has failed to isolate
- 2-RE-90-272A Drywell High Range Radiation Monitor indicates 310 R/HR
- 2-RE-90-273A Drywell High Range Radiation Monitor indicates 300 R/HR

Which ONE of the following identifies the highest required Emergency Action Level/Designator to be declared in accordance with the Emergency Classification Procedure - EPIP-1?

- A. Notification of Unusual Event
- B. Alert
- C. Site Area Emergency
- D. General Emergency

Unit 1 has scrammed with the following conditions:

- 19 Control Rods are at position 48
- ATWS actions are complete
- SLC is NOT injecting
- MSIVs are open and Reactor pressure is stable on bypass valves
- 1-EOI-1, RPV Control was entered on low Reactor water level

Subsequently, the OATC reports that Reactor Power is on range 8 of the IRMs.

Which ONE of the following identifies the required procedures for reactor power <u>and</u> level control?

- A. 1-EOI-1, RPV Control, RC/Q and RC/L
- B. 1-AOI-100-1, Reactor Scram and RC/L
- C. 1-EOI-1, RPV Control, RC/Q and 1-C-5, Level / Power Control
- D. 1-AOI-100-1, Reactor Scram and 1-C-5, Level / Power Control

Unit 2 is operating at 100% power.

A leak has occurred in the RWCU Heat Exchanger Room.

- At 0800 Heat Exchanger Room maximum safe temperature (220 °F) was reached.
- RWCU Heat Exchanger Room temperature is rising at 1°F per minute
- At 0815 The UO reports the following RPV water level readings: 2-LI-3-58A, Emergency Range reading (-) 4 inches
 2-LI-3-208A, Normal Range reading 16 inches

ASSUME that: RPV water level remains constant

RWCU Heat Exchanger room temperature rate of rise remains constant.

In accordance with Caution 1 which ONE of the following completes the statements below?

At 0815 __ (1) __ level instrument(s) may be used to determine or trend Reactor Water level.

At 0840 based on the level indications given 2-EOI-C4, RPV flooding, __(2) __ required to be entered.

- A. (1) 2-LI-3-208A and 2-LI-3-58A (2) is
- B. (1) 2-LI-3-208A and 2-LI-3-58A (2) is Not
- C. (1) 2-LI-3-58A only (2) is
- D. (1) 2-LI-3-58A only (2) is Not

Unit 1 is operating at 100% power when the following alarms are received.

- 1-9-3B window 10 PRI CONTAINMENT NITROGEN PRESS HI
- 1-9-3B window 19 DRYWELL NORM OPERATING PRESS HIGH
- 1-9-3B window 23 DRYWELL PRESSURE HIGH
- 1-9-3B window 30 DRYWELL PRESS APPROACHING SCRAM

Which one of the above alarms, if valid, requires the Shift Manager to classify an event in accordance with EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE?

A. Window 10

- B. Window 19
- C. Window 23
- D. Window 30

Q 85

Unit 2 is Shutdown with a cooldown in progress at a Reactor Pressure of 90 psig.

In accordance with the bases for Technical Specifications 3.4.7, Residual Heat Removal (RHR) Shutdown Cooling System-Hot Shutdown, and 3.5.2, ECCS-Shutdown:

- (1) How many RHR Shutdown Cooling Subsystems are there?
- (2) If RHR Loop I is aligned for Shutdown Cooling can it be considered an Operable LPCI Subsystem?
- A. Two; Yes
- B. Four; Yes
- C. Two; No
- D. Four; No

Unit 1 startup in accordance with 1-GOI-100-1A, UNIT STARTUP, is in progress. The following conditions currently exist:

- SRMs are reading between 700 and 1000 cps
- All IRMs are on range 1 or 2

Subsequently:

- At **08:00** SRM A began spiking causing a control rod block and was bypassed.
- SRM A continued to spike until **08:20** and then returned to a stable reading, comparable to the other SRMs.

What is/are the MINIMUM action(s) required, if any, for that SRM to be returned to OPERABLE status?

- A. Un-bypass the SRM, no further actions are required.
- B. Observe the SRM for at least 15 minutes before returning the instrument to service with concurrence from System Engineering.
- C. Perform Surveillance 1-SR-3.3.1.2.4, SRM System Count Rate and Signal to Noise Ratio Check.
- D. Perform Surveillance 1-SR-3.3.1.2.5 & 6, SRM Functional Test with Reactor Mode Switch Not in Run.

Unit 3 has entered Mode 1 at **08:00** on June 1st when the following sequence of events occurs:

10:00 APRM Voter 1 failed its surveillance and did NOT generate an output signal to RPS.

11:00 The IMs report that a review of the surveillance indicates that the APRM **Voter 4** also failed acceptance criteria and has been declared INOP.

Which of the following is the most limiting Technical Specification required actions for these conditions?

- A. Required Action A.1 <u>OR</u> A.2 must be performed by 22:00.
- B. Required Action B.1 OR B.2 must be performed by 17:00.
- C. Required Action C.1 must be performed by 12:00.
- D. Required Action G.1 must be performed by 23:00.

Unit 3 is operating at 100% power, at 09:00:00 a LOCA occurs.

09:00:30 (30 seconds after the LOCA) the following conditions exist:

- RPV water level is (-) 125 inches and lowering
- Drywell Pressure is 15 psig and rising
- ADS BLOWDOWN AUX RELAYS ENERGIZED (3-9-3C, window 4) in alarm
- ADS BLOWDOWN TIMERS INITIATED (3-9-3C, window 11) in alarm
- ALL ECCS Systems are operating as expected

09:01:30:

• ECCS systems are injecting and the Unit Supervisor has determined that Reactor water level **can** be restored and maintained above (-) 162 inches

Which ONE of the following is required concerning ADS and what is the required procedure for Reactor water level control?

- A. Do Not Inhibit ADS and execute C-1Alternate Level Control
- B. Do Not Inhibit ADS and execute EOI-1 RC/L
- C. Inhibit ADS and execute C-1Alternate Level Control
- D. Inhibit ADS and execute EOI-1 RC/L

Unit 2 is operating at 100% power when the following event occurs:

6-1-14 at 09:00 Reactor Vessel Steam Dome Pressure (2-PIS-3-22AA) fails upscale causing a half scram. All required Tech Spec actions were taken.

6-1-14 at 10:00 Drywell Pressure High (2-PIS-64-56B) fails downscale.

Which one of the following identifies the <u>earliest</u> time that Unit 2 is required to be in cold shutdown if 2-PIS-64-56B cannot be restored to operable status or placed in the tripped condition.

(Consider ONLY Tech Spec 3.3.6.1, PCIS instrumentation requirements)

- A. 6-2-14 at 22:00
- B. 6-2-14 at 23:00
- C. 6-3-14 at 10:00
- D. 6-3-14 at 22:00

Unit 3 is operating at 100% power with several control rods declared SLOW due to scram time testing data in accordance with Tech Spec 3.1.4, Control Rod Scram Times.

(See attached illustration).

Subsequently,

The CRD pump tripped and was restarted in accordance with 3-AOI-85-3, CRD System Failure.

- During the time the CRD pump was not running, the CONTROL ROD DRIVE UNIT TEMP HIGH (3-9-5A, Window 17) annunciator alarmed.
- ALL actions required by 3-ARP-9-5A, Window 17 were completed.
- CRD 34-19 temperature is now 351°F and stable.

Which one of the following completes both statements?

CRD 34-19 (1) required to be declared SLOW. Tech Spec LCO 3.1.4, Control Rod Scram Times, (2) met.

[Reference and Illustration Provided]

- A. (1) is (2) is Not
- B. (1) is (2) is
- C. (1) is Not (2) is Not
- D. (1) is Not (2) is



Which one of the following completes both statements regarding the Traversing Incore Probe (TIP) system on Unit 2?

In accordance with Tech Spec 3.6.1.3 the TIP ball valves ____(1) ___ primary containment isolation valves.

__(2) __ is the procedure that contains guidance for firing a TIP shear valve when a TIP ball valve fails to close.

A. (1) are

(2) 2-AOI-64-2E, Traversing Incore Probe Isolation

- B. (1) are(2) 2-OI-94, Traversing Incore Probe System
- C. (1) are Not(2) 2-AOI-64-2E, Traversing Incore Probe Isolation
- D. (1) are Not(2) 2-OI-94, Traversing Incore Probe System

On 7-1-14, Unit 3 is operating at 100% power.

10:00 Loss of RPS "A" occurred and the RPS transformer is tagged out.

If RPS A cannot be restored when is the unit <u>required</u> by Tech Specs to be in MODE 3?

(Assume No Operator actions)

- A. 7-1 @ 20:00
- B. 7-1 @ 23:00
- C. 7-2 @ 10:00
- D. 7-2 @ 22:00

Unit 1 is MODE 2 with a startup in progress in accordance with 1-GOI-100-1A, Unit Startup. Reactor Pressure is 955 psig and the first bypass valve is 10% open. The US is preparing to brief the crew on swapping auxiliary steam loads to Main Steam.

Chemistry reports the following reactor water chemistry parameters to the Control Room:

- Chlorides: 0.09 ppm
- Conductivity: 1.5 µmhos/cm
- pH: 5.0

Which ONE of the following identifies the minimum **required** action(s) in accordance with TRM 3.4.1, Coolant Chemistry Limits?

- A. Condition A.1 only
- B. Condition B.1 only
- C. Condition C.1 only
- D. Condition D.1

Which one of the following completes both statements as they pertain to EECW/RHRSW in accordance with OPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking?

- 1. When three operable EECW pumps exist prior to a DG becoming inoperable, Then__(1) __redundant EECW Pump needs to be declared inoperable.
- 2. There are ____ (2) ____ redundant RHRSW subsystems per unit.
- A. (1) one (2) two
- B. (1) one (2) four
- C. (1) no (2) two
- D. (1) no (2) four

Which ONE of the following completes both statements regarding emergency (priority 1) work orders?

In accordance with NPG-SPP-07.1.4, Work Control Prioritization - On Line, emergency (priority 1) work orders require the approval of the (1).

In accordance with NPG-SPP-06.1, Work Order Process, planningfor emergency (priority 1) work orders___(2) ___ required prior to work performance.

- A. (1) Shift Manager(2) is
- B. (1) Shift Manager(2) is Not
- C. (1) Plant Manager (2) is
- D. (1) Plant Manager(2) is Not

To comply with Technical Specifications:

Which of the following completes the statements below?

LCO (1) establishes the allowance to restore inoperable equipment to service to demonstrate its operability.

This allowance ___(2) ___ applicable to restoring equipment to service to demonstrate the operability of OTHER equipment.

- A. (1) 3.0.5 (2) is also
- B. (1) 3.0.5 (2) is Not
- C. (1) 3.0.4 (2) is also
- D. (1) 3.0.4 (2) is Not

An event involving fuel damage has occurred on Unit 1. The following conditions exist at 09:00:

- Stack Noble Gas WRGERM: 7.1 X 10⁹µCi/sec
- 0-SI-4.8.B.1.a.1, Airborne Effluent Release Rate, Release Fraction: 15
- Four areas in the Reactor Building exceed their Max Safe Radiation levels
- Site Boundary Radiation Readings not obtained yet, but will be available at 09:30

What is the highest REQUIRED emergency classification at 09:15?

- A. Unusual Event per 4.1-U
- B. Alert per 4.1-A
- C. Site Area Emergency per 4.1-S
- D. General Emergency per 4.1-G

What procedure provides the guidance for abnormal/emergency annunciator response and what are the requirements for returning to normal annunciator response?

- A. BFN-ODM-4.20, Strategies for Successful Transient Mitigation; Normal annunciator response will be resumed when abnormal/emergency procedures are exited only.
- B. BFN-ODM-4.20, Strategies for Successful Transient Mitigation; Normal annunciator response will be resumed when abnormal/emergency procedures are exited or at SM/US discretion.
- C. OPDP-1, Conduct of Operations; Normal annunciator response will be resumed when abnormal/emergency procedures are exited only.
- D. OPDP-1, Conduct of Operations; Normal annunciator response will be resumed when abnormal/emergency procedures are exited or at SM/US discretion.

Which one of the following completes both statements In accordance with EPIP-1 Emergency Classification Procedure?

IF an Emergency Action Level (EAL) for a higher classification was exceeded, but the present situation indicates a lower classification, THEN the higher classification __ (1) __ be declared.

IF an Emergency Action Level (EAL) was exceeded but has now been totally resolved, THEN the NRC (2) required to be notified.

- A. (1) should Not (2) is
- B. (1) should Not(2) is Not
- C. (1) should still (2) is
- D. (1) should still (2) is Not

RO EXAM REFERENCES

RO Q 9. 2-AOI-74-1, Loss of Shutdown Cooling, Rev 39

SRO EXAM REFERENCES

- SRO Q 76. NPG-SPP-03.5, Regulatory Reporting Requirements, Rev. 0010
- SRO Q 77. EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, Rev 50
- SRO Q 78. EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, Rev 50
- SRO Q 82. EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, Rev 50
- SRO Q 98. EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, Rev 50
- SRO Q 79. EOI-CURVE 3, Heat Capacity Temp Limit.
- SRO Q 84. EOI-CAUTION 1, EOI-CURVE 8 and Table 6.
- SRO Q 87. T.S. 3.3.1.2 Source Range Monitor (SRM) Instrumentation
- SRO Q 88. T.S. 3.3.1.1 Reactor Protection System (RPS) Instrumentation
- SRO Q 90. T.S. 3.3.6.1 Primary Containment Isolation Instrumentation
- SRO Q 91. T.S. 3.1.4 Control Rod Scram Times / Core Matrix showing location of slow rods
- SRO Q 93. T.S. 3.4.5 RCS Leakage Detection Instrumentation
- SRO Q 94. TR 3.4.1 Coolant Chemistry

M NPG Standard Programs and	Regulatory Reporting Requirements	NPG-SPP-03.5 Rev. 0010 Page 1 of 100 Quality Related □ Yes ☑ No
Processes		Validation Date 07-28-2011 Review Frequency 3 years Validated By Henry Lee Effective Date 03-19-2014
	L	
Responsible P	eer Team/Working Group: Licensing	
Approved by:	Ed Schrull Corporate Functional Area Manage	03-18-2014 er Date

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6.2 Developmental References (continued)

NPG-SPP-01.6, Nuclear Power Group Corporate Duty Officer

NSDP-1, Safeguards Event Reporting Guidelines

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

- 1. (Technical Specifications) Safety Limits as defined by the Technical Specifications which have been violated.
- 2. §50.72 (a)(1)(i) The declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan.



- 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 in accordance with 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.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

- 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.
- 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.
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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

D. The following criteria require 8-hour notification:

NOTE

With the exception of "Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time 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.
- 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.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

- (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).

NOTE

For systems within scope, the inadvertent TS inoperability of a system in a required mode of applicability constitutes an event or condition for which there is no longer reasonable expectation that equipment can fulfill its safety function. Therefore, such events or conditions are reportable as an "Event or Condition that Could Have Prevented Fulfillment of a Safety Function."

- 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.

Appendix A (Page 7 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

- 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:
 - 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 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.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

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:

If the verbal notification option is selected (NUREG 1022, Revision 3, 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.

Appendix A (Page 9 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.4 Sixty-Day Verbal Report (continued)

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
 - 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)

Appendix A (Page 10 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

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

Appendix A (Page 11 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

- 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.
- §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.

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]

- §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).

Appendix A (Page 12 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

- 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;
 - (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)].

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

- 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.
- §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.

Appendix A (Page 14 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

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.
- 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:
 - (1) Excessive bird impaction events;
 - (2) Onsite plant or animal disease outbreaks;
 - Unusual mortality of any species protected by the Endangered Species Act of 1973;
 - (4) Fish kills near the plant site;

Appendix A (Page 15 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

(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.



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BROWNS FERRY NUCLEAR PLANT

Unit 0

Emergency Plan Implementing Procedure

EPIP-1

EMERGENCY CLASSIFICATION PROCEDURE

Revision 0050 Quality Related Level of Use: Reference Use

Effective Date: 9/8/2014

Responsible Organization: Radiological Emergency Preparedness

PREPARED BY: Matthew L. Clark

APPROVED BY: Steven M. Bono

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REACTOR 1.0

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

TABLE 1.1 - G2 MINIMUM STEAM COOLING PRESS (MSCP)		
NUMBER OF OPEN MSRVs	MSCP (PSIG)	
6 or More	190	
5	230	
4	290	

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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WATER	LEVEL	
Description	Description	
1.1-U1NOTEUncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water.	1.1-U2 Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain covered by water.	UNUSUAL
OPERATING CONDITION: Mode 5	OPERATING CONDITION ALL	EVENT
1.1-A1NOTEUncontrolled water level decrease in Reactor Cavity expected to result in irradiated fuel assemblies being uncovered.OPERATING CONDITION: Mode 5	1.1-A2 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 Reactor water level can NOT be maintained above -162 inches. (TAF) OPERATING CONDITION: ALL	1.1-S2 Reactor water level can NOT be determined. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
1.1-G1 Reactor water level can NOT be restored and maintained above -180 inches. OPERATING CONDITION:	1.1-G2 NOTE TABLE 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 MSCP in Table 1.1-G2. OPERATING CONDITION: Mode 1 or 2 or 3	GENERAL EMERGENCY

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



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\$	SCRAN	1 FAIL	URE		RE	ACTO AC		OLAN1 Y		
	De	scription				De	scription			
					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			1.3-A					
And Manual scram or ARI (automatic or manual) was successful. OPERATING CONDITION: Mode 1 or 2			bring was	Reactor coolant activity exceeds 300 µCi/gm dose equivalent lodine-131 as determined by chemistry sample. OPERATING CONDITION: Mode 1 or 2 or 3			ALERT			
1.2-S		NOTE								
OPERATI Mode 1 of	automatic sc ng the reacto ING CONDIT r 2	rram, manu or subcritica FION:	iai scram, a	ind						SITE EMERGENCY
1.2-G	CURVE									
Failure of ARI. Read Either of t & S R & R a OPERAT Mode 1 o	automatic so ctor power is uppression F tefer to Curve teactor water nd maintaine ING CONDIT r 2	cram, manu above 3% AND conditions Pool temp e e 1.2-G. r level can l ed at or abo	ual scram, a exists: exceeds HC NOT be res ove -180 inc	and CTL. tored thes.						GENERAL EMERGENCY

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

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PRIMARY CONTAINMENT 2.0

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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



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PRIM	PRIMARY CONTAINMENT PRESSURE			INT	PRIM	ARY C HYD		AINME En	NT	
	De	escription				De	scription			
										UNUSUAL EVENT
2.1-A			TABLE	L						
Drywell pr	essure at or	above 2.4	5 psig							
		AND								
Indication Primary C	of Primary Sontainment.	System lea Refer to 1	kage into Γable 2.1-A.							LERT
OPERATI Mode 1 or	NG CONDI 2 or 3	FION:								
2.1-S	CURVE				2.2-S					
Suppressi maintaine	on Chambe d in the safe	r pressure area of Cu	can NOT be urve 2.1-S.	Ð	Drywell or hydrogen	 Suppressio concentratio 	n Chambe on at or abo	r ove 4%		LIS
					,					Ē
							AND			ME
					Drywell or	Suppressio	n Chambe	r 5.5%		RO
					Uxygen cc	ncentration		3 0 %.		ΈN
OPERATI Mode 1 or	NG CONDI ⁻ 2 or 3	TION:			OPERATI Mode 1 or	NG CONDIT 2 or 3	FION:			ICY
2.1-G					2.2-G					
Suppressi maintaine	on Chambe d below 55 i	r pressure	can NOT be	e	Drywell or bydrogen	Suppressio	n Chambe	r ove 6%		ര
		3								EZ
							AND			ER
					Drywell or	Suppressio	n Chambe	r - 50(P-
					oxygen co	ncentration	at or abov	e 5%.		Ē
										핏
										GE
OPERATI Mode 1 o	NG CONDI	TION:			OPERATI Mode 1 or	NG CONDI [®] r 2 or 3	FION:			ဂြ
						, .				

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

<u>NOTES</u>

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	WITH RCS B	TABLE 2.3-A/ DRYWELL RADIATI ARRIER INTACT INSID	2.3-S2 ON LEVELS E PRIMARY C	ONTAINMENT	
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

w	ITH RCS BAF	TABLE 2.3-S1/ DRYWELL RADIATI RRIER <u>NOT</u> INTACT INS	2.3-G2 ON LEVELS IDE PRIMAR	Y 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

N	/ITH RCS BAF	TABLE 2.3 DRYWELL RADIATI RRIER <u>NOT</u> INTACT INS	-G1 ON LEVELS IDE PRIMAR	Y 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

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DRYWELL RADIATION										
	De	scription				De	scription			
										UNUSUAL EVENT
2.3-A Drywell ra listed in Ta intact insid OPERATI Mode 1 or	diation level able 2.3-A/2 de Primary C NG CONDIT 2 or 3	s at or abo .3-S2, with Containmer FION:	TABLE ve the value the RCS ba it.	US s rrier						ALERT
2.3-S1			TABLE	US	2.3-S2			TABLE	US	
Drywell ra listed in T NOT intac OPERATI Mode 1 o	Idiation level able 2.3-S1/ It inside Prim NG CONDI ⁻ r 2 or 3	s at or abo 2.3-G2 with nary Contai FION:	ve the value n the RCS b inment.	es arrier	Drywell ra listed in Ta intact insid Either of tl Indicatio Refer to Primary maintain OPERATI Mode 1 or	diation leve able 2.3-A/2 de Primary (he following ns of loss o Table 2.3/2 Containmen red. NG CONDI	Is at or abo 2.3-S2, with Containmer AND exists: f Primary C 2.5-U. nt integrity of TION:	ve the value the RCS ba t, containment. can NOT be	s rrier	SITE EMERGENCY
2.3-G1 Drywell ra listed in T intact insid OPERAT	Indiation level able 2.3-G1 de Primary (ING CONDI [*] r 2 or 3	s at or abo with the R Containmer	TABLE ve the value CS barrier N ht.	US PS OT	2.3-G2 Drywell ra listed in T NOT intac Either of t • Indicatic Refer to • Primary maintair	idiation leve able 2.3-S1 able 2.3-S1 trinside Prin he following ons of loss of Table 2.3/2 Containme hed.	Is at or abc /2.3-G2 wit mary Conta AND g exists: of Primary C 2.5-U. nt integrity TION:	TABLE we the value h the RCS b inment, containment.	US arrier	GENERAL EMERGENCY

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

<u>NOTES</u>

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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

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DRYWELL INTERNAL LEAKAGE	LOSS OF PRIMARY CONTAINMENT	
	Description	
Drywell unidentified leakage exceeds 10 gpm OR Drywell identified leakage exceeds 40 gpm.	2.5-0 Inability to maintain Primary Containment pressure boundary. Refer to Table 2.3/2.5-U.	UNUSUA
OPERATING CONDITION: Mode 1 or 2 or 3	OPERATING CONDITION: Mode 1 or 2 or 3	L 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

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<u>NOTES</u>

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TABLE 3.1 MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS						
AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS	MAX S	ATING			
	(UNLESS OTHERWISE NOTED)	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	(XA-55-5B-32/33) Panel 9-5	175	170	175		
High	69-835A, B, C, D Aux Inst Room					
RWCU Outbd Isol VIv Area	69-29F	220	220	220		
RWCU Hx Area	69-29G	220	220	220		
RWCU Hx Exh Duct	<u>69-29H</u>	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								
l II	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 DRY					ELL RADIATION			
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	<u>></u> 196 R/HR			
1-RE-90-273A	<u>></u> 297 R/HR	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR			
Reactor Coolant Ac	tivity	Reactor Coolant A	ctivity	Reactor Coolant Activity				
$> 300 \mu$ Ci/gm Dose Equivalent		\geq 300 µCi/gm Dose Equivalent		\geq 300 µCi/gm Dose Equivalent				
Iodine 131	-	Iodine 131		Iodine 131				

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SECONDARY CONTAINMENT TEMPERATURE		
Description		
	UNUSUAL EVENT	
	ALERT	
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	SITE 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	GENERAL EMERGENCY	

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TABLE 3.2 MAXIMUM SAFE OPERATING AREA RADIATION LIMITS					
AREA	RAD MONITOR	MAX SA	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	<u>1</u> 000	1000	
MG Set Area	<u>90</u> -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	N/A	N/A	

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 DRYW	ELL RADIATION
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	<u>></u> 196 R/HR
1-RE-90-273A	<u>> 297 R/HR</u>	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR
Reactor Coolant Activity		Reactor Coolant Activity		Reactor Coolant A	ctivity
\geq 300 µCi/gm Dose Equivalent \geq 300 µCi/gm Dose Equivalent		\geq 300 µCi/gm Dose Equivalent			
Iodine 131 Iodine 131					

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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 • 1, 2, or 3-RA-90-140A, Refueling Zone Exhaust • 1, 2, or 3-RA-90-140A, Refueling Zone Exhaust • 0, 2, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0,	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

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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.

Table 4.1-U RELEASE LIMITS FOR UNUSUAL EVENT			
ТҮРЕ	MONITORING METHOD		DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88 X 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			
ТҮРЕ	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88 X 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			
ТҮРЕ	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9 X 10 ⁹ μCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	100 MREM/HR Gamma	1 Hour
Site Boundary lodine-131	Field Assessment Team	3.9 X 10 ⁻⁷ µCl /cm ³	1 Hour

Table 4.1-G RELEASE LIMITS FOR GENERAL EMERGENCY			
ТҮРЕ	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9 X 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 X 10 ⁻⁶ µCl / cm ³	1 Hour

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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

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

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LOSS OF POWER 5.0

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- **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.

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	

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LOSS OF AC POWER			
Description	Description		
5.1-UNOTETABLEUSLoss of normal and alternate supply voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes ANDANDAt least two Diesel Generators supplying power to unit specific 4KV shutdown boards listing in Table 5.1. OPERATING CONDITION: ALLIsting in CONDITION:		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	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	

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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.

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
	Control Power for 4KV Unit Boards A, B, and C	Loss of Main Condenser
	AND	AND
	Control Power for 480V Unit Boards A and B	Loss of Both EHC Pumps
	AND	AND
	Power for Panel 9-9 Cabinet 1	Loss of All Reactor Feed Pumps
	Power for 250V DC RMOV Board A	Loss of HPCI
111	Power for 250V DC RMOV Board C	Loss of RCIC
IV	Power for 250V DC RMOV Boards A, B, and C	Less than 4 MSRVs
	AND	AND
	Control Power for 4KV Shutdown Boards A, B, C, and D (4KV Shutdown Boards 3A, 3B, 3C, and 3D for Unit 3)	Loss of All RHR Pumps And Core Spray Pumps
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		LOS	SS OF	250	V DC F	POWER	२		
	De	scription				Des	scription		
5.2-U Unplanne unit specif Table 5.2- Unplanne specific 44 for greate OPERATI Modes 4 d	d loss of 250 fic 4KV shuto U for greate d loss of 250 30V shutdow r than 15 mir NG CONDIT or 5	NOTE V DC contr down board r than 15 m OR V DC contr m boards A nutes.	TABLE rol power to ls from ninutes rol power to and B	US ALL unit					UNUSUAL EVENT
								r i	
	Co								ALERT
5.2-S		NOTE	TABLE	US					
Loss of 2 (I, II, III, a Table 5.2 OPERAT Mode 1 o	50V DC pow nd IV) of ess -S for greate ING CONDIT r 2 or 3	er to ALL c ential syste r than 15 m ΓΙΟΝ:	ombinations ems from hinutes.	5					SITE EMERGENCY
									GENERAL EMERGENCY

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	De	scription				De	scription		
6.1-U									
Valid, une reading to OPERATI ALL	xpected incr 1000 mrem NG CONDIT	ease of AN /hr (except ⁻ ION:	IY in-plant A TIP room).	λRΜ					UNUSUAL EVENT
6.1-A1					6.1-A2				
Valid, une reading to Personne OPERATI ALL	xpected incr 1000 mrem I required in NG CONDIT	ease of AN /hr (except AND the affecter TON:	IY in-plant <i>I</i> TIP room). d area(s).	ARM	Control Ro 15 mrem/h OPERATI ALL	oom radiatio hr. NG CONDIT	n levels gr	eater than	ALERT
									SITE EMERGENCY
									GENERAL EMERGENCY

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CONTROL ROOM EVACUATION	TURBINE FAILURE	
Description	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	6.3-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	Turbine failure resulting in visible structural damage to or visible penetration of ANY of the following structures from missles: •Reactor 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

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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							

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			FIRE	/ EX	PLOS	ON				
	De	scription				De	scription			
6.4-U1			TABLE		6.4-U2					
Confirmeo Table 6.4-	l fire in ANY U1	plant area AND	listed in		Unanticipa resulting ir structure c	ated explosion n visible dam pr equipment	on within th nage to AN t.	e protected Y permaner	area nt	UNUSUA
NOT extin	guished with	nin 15 minu	tes.							
OPERATI ALL	NG CONDIT	TON:			OPERATI ALL	NG CONDIT	ION:			EVENT
6.4-A			TABLE							
Fire or exp Table 6.4- Fire or exp permanen plant area OPERATI ALL	olosion in AN A affecting s olosion caus t structure o listed in Tal NG CONDIT	NY plant are safety syste OR ing visible f safety system ole 6.4-A.	ea listed in em performa damage to stems in AN	ance Y						ALERT
			Parameteristi							SITE EMERGENCY
										GENERAL EMERGENCY

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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

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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

<u>NOTES</u>

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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							

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FLAMMABLE GASES	
Description	
 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 	UNUSUAL EVENT
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	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

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SECI	JRITY	
Description	Description	
6.7-U		
1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.		UNUS
2. A credible Browns Ferry threat notification OR		UAL
 A validated notification from NRC providing information of an aircraft threat. 		EVEN
OPERATING CONDITION: ALL		=
6.7-A		
1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the Security Shift Supervisor.		AL
OR		ц.
2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.		۲T
OPERATING CONDITION: ALL		
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		
 A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. OR 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. 		GENERAL EMERGENCY
OPERATING CONDITION: ALL		

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VEHICLE CRASH	
Description	
6.8-U	
Vehicle crash (for example; aircraft or barge) into plant structures or systems within the protected area boundary.	UNUSUAL E
OPERATING CONDITION: ALL	/ENT
6.8-A	
Vehicle crash (for example; aircraft or barge) into ANY plant vital area.	A
	_ERT
OPERATING CONDITION: ALL	
	SITE EMERGENCY
	GENERAL EMERGENCY

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SPENT FUEL STORAGE	
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 ISESI. 	UNUSUAL EV
OPERATING CONDITION: ALL	EN T
	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

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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	A
Assessment by Unit One and Two Control Room personnel that an earthquake has occurred	Ē
Assessment by onit one and two control Room personnel that an eartiquake has been ed.	2 T
OPERATING CONDITION:	
	SITE EMERGENCY
	GENERAL EMERGENCY

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EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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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

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EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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CURVES/TABLES:

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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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

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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, <u>and</u> OPS F4)	Digital Microwave			
Sound Power Phones	NRC Emergency Telecommunication System			
Nextel Communication System	Cellular Phones (If Available)			
	Health Physics Radio Network			

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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LOSS OF COMMUNICATION	
8.2-U TABLE TABLE	
Unplanned loss of onsite communication listed in Table 8.2-U that defeats the Plant Operations Staff's ability to perform routine operations OR	USUND
Unplanned loss of ALL off-site communication listed in Table 8.2-U.	ALE
OPERATING CONDITOIN: ALL	VENT
	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 SAFETT FUNCTIONS	
eactor Power	
eactor Pressure	
eactor Level	
ubcriticality	
rywell Temperature	
rywell Pressure	
uppression Chamber Pressure	
uppression Pool Temperature	
uppression Pool Level	

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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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

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- **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

EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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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.	UNUSU/
OR	Ĕ
Any loss or any potential loss of containment.	EVEN
OPERATING CONDITION: ALL	F
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:	GENERAL EMERGENCY

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EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX

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CAUTIONS				
CAUTION #1				
 An RPV water IVI Minimum Indicate 	instrument may ed LvI associated	be used to determ I with the highest r	ine or trend IvI only when it nax DW or SC run temp	reads above the
 If DW temps or S associated instru 	C area temps (T ment may be un	able 6), as applica reliable due to boil	ble, are outside the safe req ing in the run	jion of Curve 8, the
		MINIMUM	MAX DW RUN TEMP	MAX SC
INSTRUMENT	RANGE	INDICATED	(FROM XR-64-50	
			UK 11-04-52AD)	(FROM TABLE 0)
		on scale	N/A	below 150
	Emorgonau	-145	N/A	151 to 200
LI-3-58A/B	-155 to +60	-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53				
LI-3-60		on scale	N/A	below 150
11.3.206	Normal	+5	N/A	151 to 200
	0 to +60	+15	N/A	201 to 250
LI-3-203		+20	N/A	201 10 300
LI-3-208A, B, C, D		+30	N/A	301 10 350
LI-3-52	Post	_		
LI-3-62A	Accident -268 to +32	on scale	N/A	N/A
		+10	Below 100	N/A
		+15	100 to 150	N/A
	Shutdown	+20	151 to 200	N/A
LI-3-55	Floodup	+30	201 to 250	N/A
	0 to +500	+40	251 to 300	N/A
		+50	301 to 350	N/A



Table 6 Secondary Cntmt Instrument Runs					
INSTRUMENT	INSTRUMENT SC TEMP ELEMENTS AND LOCATIONS				
	El 621 (74-95F)	El 593 (74-95C and D)	El 565 (69-835A thru D)	RWCU HXRM (69-29F, G, H)	
LI-3-58A	°F	۴F	N/A	°F	
LI-3-58B	۴F	°F	N/A	N/A	
LI-3-53	۴F	۴F	N/A	°F	
LI-3-60	°F	°F	N/A	N/A	
LI-3-206	°F	°F	N/A	°F	
LI-3-253	۴F	°F	N/A	N/A	
LI-3-52	۴F	°F	°F	N/A	
LI-3-62A	°F	°F	°F	N/A	
LI-3-55	°F	°F	N/A	N/A	
LI-3-208A, B	°F	°F	N/A	°F	
LI-3-208C, D	۴F	°F	N/A	N/A	

SRM Instrumentation 3.3.1.2

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION		REQUIRED ACTION	COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1	Restore required SRMs to OPERABLE status.	4 hours
 B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below. 	B.1	Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours
			(a susting of a line of a

SRM Instrumentation 3.3.1.2

ACTIONS (continued)			
CONDITION		REQUIRED ACTION	COMPLETION TIME
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1	Fully insert all insertable control rods.	1 hour
	AND		
	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

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Refer to Table 3.3.1.2-1 to determine which SRs apply for each applicable MODE or other specified conditions.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2.2	 NOTES	12 hours
SR 3.3.1.2.3	Perform CHANNEL CHECK.	24 hours
		(continued)

SRM Instrumentation 3.3.1.2

SURVEILLANCE REQUIREMENTS (continued)

4

	FREQUENCY	
SR 3.3.1.2.4	NOTENOTE Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.	
	Verify count rate is ≥ 3.0 cps with a signal to noise ratio $\ge 3:1$.	12 hours during CORE ALTERATIONS
		AND
		24 hours
SR 3.3.1.2.5	Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	7 days
SR 3.3.1.2.6	NOTENOTE Not required to be performed until 12 hours after IRMs on Range 2 or below.	
	Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	31 days
		(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE			
SR 3.3.1.2.7	 Neutron detectors are excluded. Not required to be performed until 12 hours after IRMs on Range 2 or below. 			
	Perform CHANNEL CALIBRATION.	92 days		

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2 ^(a)	3	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	5	2(b)(c)	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7

Table 3.3.1.2-1 (page 1 of 1) Source Range Monitor Instrumentation

(a) With IRMs on Range 2 or below.

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(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

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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

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

RPS Instrumentation 3.3.1.1

ACTIONS (continued)

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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

RPS Instrumentation 3.3.1.1

ACTIONS (COnunded)

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	CONDITION		REQUIRED ACTION	COMPLETION TIME
Ι.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	l.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	NOTENOTE-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 \leq 2% RTP while operating at \geq 25% RTP.	7 days
SR 3.3.1.1.3	NOTENOTE 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

RPS Instrumentation 3.3.1.1

SURVEILLANCE REQUIREMENTS (continued)

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	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	NOTENOTE 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	 Neutron detectors are excluded. 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

RPS Instrumentation 3.3.1.1

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	NOTENOTENOTENOTENOTE	
	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. 	184 days
SR 3.3.1.1.17	Verify OPRM is not bypassed when APRM Simulated Thermal Power is $\ge 25\%$ and recirculation drive flow is < 60% of rated recirculation drive flow.	24 months

RPS Instrumentation 3.3.1.1

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		FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
B							
-	0			•			
				6	•		
	0		۲	0			
			æ	۲	•		
2.	Av	erage Power Range Monitors					
	a.	Neutron Flux - High, Setdown	2	3(p)	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	
	b.	Flow Biased Simulated Thermal Power - High	1	3(p)	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	
	c.	Neutron Flux - High	1	3(p)	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	(continued)

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

(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.

BFN-UNIT 1

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Amendment No. 236, 262, 269 March 06, 2007

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RPS Instrumentation 3.3.1.1

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(p)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(p)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
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Table 3.3.1.1-1 (page 2 of 3) Reactor Protection System Instrumentation

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Amendment No. 269 234, 262, 259, 257,258, 266 March 06, 2007

RPS Instrumentation 3.3.1.1

Table 3.3.1.1-1 (page 3 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
		۲	•		
		۲			
			•		
			0		
	()	0	0		
		0	•		
		0	0		

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Amendment No. 234, 257 September 14, 2006

Primary Containment Isolation Instrumentation 3.3.6.1

3.3 INSTRUMENTATION

- 3.3.6.1 Primary Containment Isolation Instrumentation
- LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

Primary Containment Isolation Instrumentation 3.3.6.1

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	NOTE Only applicable for Function 1.d if two or more channels are inoperable.	
		Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c
			AND
			24 hours for Functions other than Functions 2.a, 2.b, 5.h, 6.b, and 6.c
	AND		
	A.2	NOTE Only applicable for Function 1.d when 15 of 16 channels are OPERABLE.	
		Place channel in trip.	30 days

Primary Containment Isolation Instrumentation 3.3.6.1

ACTIONS (continued)

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CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with isolation capability not maintained.	B.1	Restore isolation capability.	1 hour <u>OR</u> 4 hours for Function 1.d when normal ventilation is not available
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated Main Steam Line (MSL).	12 hours
	OR		
	D.2.1	Be in MODE 3.	12 hours
	<u>A</u>	ND	
	D.2.2	Be in MODE 4.	36 hours

Primary Containment Isolation Instrumentation 3.3.6.1

ACTIONS ((continued)

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CONDITION		REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1	Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1	Isolate the affected penetration flow path(s).	1 hour
 G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. <u>OR</u> Required Action and associated Completion Time for Condition F not met 	G.1 <u>AND</u> G.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours
Primary Containment Isolation Instrumentation 3.3.6.1

ACTIO	ACTIONS (continued)			
	CONDITION		REQUIRED ACTION	COMPLETION TIME
H. A A r T	As required by Required Action C.1 and referenced in Fable 3.3.6.1-1.	Н.1 <u>OR</u>	Declare standby liquid control system (SLC) inoperable.	1 hour
		H.2	Isolate the Reactor Water Cleanup System.	1 hour
I. A A r	As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	l.1 <u>OR</u>	Initiate action to restore channel to OPERABLE status.	Immediately
		1.2	Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.	Immediately

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SURVEILLANCE REQUIREMENTS

- Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation 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 isolation capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	122 days
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Primary Containment Isolation Instrumentation 3.3.6.1

		FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Mair	n Steam Line Isolation					
	a. 	Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b. 1	Main Steam Line Pressure - Low ^(C)	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	c.	Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	d. I	Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
2.	Prim	nary Containment Isolation					
	a.	Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b.	Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
3.	High Injeo Isola	n Pressure Coolant ction (HPCI) System ation					
	a.	HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b.	HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	Contraction
	c.	HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	Canan

Table 3.3.6.1-1 (page 1 of 3) Primary Containment Isolation Instrumentation

(continued)

(c) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

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Amendment No. 253, 260, 296 September 14, 2006

Primary Containment Isolation Instrumentation 3.3.6.1

		FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.	HP (co	CI System Isolation ntinued)					
	d.	HPCI Steam Line Space HPCI Pump Room Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	e.	HPCI Steam Line Space Torus Area (Exit) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	f.	HPCI Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	g.	HPCI Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
4.	Re (R0	actor Core Isolation Cooling CIC) System Isolation					
	a.	RCIC Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b.	RCIC Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	c.	RCIC Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	d.	RCIC Steam Line Space RCIC Pump Room Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	e.	RCIC Steam Line Space Torus Area (Exit) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	f.	RCIC Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	g.	RCIC Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	

Table 3.3.6.1-1 (page 2 of 3) Primary Containment Isolation Instrumentation

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		FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5.	Re: (RV	actor Water Cleanup NCU) System Isolation					
	a.	Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b.	Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	c.	Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	d.	Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	e.	Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	f.	Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	g.	SLC System Initiation	1,2	₁ (a)	Н	SR 3.3.6.1.6	NA
	h.	Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
6.	Sh Iso	utdown Cooling System lation					
	a.	Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	b.	Reactor Vessel Water Level - Low, Level 3	3,4,5	2 ^(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	
	с.	Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	

Table 3.3.6.1-1 (page 3 of 3) Primary Containment Isolation Instrumentation

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

- LCO 3.1.4 a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and
 - b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure \geq 800 psig.	200 days cumulative operation in MODE 1

(continued)

Control Rod Scram Times 3.1.4

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.4.3	Verify for each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell <u>AND</u> Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1) Control Rod Scram Times

- OPERABLE control rods with scram times not within the limits of this Table are considered "slow."
- Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

	SCRAM TIMES(a)(b) (seconds)
NOTCH POSITION	REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

- (a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.
- (b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.



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RCS Leakage Detection Instrumentation 3.4.5

3.4 REACTOR COOLANT SYSTEM (RCS)

- 3.4.5 RCS Leakage Detection Instrumentation
- LCO 3.4.5 The following RCS leakage detection instrumentation shall be OPERABLE:
 - a. Drywell floor drain sump monitoring system; and
 - b. One channel of either primary containment atmospheric particulate or atmospheric gaseous monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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CONDITION		REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	A.1	Restore drywell floor drain sump monitoring system to OPERABLE status.	24 hours
			(continued)

RCS Leakage Detection Instrumentation 3.4.5

ACTIONS (continued)

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CONDITION		REQUIRED ACTION	COMPLETION TIME
 Required primary containment atmospheres monitoring system inoperable. 	B.1	Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND		
	B.2	Restore required primary containment atmospheric monitoring system to OPERABLE status.	30 days
C. Required Action and associated Completion Time of Condition A o	C.1 r B <u>AND</u>	Be in MODE 3.	12 hours
not met.	C.2	Be in MODE 4.	36 hours
D. All required leakage detection systems inoperable.	D.1	Enter LCO 3.0.3.	Immediately

RCS Leakage Detection Instrumentation 3.4.5

SURVEILLANCE REQUIREMENTS

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	SURVEILLANCE	FREQUENCY
SR 3.4.5.1	Perform a CHANNEL CHECK of required primary containment atmospheric monitoring system instrumentation.	12 hours
SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of required primary containment atmospheric monitoring system instrumentation.	31 days
SR 3.4.5.3	Perform a CHANNEL CALIBRATION of required drywell sump flow integrator instrumentation.	184 days
SR 3.4.5.4	Perform a CHANNEL CALIBRATION of required leakage detection system instrumentation.	24 months

TR 3.4 REACTOR COOLANT SYSTEM

- TR 3.4.1 Coolant Chemistry
- LCO 3.4.1 Reactor coolant chemistry shall be maintained within the limits of Table 3.4.1-1.

APPLICABILITY: According to Table 3.4.1-1

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME	
А.	Conductivity greater than the limit of Table 3.4.1-1 Column B but \leq 10 μ mho/cm at 25°C.	A.1	Verify by administrative means that conductivity has not been > 1.0 μ mho/cm at 25°C for > 2 weeks in the past year.	Immediately	
В.	Chloride concentration greater than the limit of Table 3.4.1-1 Column B or E but ≤ 0.5 ppm.	B.1	Verify by administrative means that chloride concentration has not been > 0.2 ppm for > 2 weeks in the past year.	Immediately	
C.	pH not within limits of Table 3.4.1-1 Column A, B, and E.	C.1	Restore pH to within limits.	24 hours	
	(continued)				

Coolant Chemistry TR 3.4.1

ACTIONS

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CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	Required Action and associated Completion Time of Conditions A, B, or C not met.	D.1 <u>AND</u>	Initiate an orderly shutdown.	Immediately
	<u>OR</u>	D.2	Be in MODE 4.	As rapidly as cooldown rate permits
	Conductivity > 10 μmho/cm at 25°C.			
	<u>OR</u>			
	Chloride concentration > 0.5 ppm.			
	<u>OR</u>			
	Conductivity or chloride concentration limits of Table 3.4.1-1 Column A exceeded.			
E.	Coolant chemistry limits of Table 3.4.1-1 Column C, D, or E exceeded.	E.1	Initiate action to restore coolant chemistry within limits.	Immediately

TECHNICAL SUR	/EILLANCE REQUIREMENTS		
	SURVEILLANCE	FREQUENCY	
TSR 3.4.1.1	NOTENOTENOTENOTENOTENOTENOTE	Continuously <u>OR</u>	
	Monitor reactor coolant conductivity.	4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5	
		<u>OR</u>	
		8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5	
TSR 3.4.1.2	NOTE	7 days	
	Not required when there is no fuel in the reactor vessel.	AND	
	Check the continuous conductivity monitor with an in-line flow cell.	24 hours whenever the reactor coolant conductivity is >1.0 μmho/cm at 25°C	
		(continued)	

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Coolant Chemistry TR 3.4.1

TECHNICAL SURVEILLANCE REQUIREMENTS

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	SURVEILLANCE		
TSR 3.4.1.3	Verify reactor coolant conductivity and chloride concentration within limits of Table 3.4.1-1 Column A.	Once during startup prior to pressurizing the reactor above atmospheric pressure	
TSR 3.4.1.4	NOTENOTE Only required when the reactor is operating in MODES 1 or 2.	96 hours <u>AND</u>	
	Verify chloride ion content and pH within the limits of Table 3.4.1-1.	8 hours whenever the reactor conductivity is >1.0 μmho/cm at 25°C (not required for Column E.)	
TSR 3.4.1.5	NOTE Only required when the reactor is not pressurized with fuel in the reactor vessel.		
	Verify conductivity, chloride ion content, and pH within the limits of Table 3.4.1-1.	96 hours	

Coolant Chemistry TR 3.4.1

Table 3.4.1-1 Coolant Chemistry Limits⁽¹⁾

CHEMISTRY PARAMETERS	COLUMN A APPLICABLE CONDITION Prior To Startup And At Steaming Rates < 100,000 lb/hr	COLUMN B APPLICABLE CONDITION Steaming Rates > 100,000 lb/hr	COLUMN C APPLICABLE CONDITION Reactor Not Pressurized With Fuel In Reactor Vessel, Except During Startup Condition	COLUMN D ⁽²⁾ APPLICABLE CONDITION Noble Metal Chemical Application and Subsequent Reactor Coolant Cleanup	COLUMN E ⁽³⁾ APPLICABLE CONDITION Operation of HWC Following Noble Metal Chemical Application
CHLORIDE (ppm)	≤ 0.1	≤ 0.2	≤ 0.5	≤ 0.1	≤ 0.2
CONDUCTIVITY (μmho/cm at 25°C)	≤ 2.0	≤ 1.0	≤ 10.0	≤ 20.0	≤ 2.0
pН	5.6-8.6	5.6-8.6	5.3-8.6	4.3-9.9	5.6-8.8

⁽¹⁾ When there is no fuel in the reactor vessel, Technical Requirement reactor coolant chemistry limits do not apply.

(2) During the Noble Metal Chemical Application and subsequent reactor coolant cleanup, CONDITIONS A, B, C, and D (including Required Actions and Completion Times) do not apply.

⁽³⁾ During operation of HWC following the Noble Metal Chemical Application, CONDITION A (including Required Action and Completion Time) does not apply.