U.S. Nuclear Regulatory Commission

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.

____

Applicant's Signature

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

Test 15-01 NKC Exam KEY
Class SRO Exam
Instructor Q# 76-100

LXR•TEST™
Response Form
LXR-20020
Side 1

Name:
Signature:
Date:

READ CAREFULLY!
OK NOT OK

- **Use black ink only.**
- Mark responses darkly and fill completely.
- Erase unwanted marks clearly.
- Do NOT make any stray marks on the page
- No credit will be given for improper marks
- If Side 2 is used, fill in ID on both sides

DOCKET NUMBER

version

C1 C2 C3 C4 V

SECTION 1

Q	Answer	Q	Answer
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8	B	58	D
9	B	59	D
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13	A	63	A
14	C	64	B
15	C	65	B
16	A	66	D
17	D	67	D
18	A	68	B
19	C	69	A
20	D	70	D
21	B	71	A
22	C	72	D
23	B	73	B
24	C	74	B
25	B	75	D
26	A	76	C
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28	C	78	A
29	B	79	B
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31	A	81	C
32	A	82	C
33	C	83	C
34	D	84	B
35	C	85	C
36	D	86	B
37	C	87	B
38	C	88	B
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41	A	91	A
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43	A	93	B
44	A	94	C
45	D	95	D
46	D	96	B
47	A	97	A
48	C	98	C
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50	A	100	A

SECTION 2

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SRO Exam Q# 76-100

QUESTION ID	NUREG K/A CROSS REF	QUESTION ID	NUREG K/A CROSS REF	QUESTION ID	NUREG K/A CROSS REF	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
9	295021 AK1.01	34	212000 A2.21	59	230000 K2.02	84	295032 EA2.02
10	295023 AK1.03	35	215003 K1.02	60	233000 A4.05	85	295010 G2.4.45
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
15	295030 EA2.04	40	217000 K1.04	65	290003 A3.01	90	223002 A2.07
16	295031 EA1.05	41	218000 K5.01	66	2.1.19	91	201001 A2.08
17	295037 EA1.08	42	223002 G2.1.7	67	2.1.6	92	215001 G2.1.20
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

Q 76

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

[Reference Provided]

A. (1) 1400
(2)Site Licensing

B. (1) 1800
(2) Site Licensing

C. (1) 1400
(2) Operations

D. (1) 1800
(2) Operations

Q 77

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)___.

[Reference Provided]

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

Q 78

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.

[Reference 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

Q 79

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 Minimum required action in accordance with 1-EOI-1, RPV Control and 1-EOI-2, Primary Containment Control?

[Reference Provided]

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

Q 80

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

Q 81

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%.

Q 82

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?

[Reference Provided]

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Q 83

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

Q 84

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.

[Reference Provided]

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

Q 85

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 86

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

Q 87

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.

Q 88

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?

[Reference Provided]

A. Required Action A.1 OR 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.

Q 89

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

Q 90

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 earliest 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)

[Reference Provided]

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

Q 91

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.

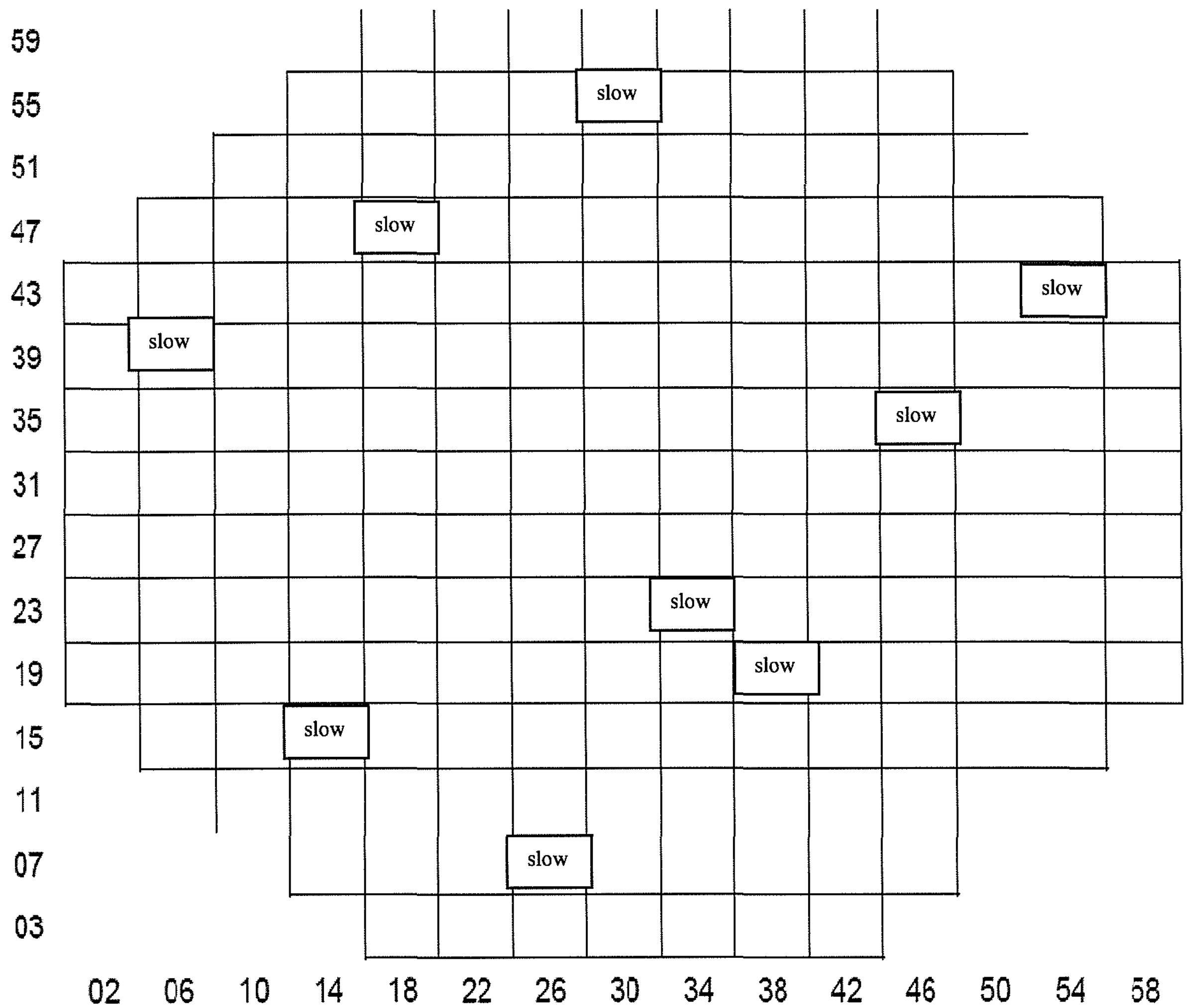
[Reference and Illustration Provided]

A. (1) is
(2) is Not

B. (1) is
(2) is

C. (1) is Not
(2) is Not

D. (1) is Not
(2) is


59
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Q 92

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

Q 93

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 **required** by Tech Specs to be in MODE 3?

(Assume No Operator actions)

[Reference Provided]

A. 7-1 @ 20:00

B. 7-1 @ 23:00

C. 7-2 @ 10:00

D. 7-2 @ 22:00

Q 94

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?

[Reference Provided]

A. Condition A.1 only

B. Condition B.1 only

C. Condition C.1 only

D. Condition D.1

Q 95

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

Q 96

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

Q 97

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

Q 98

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

- Stack Noble Gas WRGERM: 7.1 X $10^9 \mu Ci/sec$
- 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?

[Reference Provided]

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

Q 99

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.

Q 100

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

NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0010** **Page 1 of 100**
		Quality Related ☐ Yes ☑ No
		Validation Date 07-28-2011 Review Frequency 3 years Validated By Henry Lee
		Effective Date 03-19-2014

Responsible Peer Team/Working Group: Licensing

Approved by: Ed Schrull — 03-18-2014

Corporate Functional Area Manager — 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.

> **NOTE**
>
> [redacted]

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.

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.

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.

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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:

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

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.

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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)

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.

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

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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.

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

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

   b. The actuation was invalid and

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

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

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

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

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

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

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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.

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

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

    b. Remove residual heat;

    c. Control the release of radioactive material; or

    d. Mitigate the consequences of an accident.

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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.

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

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

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

       (1) Excessive bird impaction events;

       (2) Onsite plant or animal disease outbreaks;

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

       (4) Fish kills near the plant site;

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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.

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

NOTES

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

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

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

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

CURVES/TABLES:

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

WATER LEVEL

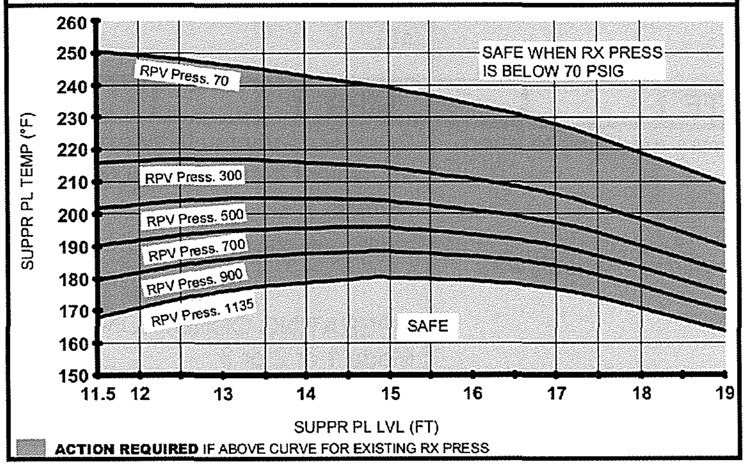
Description	Description	
1.1-U1 **NOTE** Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION: Mode 5	**1.1-U2** Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION ALL	**UNUSUAL EVENT**
1.1-A1 **NOTE** Uncontrolled 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: Mode 1 or 2 or 3	**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**

NOTES

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

CURVES/TABLES:

CURVE 1.2-G
HEAT CAPACITY TEMP LIMIT

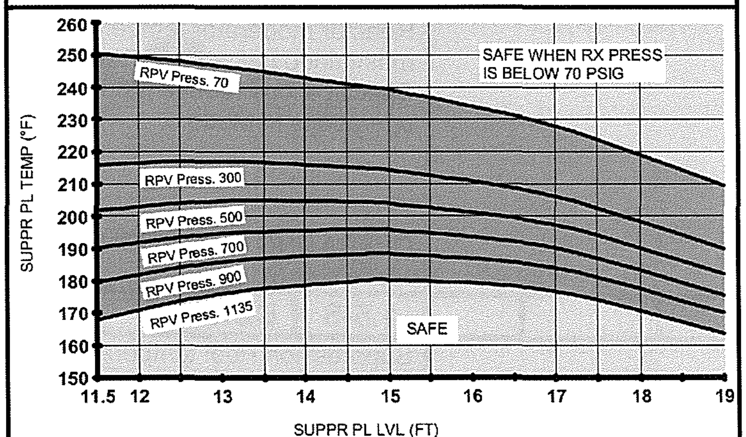


	SCRAM FAILURE	REACTOR COOLANT ACTIVITY
	Description	Description
UNUSUAL EVENT		**1.3-U** Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample. OPERATING CONDITION ALL
ALERT	**1.2-A** **NOTE** Failure of RPS automatic scram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

NOTES

CURVES/TABLES:

CURVE 1.5-S
HEAT CAPACITY TEMP LIMIT



ACTION REQUIRED IF ABOVE CURVE FOR EXISTING RX PRESS

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

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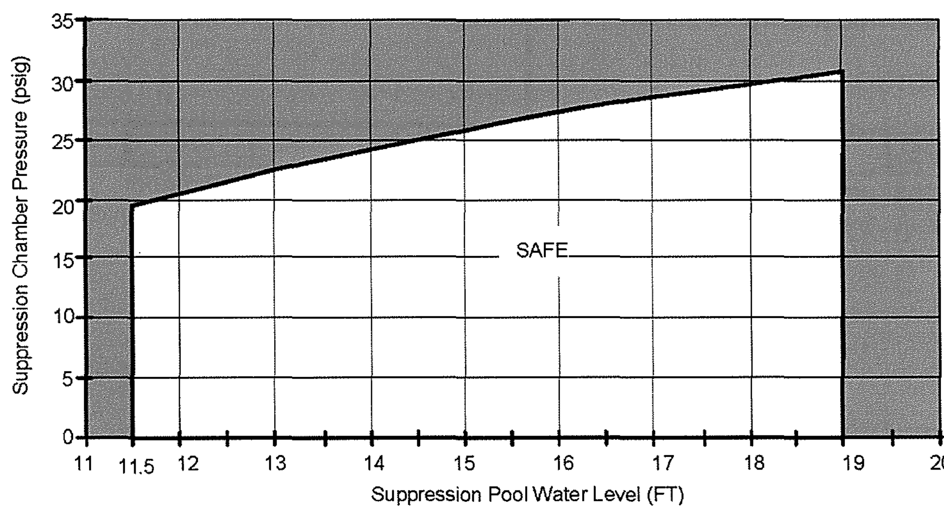
PRIMARY CONTAINMENT 2.0

NOTES

CURVES/TABLES:

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

CURVE 2.1-S
PRESS SUPPR PRESS



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

NOTES

CURVES/TABLES:

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

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

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

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

DRYWELL RADIATION

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

NOTES

CURVES/TABLES:

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

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

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SECONDARY CONTAINMENT 3.0

NOTES

CURVES/TABLES:

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

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

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

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

SECONDARY CONTAINMENT TEMPERATURE

Description

UNUSUAL EVENT

ALERT

SITE EMERGENCY

3.1-S			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

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

OPERATING CONDITION:
Mode 1 or 2 or 3

GENERAL EMERGENCY

3.1-G			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

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

AND

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

OPERATING CONDITION
Mode 1 or 2 or 3

NOTES

CURVES/TABLES:

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

AREA	RAD MONITOR	MAX SAFE VALUE MR/HR		
		UNIT 1	UNIT 2	UNIT 3
RHR West Room	90-25A	1000	1000	1000
RHR East Room	90-28A	1000	1000	1000
HPCI Room	90-24A	1000	1000	1000
CS/RCIC Room	90-26A	1000	1000	1000
Core Spray Room	90-27A	1000	1000	1000
Suppr Pool Area	90-29A	1000	1000	1000
CRD-HCU West Area	90-20A	1000	1000	1000
CRD-HCU East Area	90-21A	1000	1000	1000
TIP Drive Area	90-23A	1000	1000	1000
North RWCU System Area	90-13A	1000	1000	1000
South RWCU System Area	90-14A	1000	1000	1000
RWCU System Area	90-9A	1000	1000	1000
MG Set Area	90-4A	1000	1000	1000
Fuel Pool Area	90-1A	1000	1000	1000
Service Flr Area	90-2A	1000	1000	1000
New Fuel Storage	90-3A	1000	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 DRYWELL RADIATION	
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	≥ 196 R/HR
1-RE-90-273A	≥ 297 R/HR	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR
Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 μCi/gm Dose Equivalent Iodine 131	

SECONDARY CONTAINMENT RADIATION

Description

UNUSUAL EVENT

ALERT

3.2-A				

Any of the following high radiation alarms on Panel 9-3:

- 1, 2, or 3-RA-90-1A, Fuel Pool Floor Alarm
- 1, 2, or 3-RA-90-250A, Reactor, Turbine, Refuel Exhaust
- 1, 2, or 3-RA-90-142A, Reactor Refuel Exhaust
- 1, 2, or 3-RA-90-140A, Refueling Zone Exhaust

AND

Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred.

OPERATING CONDITION:
ALL

SITE EMERGENCY

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

GENERAL 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

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RADIOACTIVITY RELEASES 4.0

NOTES

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

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

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

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

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

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

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

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

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

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

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

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

CURVES/TABLES:

Table 4.1-U
RELEASE LIMITS FOR UNUSUAL EVENT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88 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

TYPE	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

TYPE	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 Iodine-131	Field Assessment Team	3.9 X 10^{-7} µCI /cm^{3}	1 Hour

Table 4.1-G
RELEASE LIMITS FOR GENERAL EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9 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} µCI / cm^{3}	1 Hour

GASEOUS EFFLUENT

Description

4.1-U		NOTE	TABLE	

Gaseous release exceeds ANY limit and duration in Table 4.1-U.

OPERATING CONDITION:
ALL

UNUSUAL EVENT

4.1-A		NOTE	TABLE	

Gaseous release exceeds ANY limit and duration in Table 4.1-A.

OPERATING CONDITION:
ALL

ALERT

4.1-S		NOTE	TABLE	

EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-S.
- Dose assessment indicates actual or projected dose consequences above 100 mrem TEDE or 500 mrem thyroid CDE.

OPERATING CONDITION:
ALL

SITE EMERGENCY

4.1-G		NOTE	TABLE	

EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-G.
- Dose assessment indicates actual or projected dose consequences above 1000 mrem TEDE or 5000 mrem thyroid CDE.

OPERATING CONDITION
ALL

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

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

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LOSS OF POWER
5.0

NOTES

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

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

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

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

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

CURVES/TABLES:

Table 5.1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
APPLICABLE UNIT	**APPLICABLE 4KV SHUTDOWN BOARDS**
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

LOSS OF AC POWER

Description	Description	
5.1-U | | **NOTE** | **TABLE** | **US**		**UNUSUAL EVENT**
Loss of normal and alternate supply voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes **AND** At least two Diesel Generators supplying power to unit specific 4KV shutdown boards listing in Table 5.1. OPERATING CONDITION: ALL		
5.1-A1 | | **NOTE** | **TABLE** | **US**	**5.1-A2** | | **NOTE** | **TABLE** | **US**	**ALERT**
Loss of voltage to ANY THREE unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes **AND** Only ONE source of power available to the remaining board. OPERATING CONDITION: Mode 1 or 2 or 3	Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes. OPERATING CONDITION: Mode 4 or 5 or Defueled	
5.1-S | | **NOTE** | **TABLE** | **US**		**SITE EMERGENCY**
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		
5.1-G | | **NOTE** | **TABLE** | **US**		**GENERAL EMERGENCY**
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		

NOTES

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

CURVES/TABLES:

Table 5.2-U
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

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

Table 5.2-S
CRITICAL DC POWER AND ESSENTIAL SYSTEMS

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

LOSS OF 250V DC POWER

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

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HAZARDS
6.0

NOTES

CURVES/TABLES:

RADIOLOGICAL

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

NOTES

CURVES/TABLES:

	CONTROL ROOM EVACUATION	TURBINE FAILURE
	Description	Description
UNUSUAL EVENT		**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
ALERT	**6.2-A** Control Room Abandonment from entry into 1, 2, or 3-AOI-100-2 or 0-SSI-16 for ANY Unit Control Room. OPERATING CONDITION: ALL	**6.3-A** Turbine failure resulting in visible structural damage to or visible penetration of ANY of the following structures from missles: ◆Reactor Building ◆Diesel Generator Building ◆Intake Structure ◆Control Bay OPERATING CONDITION: Mode 1 or 2
SITE EMERGENCY	**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	
GENERAL EMERGENCY		

NOTES

CURVES/TABLES:

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

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

FIRE / EXPLOSION

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

NOTES

CURVES/TABLES:

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

TOXIC GASES

Description

6.5-U			TABLE	

EITHER of the following conditions exists:

- Normal operations impeded due to access restrictions caused by toxic gas concentrations within any building or structure listed in Table 6.5/6.6.
- Confirmed report by local, county, or state officials that a large offsite toxic gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above the Permissible Exposure Limit (PEL) causing an evacuation of any site personnel.

OPERATING CONDITION:
ALL

UNUSUAL EVENT

6.5-A			TABLE	

ALL of the following conditions exist:

- Plant personnel report toxic gas within any building or structure listed in Table 6.5/6.6.
- Plant personnel report severe adverse health reactions due to toxic gas (i.e., burning eyes, throat, or dizziness), or sampling results by Fire Protection or Industrial Safety personnel indicate levels above the Permissible Exposure Limit (PEL).
- Determination by the Site Emergency Director that plant personnel would be unable to perform actions necessary to establish and maintain cold shutdown conditions while utilizing appropriate personnel protective equipment.

OPERATING CONDITION:
ALL

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

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

FLAMMABLE GASES

Description

6.6-U			TABLE	

EITHER of the following conditions exists:

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

OPERATING CONDITION:
ALL

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

NOTES

CURVES/TABLES:

SECURITY

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

NOTES

CURVES/TABLES:

VEHICLE CRASH

Description

6.8-U — UNUSUAL EVENT

Vehicle crash (for example; aircraft or barge) into plant structures or systems within the protected area boundary.

OPERATING CONDITION:
ALL

6.8-A — ALERT

Vehicle crash (for example; aircraft or barge) into ANY plant vital area.

OPERATING CONDITION:
ALL

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

SPENT FUEL STORAGE

Description

6.9-U

Damage to a loaded cask CONFINEMENT BOUNDARY from ANY of the following:

- Natural phenomena (e.g., seismic event, tornado, flood, lightning, snow/ice accumulation, etc.)
- Accident (e.g., dropped cask, tipped over cask, explosion, missile damage, fire damage, burial under debris, etc.).
- Judgement of the Site Emergency Director that the CONFINEMENT BOUNDARY damage is a degradation in the level of safety of the ISFSI.

OPERATING CONDITION:
ALL

UNUSUAL EVENT

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

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NATURAL EVENTS
7.0

NOTES

CURVES/TABLES:

EARTHQUAKE

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

NOTES

CURVES/TABLES:

TORNADO / HIGH WINDS

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

NOTES

CURVES/TABLES:

FLOOD

Description

UNUSUAL EVENT

7.3-U

Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet.

AND

Water entering permanent plant structures due to flooding.

OPERATING CONDITION:
ALL

ALERT

7.3-A

Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet.

AND

EITHER of the following conditions exists:

- Breech or failure of any water-tight structure is causing flooding of the structure
- Equipment required for safe shutdown is affected.

OPERATING CONDITION:
ALL

SITE EMERGENCY

GENERAL EMERGENCY

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EMERGENCY DIRECTOR JUDGMENT 8.0

NOTES

CURVES/TABLES:

TECHNICAL SPECIFICATIONS

Description

8.1-U

Inability to reach required shutdown condition (Mode 3 or Mode 4) within Technical Specification Limiting Conditions for Operation (LCO) limits.

OPERATING CONDITION:
Mode 1 or 2 or 3

UNUSUAL EVENT

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

Table 8.2-U LOSS OF COMMUNICATIONS	
Onsite Communications	Offsite Communication
Plant Phone System Node 1	Bell Lines
Two-Way Radio System (NSS 1, NSS 2, OPS F2, and OPS F4)	Digital Microwave
Sound Power Phones	NRC Emergency Telecommunication System
Nextel Communication System	Cellular Phones (If Available)
	Health Physics Radio Network

LOSS OF COMMUNICATION

Description	
8.2-U | | | TABLE | Unplanned loss of onsite communication listed in Table 8.2-U that defeats the Plant Operations Staff's ability to perform routine operations **OR** Unplanned loss of ALL off-site communication listed in Table 8.2-U. OPERATING CONDITOIN: ALL	UNUSUAL EVENT
	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

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

CURVES/TABLES:

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

LOSS OF ASSESSMENT CAPABILITY

Description	
8.3-U Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes **AND** Compensatory non-alarming safety system indications are available (SPDS, ICS) **AND** In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant. OPERATING CONDITION: MODE 1, or 2, or 3	**UNUSUAL EVENT**
8.3-A | NOTE Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes **AND** In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant **AND** EITHER of the following conditions exists: • Compensatory non-alarming safety system indications are NOT available (SPDS, ICS) • A significant transient is in progress. OPERATING CONDITION: MODE 1, or 2, or 3	**ALERT**
8.3-S | NOTE | TABLE Loss of most or all annunciators associated with safety systems **AND** Compensatory non-alarming safety system indications are NOT available (SPDS, ICS) **AND** Indications needed to monitor safety functions are NOT available (Refer to Table 8.3-S) **AND** A significant transient is in progress. OPERATING CONDITION: MODE 1, or 2, or 3	**SITE EMERGENCY**
	GENERAL EMERGENCY

NOTES

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

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

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

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

CURVES/TABLES:

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

OTHER

Description	
8.4-U **NOTE** **TABLE** Events are in process or have occurred which indicate a potential degradation in the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Refer to Table 8.4-U for examples. **OR** Any loss or any potential loss of containment. OPERATING CONDITION: ALL	**UNUSUAL EVENT**
8.4-A **NOTE** Events are in process or have occurred which involve an actual or potential substantial degradation in the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. **OR** Any loss or potential loss of fuel cladding or RCS pressure boundary. OPERATING CONDITION: ALL	**ALERT**
8.4-S **NOTE** Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts (1) toward site personnel or equipment that could lead to the likely failure thereof or, (2) prevent effective access to equipment needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary. **OR** Any loss or potential loss of both fuel cladding and RCS pressure boundary. **OR** Potential loss of either fuel cladding or RCS pressure boundary and loss of any additional barrier. OPERATING CONDITION: ALL	**SITE EMERGENCY**
8.4-G **NOTE** Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. **OR** Loss of any two barriers and potential loss of third barrier. OPERATING CONDITION: ALL	**GENERAL EMERGENCY**

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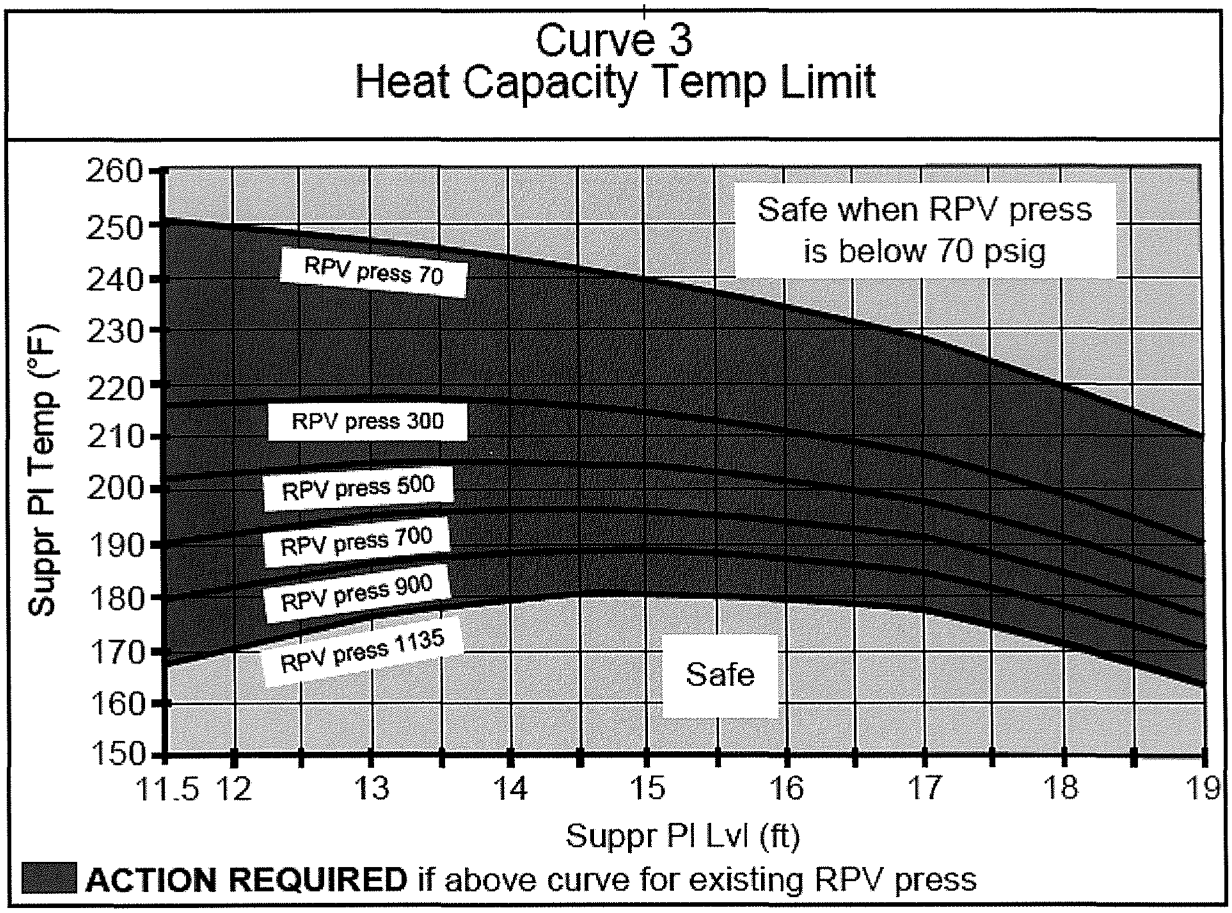


Curve 3
Heat Capacity Temp Limit
Safe when RPV press
is below 70 psig
RPV press 70
RPV press 300
RPV press 500
RPV press 700
RPV press 900
RPV press 1135
Safe
Suppr Pl Temp (°F)
260
250
240
230
220
210
200
190
180
170
160
150
11.5
12
13
14
15
16
17
18
19
Suppr Pl Lvl (ft)
ACTION REQUIRED if above curve for existing RPV press

CAUTIONS

CAUTION # 1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

Curve 8
RPV Saturation Temp

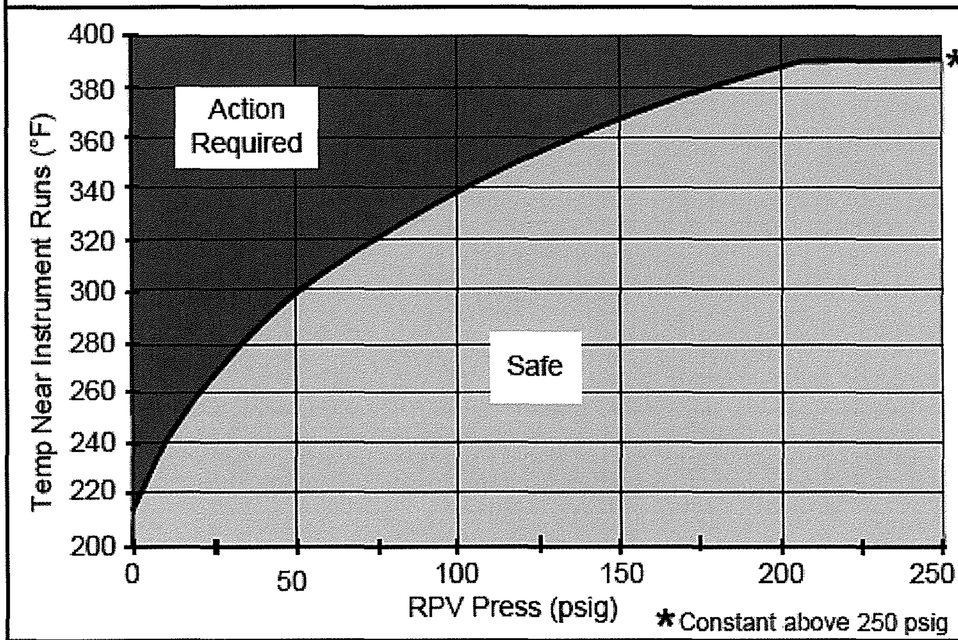


Table 6
Secondary Cntmt Instrument Runs

INSTRUMENT	SC TEMP ELEMENTS AND LOCATIONS			
	EI 621 (74-95F)	EI 593 (74-95C and D)	EI 565 (69-835A thru D)	RWCU HXRM (69-29F, G, H)
LI-3-58A	°F	°F	N/A	°F
LI-3-58B	°F	°F	N/A	N/A
LI-3-53	°F	°F	N/A	°F
LI-3-60	°F	°F	N/A	N/A
LI-3-206	°F	°F	N/A	°F
LI-3-253	°F	°F	N/A	N/A
LI-3-52	°F	°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	°F	°F	N/A	N/A

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

(continued)

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

---NOTE---
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----------------------- 1. Only required to be met during CORE ALTERATIONS. 2. One SRM may be used to satisfy more than one of the following. --- Verify an OPERABLE SRM detector is located in: a. The fueled region; b. The core quadrant where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region; and c. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region.	12 hours
SR 3.3.1.2.3	Perform CHANNEL CHECK.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.4	-------------------------NOTE------------------------ 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 ≥ 3:1.	 12 hours during CORE ALTERATIONS <u>AND</u> 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	-------------------------NOTE------------------------ 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		FREQUENCY
SR 3.3.1.2.7	-------------------------NOTES------------------------ 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. -- Perform CHANNEL CALIBRATION.	92 days

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

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

(a) With IRMs on Range 2 or below.

(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.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

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

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

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

(continued)

ACTIONS (continued)

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

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

-----NOTES-----

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

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
[redacted]					
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]

(continued)

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

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

[redacted]

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	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(b)	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
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]					
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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
[redacted]					
[redacted] [redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

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.

ACTIONS

--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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour OR 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
	AND	
	D.2.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

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 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	H.1	Declare standby liquid control system (SLC) inoperable.	1 hour
	OR		
	H.2	Isolate the Reactor Water Cleanup System.	1 hour
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	I.1	Initiate action to restore channel to OPERABLE status.	Immediately
	OR		
	I.2	Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. 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

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main 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	[redacted]
b. 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	[redacted]
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	[redacted]
d. 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	[redacted]
2. Primary 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	[redacted]
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	[redacted]
3. High Pressure Coolant Injection (HPCI) System Isolation					
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	[redacted]
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	[redacted]
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	[redacted]

(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.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
4. Reactor Core Isolation Cooling (RCIC) 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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]

(continued)

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) 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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
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	[redacted]
g. SLC System Initiation	1,2	1(a)	H	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	[redacted]
6. Shutdown Cooling System Isolation					
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	[redacted]
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	[redacted]
c. 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	[redacted]

[redacted]

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

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

-----NOTE-----
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

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 ≥ 800 psig.	200 days cumulative operation in MODE 1

(continued)

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

---NOTES---

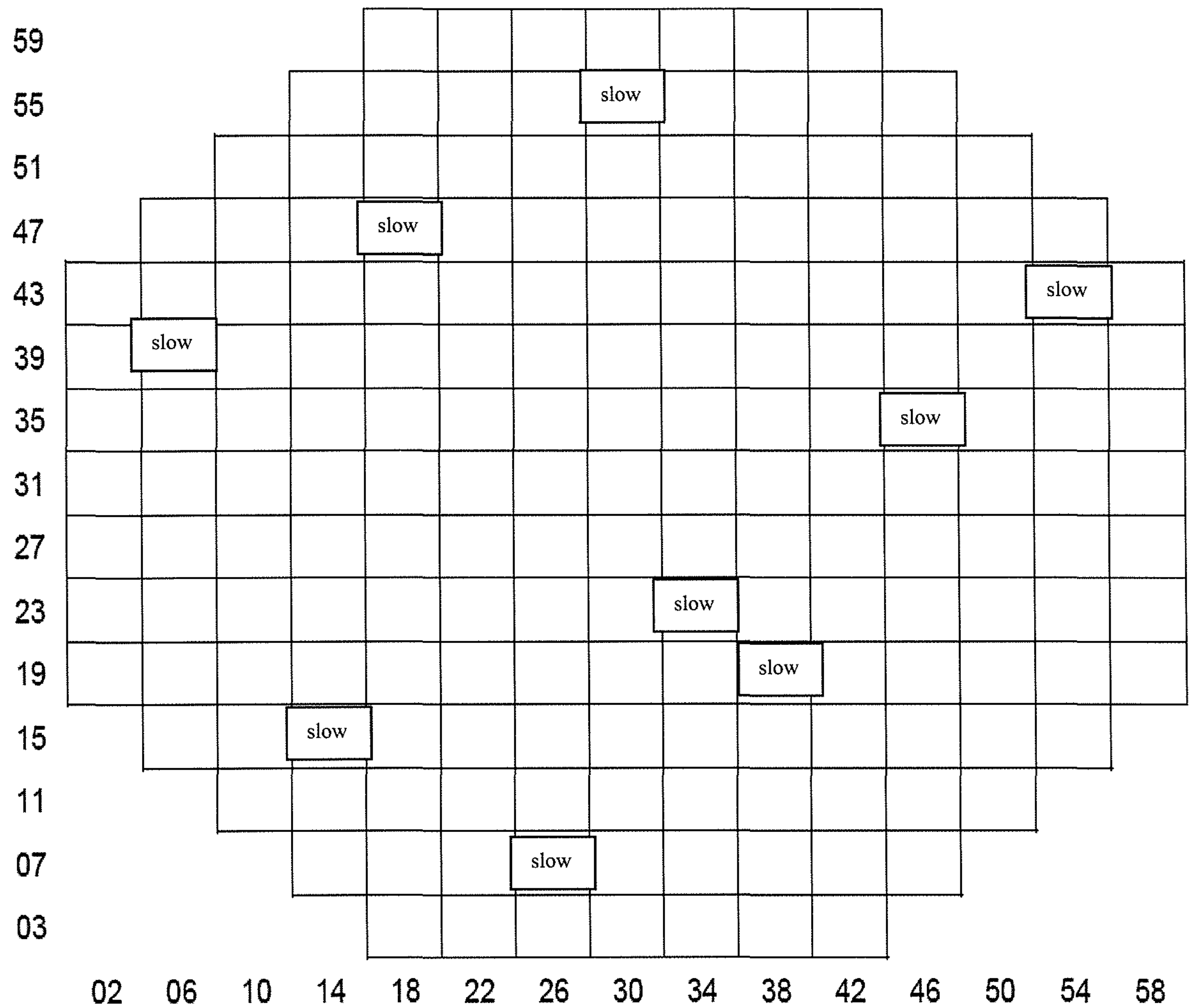
1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. 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."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 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.


59
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slow

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

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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required primary containment atmospheric 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 or B not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours
D. All required leakage detection systems inoperable.	D.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

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
A. Conductivity greater than the limit of Table 3.4.1-1 Column B but ≤ 10 μmho/cm at 25°C.	A.1 Verify by administrative means that conductivity has not been > 1.0 μmho/cm at 25°C for > 2 weeks in the past year.	Immediately
B. 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)

ACTIONS

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

---NOTE---
When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at Technical Requirement frequency is not required.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1	---NOTE--- Not required when there is no fuel in the reactor vessel. --- Monitor reactor coolant conductivity.	Continuously <u>OR</u> 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--- Not required when there is no fuel in the reactor vessel. --- Check the continuous conductivity monitor with an in-line flow cell.	7 days <u>AND</u> 24 hours whenever the reactor coolant conductivity is >1.0 μmho/cm at 25°C

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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	------------------------NOTE------------------------ Only required when the reactor is operating in MODES 1 or 2. -- Verify chloride ion content and pH within the limits of Table 3.4.1-1.	96 hours <u>AND</u> 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

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
pH	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.