U.S. Nuclear Regulatory Commission
2013 HNP NRC
Site-Specific SRO Written Examination

Applicant Information

Name:

Date:	Facility/Unit: Harris Nuclear Plant
Region: I ☐ II ☒ III ☐ IV ☐	Reactor Type: W ☒ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

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

Applicant Certification

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

____
Applicant's Signature

Results

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

Rec'd 10/2/13

#	ID	Points	Type	Answers 0
1	2013 NRC RO 1	1.00	MCS	C
2	2013 NRC RO 2	1.00	MCS	A
3	2013 NRC RO 3	1.00	MCS	A
4	2013 NRC RO 4	1.00	MCS	A
5	2013 NRC RO 5	1.00	MCS	D
6	2013 NRC RO 6	1.00	MCS	A
7	2013 NRC RO 7	1.00	MCS	B
8	2013 NRC RO 8	1.00	MCS	C
9	2013 NRC RO 9	1.00	MCS	B
10	2013 NRC RO 10	1.00	MCS	A
11	2013 NRC RO 11	1.00	MCS	A
12	2013 NRC RO 12	1.00	MCS	D
13	2013 NRC RO 13	1.00	MCS	C
14	2013 NRC RO 14	1.00	MCS	C
15	2013 NRC RO 15	1.00	MCS	C
16	2013 NRC RO 16	1.00	MCS	A
17	2013 NRC RO 17	1.00	MCS	C
18	2013 NRC RO 18	1.00	MCS	B
19	2013 NRC RO 19	1.00	MCS	D
20	2013 NRC RO 20	1.00	MCS	A
21	2013 NRC RO 21	1.00	MCS	C
22	2013 NRC RO 22	1.00	MCS	C
23	2013 NRC RO 23	1.00	MCS	A
24	2013 NRC RO 24	1.00	MCS	B
25	2013 NRC RO 25	1.00	MCS	D
26	2013 NRC RO 26	1.00	MCS	~~A~~ C
27	2013 NRC RO 27	1.00	MCS	D
28	2013 NRC RO 28	1.00	MCS	C
29	2013 NRC RO 29	1.00	MCS	A
30	2013 NRC RO 30	1.00	MCS	B
31	2013 NRC RO 31	1.00	MCS	C
32	2013 NRC RO 32	1.00	MCS	B
33	2013 NRC RO 33	1.00	MCS	C
34	2013 NRC RO 34	1.00	MCS	B
35	2013 NRC RO 35	1.00	MCS	A
36	2013 NRC RO 36	1.00	MCS	D
37	2013 NRC RO 37	1.00	MCS	A
38	2013 NRC RO 38	1.00	MCS	C
39	2013 NRC RO 39	1.00	MCS	B
40	2013 NRC RO 40	1.00	MCS	C
41	2013 NRC RO 41	1.00	MCS	C
42	2013 NRC RO 42	1.00	MCS	C
43	2013 NRC RO 43	1.00	MCS	A
44	2013 NRC RO 44	1.00	MCS	B
45	2013 NRC RO 45	1.00	MCS	B
46	2013 NRC RO 46	1.00	MCS	C
47	2013 NRC RO 47	1.00	MCS	D
48	2013 NRC RO 48	1.00	MCS	C

#	ID	Points	Type	Answers
				0
49	2013 NRC RO 49	1.00	MCS	B
50	2013 NRC RO 50	1.00	MCS	B
51	2013 NRC RO 51	1.00	MCS	C
52	2013 NRC RO 52	1.00	MCS	A
53	2013 NRC RO 53	1.00	MCS	A
54	2013 NRC RO 54	1.00	MCS	B
55	2013 NRC RO 55	1.00	MCS	A
56	2013 NRC RO 56	1.00	MCS	A
57	2013 NRC RO 57	1.00	MCS	B
58	2013 NRC RO 58	1.00	MCS	B
59	2013 NRC RO 59	1.00	MCS	C
60	2013 NRC RO 60	1.00	MCS	A
61	2013 NRC RO 61	1.00	MCS	B
62	2013 NRC RO 62	1.00	MCS	A
63	2013 NRC RO 63	1.00	MCS	A
64	2013 NRC RO 64	1.00	MCS	D
65	2013 NRC RO 65	1.00	MCS	B
66	2013 NRC RO 66	1.00	MCS	D
67	2013 NRC RO 67	1.00	MCS	C
68	2013 NRC RO 68	1.00	MCS	B
69	2013 NRC RO 69	1.00	MCS	B/D
70	2013 NRC RO 70	1.00	MCS	C
71	2013 NRC RO 71	1.00	MCS	B
72	2013 NRC RO 72	1.00	MCS	C
73	2013 NRC RO 73	1.00	MCS	D
74	2013 NRC RO 74	1.00	MCS	B
75	2013 NRC RO 75	1.00	MCS	D
76	2013 NRC SRO 1	1.00	MCS	C
77	2013 NRC SRO 2	1.00	MCS	A
78	2013 NRC SRO 3	1.00	MCS	A
79	2013 NRC SRO 4	1.00	MCS	C
80	2013 NRC SRO 5	1.00	MCS	C
81	2013 NRC SRO 6	1.00	MCS	A
82	2013 NRC SRO 7	1.00	MCS	B
83	2013 NRC SRO 8	1.00	MCS	C
84	2013 NRC SRO 9	1.00	MCS	A
85	2013 NRC SRO 10	1.00	MCS	~~C~~ D
86	2013 NRC SRO 11	1.00	MCS	D
87	2013 NRC SRO 12	1.00	MCS	B
88	2013 NRC SRO 13	1.00	MCS	A
89	2013 NRC SRO 14	1.00	MCS	A
90	2013 NRC SRO 15	1.00	MCS	B
91	2013 NRC SRO 16	1.00	MCS	D
92	2013 NRC SRO 17	1.00	MCS	D
93	2013 NRC SRO 18	1.00	MCS	A
94	2013 NRC SRO 19	1.00	MCS	C
95	2013 NRC SRO 20	1.00	MCS	A
96	2013 NRC SRO 21	1.00	MCS	B

				Answers
#	ID	Points	Type	0
97	2013 NRC SRO 22	1.00	MCS	A
98	2013 NRC SRO 23	1.00	MCS	D
99	2013 NRC SRO 24	1.00	MCS	A
100	2013 NRC SRO 25	1.00	MCS	C
SECTION 1 (100 items)		**100.00**		

1. Given the following plant conditions:
   - A Reactor Trip occurs due to lowering RCS Pressure
   - 'A' Reactor Trip breaker is OPEN
   - 'B' Reactor Trip breaker is CLOSED
   - The crew is implementing E-0, Reactor Trip or Safety Injection to stabilize the plant when RCS pressure reaches the low RCS pressure safety injection setpoint

   Which ONE of the following completes the statements below?

   When directed to reset safety injection in E-0, the operator must wait a MINIMUM of __(1)__ seconds after the SI signal actuation.

   Based on the current conditions, safety injection reset AND automatic block can be performed on __(2)__ .

   A. (1) 150

      (2) 'A' Train ONLY

   B. (1) 150

      (2) 'A' AND 'B' Train

   C. (1) 60

      (2) 'A' Train ONLY

   D. (1) 60

      (2) 'A' AND 'B' Train

2. Given the following plant conditions:
   - A LOCA occurs through a stuck open PZR Safety Valve
   - The crew transitions to ES-1.2, Post LOCA Cooldown and Depressurization

   WHICH ONE of the following completes BOTH of the statements below?

   In accordance with ES-1.2, Pressurizer heaters ___(1)___.

   The basis for this restriction on heater operation is that ___(2)___.

   A. (1) are NOT allowed to be energized until a TSC evaluation is provided

      (2) PZR level instruments may have measurement errors

   B. (1) are NOT allowed to be energized until a TSC evaluation is provided

      (2) heater elements may have been previously damaged

   C. (1) CAN be energized without a TSC evaluation if PZR level is at least 25%

      (2) PZR level instruments may have measurement errors

   D. (1) CAN be energized without a TSC evaluation if PZR level is at least 25%

      (2) heater elements may have been previously damaged

3. Given the following plant conditions:
   - All RCPs are running
   - RCS pressure is 920 psig and slowly LOWERING
   - SI flow is 100 GPM
   - Containment pressure is 3.2 psig and slowly RISING
   - SG pressures are 1120 psig

   Which ONE of the following completes the statements below, in accordance with E-1, Loss of Reactor Or Secondary Coolant?

   RCPs ___(1)___ be tripped.

   ___(2)___ is the MINIMUM pressure above which the crew will transition to ES-1.2, Post LOCA Cooldown and Depressurization, where the SGs will be required for RCS cooldown.

   A. (1) must NOT

      (2) 230 psig

   B. (1) must NOT

      (2) 360 psig

   C. (1) must

      (2) 230 psig

   D. (1) must

      (2) 360 psig

4. Given the following plant conditions:
    - The crew is implementing E-1, Loss Of Reactor Or Secondary Coolant
    - Intermediate range flux is $3x10^{-11}$ amps and lowering
    - Containment pressure is 26.5 psig and lowering
    - RCS pressure is 675 psig and lowering
    - SG pressure is 950 psig and lowering
    - SI flow is 630 gpm

    Which ONE of the following predicts the status of the Source Range Detectors and identifies the required RHR pump alignment in accordance with E-1?

    A. are energized; Leave RHR Pumps running

    B. are energized; Stop RHR Pumps

    C. are de-energized; Stop RHR Pumps

    D. are de-energized; Leave RHR pumps running

5. Given the following conditions:
   - The Reactor is at 45% power
   - RCP 'B' trips
   - ALB-010, 6-3A, RCS Loop A Tavg Hi/Lo Dev, is in alarm

   Given the above conditions, which of the following completes the statements below?

   SG 'B' Level will initially ___(1)___.

   In accordance with APP-ALB-010 the crew will ___(2)___.

   A. (1) rise

      (2) trip the Reactor and Go to E-0, Reactor Trip or Safety Injection

   B. (1) rise

      (2) commence a Reactor shutdown using GP-006, Normal Plant Shutdown from Power Operation to Hot Standby

   C. (1) lower

      (2) trip the Reactor and Go to E-0, Reactor Trip or Safety Injection

   D. (1) lower

      (2) Commence a Reactor shutdown using GP-006, Normal Plant Shutdown from Power Operation to Hot Standby

6. Given the following plant conditions:
   - The unit is in Mode 6
   - Auto makeup to the VCT is unavailable
   - VCT level is currently 19% and slowly lowering

   Which ONE of the following is required in accordance with AOP-003, Malfunction of Reactor Makeup Control, Attachment 5, Manual Makeup in Modes 5 & 6?

   A. From the MCB: Open 1CS-291 & 292, CSIP Suctions From RWST AND close 1CS-165 & 166 VCT Outlet valves

   B. Locally: Open 1CS-278, Emergency Boric Acid Addition AND 1CS-274, Manual Blend From RMWST Isol valve

   C. From the MCB: Start one Boric Acid pump, open 1CS-283 (FK-113 Borc Acid Flow), 1CS-156 (FCV-113B, Makeup to CSIP Suction) and 1CS-151 (FCV-114, RWMU To Boric Acid Blender)

   D. Locally: Open 1CS-287, Alt Emergency Boration Manual Isol AND 1CS-274, Manual Blend From RMWST Isol valve

7. Given the following plant conditions:
   - The unit is operating at 100% power
   - CCW Surge Tank level is 50% and lowering

   Which ONE of the following completes both statements below?

   The FIRST level at which annunciator ALB-005, 6-1, CCW Surge Tank High-Low Level, will alarm while level lowers is __(1)__.

   In accordance with AOP-014, Loss Of Component Cooling Water, an action required for this condition is __(2)__ .

   A. (1) 38%

      (2) SHUT 1CC-299, RCP Bearing Oil Coolers Return.

   B. (1) 40%

      (2) SHUT 1CC-299, RCP Bearing Oil Coolers Return.

   C. (1) 38%

      (2) SHUT 1CC-252, RCP Thermal Barriers Flow Control.

   D. (1) 40%

      (2) SHUT 1CC-252, RCP Thermal Barriers Flow Control.

8. Given the following plant conditions:
   - The unit is operating at 100% power BOL conditions
   - Steam Dumps are in the T_{avg} mode
   - A Turbine trip occurs
   - The Reactor does NOT trip

   Which ONE of the following completes both statements?
   (Assuming NO operator actions)

   Reactor Delta T indications TI-412A, 422A, and 432A, RCS Loop Prot Delta Ts will __(1)__ .

   SG Safety valves will __(2)__.

   A. (1) rise

      (2) lift

   B. (1) rise

      (2) not lift

   C. (1) lower

      (2) lift

   D. (1) lower

      (2) not lift

9. Given the following plant conditions:
   - The unit is in Mode 3
   - GP-007, Normal Plant Cooldown Mode 3 to Mode 5, is in progress
   - PRZ LO PRESS TRAIN A and B SI BLOCKED status lights are illuminated
   - STM LINE ISOL TRAIN A and B SI BLOCKED status lights are illuminated
   - RCS T_{avg} is 485°F
   - RCS pressure is 1875 psig
   - All SG pressures are 625 psig

   A fault on the 'A' SG occurs inside Containment and the following conditions exist:
   - Containment is 2.6 psig and rising
   - 'A' SG pressure has lowered to 450 psig in the last 30 seconds

   Which ONE of the following identifies (1) the ESFAS signal(s) that has (have) automatically initiated AND (2) the reason for the initiation?

   A. (1) MSL Isolation ONLY

      (2) 'A' SG pressure has lowered below the low pressure actuation setpoint.

   B. (1) MSL Isolation ONLY

      (2) 'A' SG pressure has exceeded the rate actuation setpoint.

   C. (1) MSL Isolation AND MFW Isolation

      (2) 'A' SG pressure has lowered below the low pressure actuation setpoint.

   D. (1) MSL Isolation AND MFW Isolation

      (2) 'A' SG pressure has exceeded the rate actuation setpoint.

10. Given the following plant conditions:
    - The unit is operating at 100% power

    Which ONE of the following predicts the Main FW Pump response, if any, to an inadvertant actuation of Train 'B' Safety Injection?

    A. Both Main FW pumps immediately trip

    B. No Main FW pump trip is initially generated; Both MFW pumps will trip when Tavg lowers to < 564°F

    C. ONLY 'B' Main FW pump will trip

    D. 'B' Main FW pump will trip, 'A' Main FW pump continues to run until Tavg lowers to < 564°F

11. Given the following plant conditions:
    - The unit is operating at 100% power when the following annunciators are reported to the CRS:

        - ALB-022-1-2, Start Up XFMR-A Both 230KV Bkrs Open
        - ALB-022-9-2, Start Up XFMR-B Both 230KV Bkrs Open
        - ALB-018-1-3, Turbine Trip Reactor Trip P4
        - ALB-025-3-3, Diesel Generator B Start Failure
        - ALB-002-2-4A, Condsr Pre Trip Low Vacuum

    - The crew is implementing ES-0.1, Reactor Trip Response

    Based on the above conditions, (1) which AOP is required to mitigate the current conditions AND (2) what is the status of FW isolation valves?

    1FW-159, Main FW A Isolation
    1FW-277, Main FW B Isolation
    1FW-217, Main FW C Isolation

A. (1) AOP-025, Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V)

   (2) OPEN

B. (1) AOP-025, Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V)

   (2) CLOSED

C. (1) AOP-039, Startup And Auxiliary Transformer Trouble

   (2) OPEN

D. (1) AOP-039, Startup And Auxiliary Transformer Trouble

   (2) CLOSED

12. Given the following plant conditions:
    - The unit is operating at 100% power
    - ALB-015, 4-5, Channel III UPS Trouble has just alarmed
    - Feed flows to all SG's have not changed
    - The S-III inverter static switch has shifted to the bypass alignment

Which ONE of the following completes both statements below in accordance with ALB-015, 4-5?

The 7.5 KVA Instrument Bus III INVERTER __(1)__.

Instrument Bus III is currently powered from __(2)__.

A. (1) has lost DC power ONLY

   (2) the 7.5KVA Instrument Bus III Inverter

B. (1) has lost DC power ONLY

   (2) 1A21

C. (1) has lost AC and DC power

   (2) 1D21

D. (1) has lost AC and DC power

   (2) 1A21

13. Given the following plant conditions:
    - The plant is operating at 100% power
    - 'B' Train Safety Equipment is in service
    - Both ESW Pumps are running to support surveillance testing

    The following indications and annunciators are observed:
    - ALB-02-4-5, SERV WTR LEAKAGE
    - ALB-02-5-5, SERV WTR HEADER A HIGH/LOW FLOW
    - ALB-02-6-1, SERV WTR SUPPLY HEADER A LOW PRESS
    - CNMT Sump level is increasing on ERFIS

    The crew enters AOP-022, Loss of Service Water and secures the 'A' ESW Pump.

    Which ONE of the following actions in accordance with AOP-022, identifies (1) the possible location of the rupture AND (2) the action required by the procedure?

    A. (1) CNMT Fan Coil Units

       (2) Shut 1SW-231, NNS CNMT Fan CLRS Inlet Isol, AND 1SW-242, NNS CNMT Fan CLRS Outlet Isol

    B. (1) CNMT Fan Coil Units

       (2) Shut 1SW-231, NNS CNMT Fan CLRS Inlet Isol, AND 1SW-276, Headers A&B Return to Normal Service Water

    C. (1) CNMT Fan Coolers

       (2) Shut ONLY AH-2/3 ESW Supply and Return Valves

    D. (1) CNMT Fan Coolers

       (2) Shut AH-1/2/3/4 ESW Supply and Return Valves

14. Given the following plant conditions:
    - The crew is currently implementing E-3, Steam Generator Tube Rupture
    - The OAC reports Train 'A' Phase A valves will not open after resetting Phase A

Based on the above conditions, which ONE of the following completes the statements below?

The required RCS depressurization will be accomplished using ___(1)___ .

The E-3 RCS depressurization termination criteria, when using the PZR Spray Valves, is ___(2)___ the termination criteria when the PZR PORVs are used to depressurize the RCS.

A. (1) PZR Spray Valves

   (2) different than

B. (1) PZR Spray Valves

   (2) exactly the same as

C. (1) PZR PORVs

   (2) different than

D. (1) PZR PORVs

   (2) exactly the same as

15. Given the following plant conditions:
    - The unit is operating at 100% power
    - Grid frequency is begining to lower

    Which ONE of the following completes the following statements in accordance with AOP-028, Grid Instability?

    The highest frequency, below which entry into AOP-028 will be required is ___(1)___ .

    The highest frequency at which an automatic Reactor trip, as well as a trip of all RCPs, will occur is ___(2)___ .

A. (1) 59.0 Hz

   (2) 57.5 Hz

B. (1) 59.0 Hz

   (2) 58.4 Hz

C. (1) 59.5 Hz

   (2) 57.5 Hz

D. (1) 59.5 Hz

   (2) 58.4 Hz

16. Given the following plant conditions:
    - The unit was operating at 100% power
    - A LOCA has occurred in the RAB and the crew is implementing ECA-1.2, LOCA Outside Containment, step 6 - check break isolated

    Which ONE of the following identifies (1) a parameter trend, which is used to confirm that the break is isolated, AND (2) the reason for the trend?

    A. (1) RCS pressure rising

       (2) SI flow is filling up the RCS

    B. (1) RCS pressure rising

       (2) Main Steam Lines are isolated

    C. (1) PZR level rising

       (2) SI flow is filling up the RCS

    D. (1) PZR level rising

       (2) Main Steam Lines are isolated

17. Given the following plant conditions:
    - Bleed & Feed is in progress in accordance with FR-H.1, Response to Loss of Secondary Heat Sink
    - Main Feedwater is now available
    - No AFW Pumps are available
    - Core Exit Thermocouple temperatures are stable
    - All SG wide range levels are 10%

    Which ONE of the following completes the statements below in accordance with FR-H.1, Attachment 1, Guidance on Restoration of Feed Flow?

    Feed one intact SG at no more than ___(1)___.

    Feed flow may be raised to maximum rate as soon as SG Wide Range level rises to greater than ___(2)___.

    A. (1) 50 KPPH

       (2) 15%

    B. (1) 50 KPPH

       (2) 25%

    C. (1) the lowest controllable rate

       (2) 15%

    D. (1) the lowest controllable rate

       (2) 25%

18. Given the following plant conditions:
    - A LOCA has occurred
    - Containment pressure is 15 psig and LOWERING
    - Due to a failure of A and B train, CNMT Sump to RHR Pump suction valves, the crew has transitioned from E-1, Loss of Reactor Or Secondary Coolant to ECA-1.1, Loss Of Emergency Coolant Recirculation
    - Two CSIPs, two Containment Fan Coolers, both CT pumps and both RHR pumps are running
    - RWST level is approximately 30% and lowering
    - Wide Range Containment Sump level is 140 inches

    Which ONE of the following identifies (1) the reason why 'A' CT pump is required to be secured AND (2) another required action if RWST level lowers to 3% while the crew continues with ECA-1.1?

    (Reference provided)

    A. (1) Preserve RWST inventory

       (2) secure the other Containment Spray pump when Containment pressure is less than 10 psig

    B. (1) Preserve RWST inventory

       (2) establish makeup to the RCS from an alternate source

    C. (1) Preclude unnecessary entry into FR-Z.2, Reponse To Containment Flooding

       (2) secure the other Containment Spray pump when Containment pressure is less than 10 psig

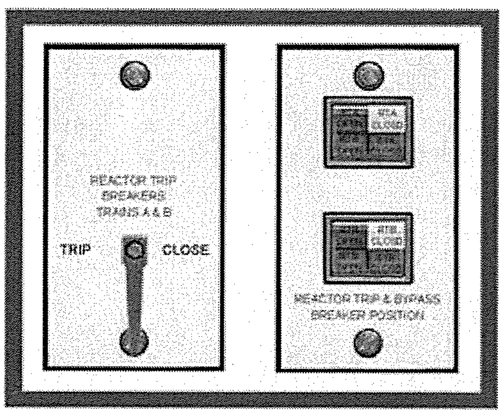
    D. (1) Preclude unnecessary entry into FR-Z.2, Reponse To Containment Flooding

       (2) establish makeup to the RCS from an alternate source

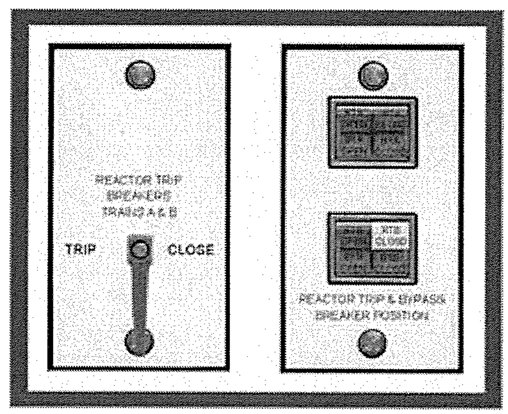
19. Given the following plant conditions:
    - A Reactor startup is in progress
    - The OAC withdraws CBD from 20 steps to the next doubling in accordance with GP-004, Reactor Startup (Mode 3 To Mode 2)
    - The OAC releases the Rod Motion switch, but CBD rods continue to withdraw
    - The MCB Rx Trip Switch #1 is taken to Trip
    - The Reactor Trip Breaker indications change as indicated in the pictures below

(NOTE: the light bulbs are not blown)

Before Rx Trip Switch # 1 taken to Trip



After Rx Trip Switch # 1 taken to Trip



Which ONE of the following completes the statement below?

The current status of the Reactor is __(1)__ AND the indication ~~of~~ for the "A" Reactor Trip Breaker~~s~~ on the MCB indicates a failure of the __(2)__ Trip coil.

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A. (1) tripped

   (2) UV

B. (1) tripped

   (2) Shunt

C. (1) NOT tripped

   (2) UV

D. (1) NOT tripped

   (2) Shunt

20. A traverse drive system (roller chain) failure has occurred on the fuel transfer system conveyor while the cart was in the horizontal position and loaded with a fuel bundle inside Containment.

    Which ONE of the following identifies (1) the back-up method of returning the fuel transfer cart to the Fuel Handling Building (FHB) in accordance with FHP-020, Refueling Operations AND (2) where the equipment is operated?

    A. (1) Emergency pull-out cable

       (2) Inside the Fuel Handling Building

    B. (1) Emergency pull-out cable

       (2) Inside the Containment Building

    C. (1) Auxiliary Crane Traverse

       (2) Inside the Fuel Handling Building

    D. (1) Auxiliary Crane Traverse

       (2) Inside the Containment Building

21. Given the following plant conditions:
    - The plant is operating at 100%
    - One SG has developed a tube leak and the crew is implementing AOP-016, Excessive Primary Plant Leakage
    - Chemistry has been directed to perform CRC-804, Primary to Secondary Leak Rate Monitoring, to quantify the leak rate

    Which ONE of the following instrument(s) is/are used to determine the primary to secondary leak rate in accordance with AOP-016?

    A. SG Blowdown Radiation Monitor, REM-01BD-3527

    B. Turbine Building Vent Stack Effluent Monitor, RM-1TV-3536-1

    C. Condenser Vacuum Pump Effluent Monitor, REM-01TV-3534

    D. Main Steam Line Radiation Monitors RM-01MS-3591 SB, 3592 SB, or 3593 SB

22. Given the following plant conditions:
    - The unit was operating at 60% power when air leakage into the Condenser resulted in entry in AOP-012, Partial Loss of Condenser Vacuum
    - A load reduction was initiated in accordance with AOP-038, Rapid Downpower

Time	Power	Control Bank C	Control Bank D
0800	60%	225 steps	130 steps
0830	50%	223 steps	95 steps
0900	45%	213 steps	85 steps
0930	40%	198 steps	70 steps
1000	35%	178 steps	50 steps

Which ONE of the following identifies the EARLIEST time that the LCO for Technical Specification 3.1.3.6, Control Rod Insertion Limits was not met?

(Reference provided)

A. 0830

B. 0900

C. 0930

D. 1000

23. Given the following plant conditions:
    - An RWST leak has occurred
    - REM-01MD-3530, Tank Area Drain Transfer Pumps Monitor, is in HIGH alarm
    - Contaminated water is filling the retention dike area

Which ONE of the following completes BOTH statements below?

As a result of this radiation alarm, ___(1)___ automatically.

In accordance with AOP-008, Accidental Release of Liquid Waste, a leak from the Refueling Water Storage Tank requires manual operation to ___(2)___.

A. (1) the Tank Area Drain Transfer Pump stops

(2) shut 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve

B. (1) the Tank Area Floor Drain Sump Pump stops

(2) shut 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve

C. (1) 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve shuts

(2) secure the Tank Area drain pump

D. (1) 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve shuts

(2) secure the Tank Area Floor Drain Sump pump

24. Given the following plant conditions:
    - The unit was operating at 100% power
    - An 86 Lockout occurs on the 'A' and 'B' SUTs
    - Sixty minutes later, the following plant conditions exist:

- RVLIS Full Range	63% and lowering
- Core Exit Thermocouples	745°F and rising
- Containment Pressure	3.5 psig and rising
- Pressurizer Level	0%
- SG NR level 'A'	38%
- SG NR level 'B'	44%
- SG NR level 'C'	23%

Based on these conditions, which ONE of the following completes the statement below?

The Core Cooling Critical Safety Function Status Tree requires entry into ___(1)___ AND the crew will depressurize the SGs to 130 psig using ___(2)___.

A. (1) FR-C.2, Response To Degraded Core Cooling

   (2) steam dumps

B. (1) FR-C.2, Response To Degraded Core Cooling

   (2) SG PORVs

C. (1) FR-C.1, Response To Inadequate Core Cooling

   (2) steam dumps

D. (1) FR-C.1, Response To Inadequate Core Cooling

   (2) SG PORVs

25. Given the following plant conditions:
    - A LOCA has occurred
    - The crew is implementing ES-1.2, Post LOCA Cooldown and Depressurization
    - Safety Injection has NOT been terminated

    Which ONE of the following identifies (1) the parameter used by the operator to determine whether the CLAs are required to be isolated AND (2) the reason the accumulators are isolated under these conditions?

    A. (1) RCS Cold Leg Temperature

       (2) To allow minimum subcooling to be established

    B. (1) RCS Cold Leg Temperature

       (2) To prevent gas binding of the S/G U-tubes

    C. (1) RCS Hot Leg Temperature

       (2) To allow minimum subcooling to be established

    D. (1) RCS Hot Leg Temperature

       (2) To prevent gas binding of the S/G U-tubes

26. The crew has transitioned to E-1, Loss of Reactor or Secondary Coolant and is presently evaluating the RHR System capable of Cold Leg Recirculation.

    The following conditions exist:
    - Offsite Power has been lost
    - EDG 'B' has tripped
    - CNMT Pressure is 17 psig and rising
    - CNMT High Range Rad Monitors are in alarm
    - CNMT Wide Range Sump Level is reading 211 inches
    - RVLIS Full Range Level is reading 60%
    - RCS Cold Leg Temperature is reading 265°F
    - RCS Wide Range Pressure is reading 225 psig
    - Core Exit Thermocouples are reading 740°F
    - Containment Spray pump 'A' has tripped

    Which ONE of the following is the procedure that the crew is required to implement at this time?

    A. FR-Z.1, Response to High Containment Pressure

    B. FR-Z.2, Response to Containment Flooding

    C. FR-C.2, Response to Degraded Core Cooling

    D. FR-P.1, Response to Imminent Pressurized Thermal Shock

27. Which ONE of the following identifies the sources of water, in accordance with the WOG Background Document for FR-Z.2, Response To Containment Flooding, that are the basis for the maximum anticipated containment water level?

A. Condensate Storage Tank, Emergency Service Water, Reactor Coolant System

B. Refueling Water Storage Tank, Emergency Service Water, Reactor Coolant System

C. Condensate Storage Tank, Emergency Service Water, Refueling Water Storage Tank

D. Refueling Water Storage Tank, Reactor Coolant System, Condensate Storage Tank

28. Given the following plant conditions:
    - A plant cooldown is in progress following a planned shutdown in accordance with GP-007, Normal Plant Cooldown Mode 3 To Mode 5 to repair the Reactor Vessel Head
    - The following conditions exist for RCP 'B'

Time	Upper Thrust Bearing Temperature	# 1 Seal Differential Pressure
0800	154°F	253 psig
0805	159°F	237 psig
0810	165°F	223 psig
0815	174°F	209 psig
0820	183°F	198 psig

Which ONE of the following completes the statements below?

The ___(1)___ is the first RCP parameter outside the normal limit.

In accordance with GP-007 the action required under these conditions is ___(2)___.

A. (1) Upper Thrust Bearing Temperature

   (2) stop RCP 'B'

B. (1) Upper Thrust Bearing Temperature

   (2) open the RCP # 1 Seal Bypass

C. (1) # 1 Seal Differential Pressure

   (2) stop RCP 'B'

D. (1) # 1 Seal Differential Pressure

   (2) isolate the RCP 'B' Seal Water Return

29. Given the following plant conditions:
    - A Large Break LOCA has occurred
    - RWST level indicates 22% and continues to lower
    - 1RH-1, RCS Loop A to RHR Pump A-SA is CLOSED

In accordance with ES-1.3, Transfer to Cold Leg Recirculation, which ONE of the following actions completes the statement below to establish the 'A' CSIP alignment for long term operation?

The operator must FIRST __(1)__ 1CS-746 AND then __(2)__ must be OPENED.

1CS-746, CSIP A Alternate Miniflow
1RH-25 SA, Suction From RHR Heat Exchanger A-SA
1SI-340, Safety Injection A train to Cold Leg

A. (1) CLOSE

   (2) 1RH-25

B. (1) CLOSE

   (2) 1SI-340

C. (1) OPEN

   (2) 1RH-25

D. (1) OPEN

   (2) 1SI-340

30. Given the following plant conditions:
    - The unit was operating at 100% power
    - ALB-007-4-3, VCT High-Low Level is in Alarm
    - VCT level transmitter LI-115 has failed high
    - VCT level transmitter LI-112 reads 14%

    Which ONE of the following completes the statements below?

    In accordance with AOP-003, Malfunction Of Reactor Makeup Control, the HIGHEST VCT level below which gas binding of the running CSIP is a concern is ___(1)___ .

    Given these conditions, RWST suction valves AND VCT Outlet valves will ___(2)___.

    1CS-291, Suction from RWST LCV-115B
    1CS-292, Suction from RWST LCV-115D
    1CS-165, VCT Outlet LCV-115C
    1CS-166, VCT Outlet LCV-115E

    A. (1) 5%

       (2) automatically realign

    B. (1) 5%

       (2) require manual realignment

    C. (1) 10%

       (2) automatically realign

    D. (1) 10%

       (2) require manual realignment

31. Which ONE of the following identifies (1) the MINIMUM Containment wide range sump level required to place the RHR system in Cold Leg Recirculation in accordance with ES-1.3, Transfer To Cold Leg Recirculation AND (2) the basis for this level?

A. (1) 137.5 inches

(2) ensures the recirculation sump strainers are completely submerged

B. (1) 137.5 inches

(2) ensures the recirculation sump pH level is acceptable

C. (1) 142 inches

(2) ensures the recirculation sump strainers are completely submerged

D. (1) 142 inches

(2) ensures the recirculation sump pH level is acceptable

32. Which ONE of the following completes both statements in accordance with OP-107, CVCS, Attachment 5, Replacing B CSIP with C CSIP?

To align the C CSIP to 1B-SB, a transfer switch located in the RAB, on elevation __(1)__ , must be operated.

First, the B Train Kirk Key Lock Switch must be rotated, then __(2)__ must be closed.

A. (1) 236' just south of the 'A' CSIP room

(2) the transfer switch, which is a knife switch,

B. (1) 236' just south of the 'A' CSIP room

(2) a handle must be placed into the handle casting and the transfer switch

C. (1) 286' Switchgear room

(2) the transfer switch, which is a knife switch,

D. (1) 286' Switchgear room

(2) a handle must be placed into the handle casting and the transfer switch

33. Given the following plant conditions:
   - The unit is operating at 100% power
   - ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press Or Temp, Alarms
   - PRT temperature indicates 105°F
   - PRT pressure indicates 8 psig
   - PRT level indicates 73%

   Which ONE of the following (1) identifies the cause of the alarm AND (2) describes the operator response for this alarm in accordance with the Annunciator Panel Procedure and OP-100, Reactor Coolant System?

   A. (1) PRT level is high

      (2) Drain the PRT to the Reactor Coolant Drain Tank

   B. (1) PRT level is high

      (2) Drain the PRT to the Waste Hold Tank

   C. (1) PRT pressure is high

      (2) Vent the PRT to the Waste Gas Vent Header

   D. (1) PRT pressure is high

      (2) Drain the PRT to the Waste Hold Tank

34. Which ONE of the following completes both statements in accordance with OP-100, Reactor Coolant System?

Per the OP-100, Precautions and Limitation, the MAXIMUM temperature below which the Pressurizer Relief Tank (PRT) should be maintained is __(1)__.

A rapid cool down of the PRT can be performed by draining the PRT and providing makeup water to the spray header from the __(2)__.

A. (1) 120°F

(2) RCDT

B. (1) 120°F

(2) RMWST

C. (1) 150°F

(2) RCDT

D. (1) 150°F

(2) RMWST

35. Given the following plant conditions:
  - The unit is at 100% Reactor power
  - A Reactor trip and Safety Injection has occurred
  - Phase 'A' fails to actuate

Which ONE of the following CCW System loads are isolated from the CCW System?

(Assume NO Operator actions)

A. Primary Sample Panel AND Gross Failed Fuel Detector

B. RCDT heat exchanger AND Excess Letdown heat exchanger

C. RCDT heat exchanger AND Gross Failed Fuel Detector

D. Primary Sample Panel AND Excess Letdown heat exchanger

36. Given the following plant conditions:
    - The unit is operating at 100% power
    - PZR Pressure Channel (PT-445) fails high

Which ONE of the following completes the statement below describing the response of the PZR Pressure Control System to this failure?

__(1)__ PZR PORV(s) will OPEN AND remain OPEN until the __(2)__ setpoint is reached.

A. (1) ONE

   (2) Safety Injection

B. (1) ONE

   (2) P-11, PZR High Pressure

C. (1) TWO

   (2) Safety Injection

D. (1) TWO

   (2) P-11, PZR High Pressure

37. Given the following plant conditions:
    - The unit is at 100% power
    - The PZR pressure master controller, PK-444A, is in AUTOMATIC
    - A PZR pressure master controller malfunction causes the setpoint to slowly drift to 61% over 10 minutes

Which ONE of the following is the expected plant response to the drifting of the setpoint?

(Assume NO Operator Actions)

A. Both spray valves will open

B. The control heaters will be at maximum output

C. Pressure will stabilize at 2280 psig

D. One PZR PORV will cycle

38. Given the following plant conditions:
    - The unit was operating at 8% power when the following parameters are indicated prior to the Reactor automatically tripping:
    - PI-455, RCS Pressure is 2380 psig
    - PI-456, RCS Pressure is 2390 psig
    - PI-457, RCS Pressure is 2400 psig
    - LI-459, PRZ Level is 92%
    - LI-460, PRZ Level is 90%
    - LI-461, PRZ Level is 93%

    Which ONE of the following (1) identifies the condition that caused the automatic Reactor trip AND (2) the associated basis for the automatic trip?

    A. (1) PZR High Level

       (2) provides protection against over pressurizing the RCS.

    B. (1) PZR High Level

       (2) precludes water relief through the Pressurizer safety valves.

    C. (1) PZR High Press

       (2) provides protection against over pressurizing the RCS.

    D. (1) PZR High Press

       (2) precludes water relief through the Pressurizer safety valves.

39. Given the following plant conditions:
    - The crew is responding to a Large Break LOCA in E-1, Loss Of Reactor Or Secondary Coolant
    - Both RHR pumps are running

    -The following actions have been taken:
      - SI and Phase A have both been reset
      - Instrument Air and Nitrogen have been restored to Containment

    Subsequently, a Loss of Off-site power occurs.

    Which ONE of the following completes the statement below?

    The sequencers will run in ___(1)___ after the Loss of Off-site power AND the RHR pumps ___(2)___.

    A. (1) Program A

       (2) will automatically start in load block 2

    B. (1) Program A

       (2) must be manually started after load block 9

    C. (1) Program B

       (2) will automatically start in load block 2

    D. (1) Program B

       (2) must be manually started after load block 9

40. Which ONE of the following completes the statement below?

Instrument Buses ___(1)___ AND ___(2)___ provide power to the ESFAS Slave Relays.

A. (1) SI

(2) SII

B. (1) SII

(2) SIII

C. (1) SI

(2) SIV

D. (1) SIII

(2) SIV

41. Given the following plant conditions:
    - The unit was operating at 100% power
    - Containment Fan Coolers are in the Normal Cooling mode
    - A steam leak into Containment occurs
    - Containment pressure is 2.6 psig and rising
    - Containment temperature is 135°F and rising

Which ONE of the following completes the statement below?

Containment Fan Coolers are running in ____(1)____ speed with the post-accident dampers ____(2)____.

(Assume NO Operator actions)

A. (1) SLOW

   (2) SHUT

B. (1) SLOW

   (2) OPEN

C. (1) HIGH

   (2) SHUT

D. (1) HIGH

   (2) OPEN

42. Which ONE of the following completes the statement below?

Following a Containment spray actuation signal, the HIGHEST Containment spray additive tank level at which Containment spray chemical addition valves 1CT-11 and 1CT-12 will auto-close is ______ .

A. 23.4%

B. 10%

C. 2%

D. 0%

43. Given the following plant conditions
- The unit was operating at 100% power
- A LOCA has occurred and the crew is implementing E-1, Loss Of Reactor Or Secondary Coolant
- The CT Pump 'A' tripped while aligned to the RWST

When RWST level reaches the Low-Low level setpoint, which ONE of the following identifies (1) the recirc sump suction valve(s) will automatically open AND (2) after the recirc suction valve(s) reach(es) the full-open position, RWST suction valve(s) which will automatically close?

1CT-105, Containment Sump To CNMT Spray Pump A-SA
1CT-102, Containment Sump To CNMT Spray Pump B-SB

1CT-26, RWST To CNMT Spray Pump A-SA
1CT-71, RWST To CNMT Spray Pump B-SB

A. (1) 1CT-102 ONLY

(2) 1CT-71 ONLY

B. (1) 1CT-102 AND 1CT-105

(2) 1CT-71 ONLY

C. (1) 1CT-102 ONLY

(2) 1CT-26 AND 1CT-71

D. (1) 1CT-102 AND 1CT-105

(2) 1CT-26 AND 1CT-71

44. Given the following plant conditions:
    - The unit is in MODE 2 at 1% power
    - T_{avg} is at the NO load reference value
    - A failure of an SG PORV results in the following:
        - Steam Generator pressures at 1028 psig

Which ONE of the following completes BOTH statements below?

Operation of the Condenser Steam Dumps is ___(1)___ at this time.

In accordance with GP-004, Reactor Startup (Mode 3 to Mode 2) the operator has 15 minutes to restore temperature to above a MINIMUM of ___(2)___.

(Assume NO operator action)

A. (1) blocked

   (2) 553°F

B. (1) blocked

   (2) 551°F

C. (1) NOT blocked

   (2) 553°F

D. (1) NOT blocked

   (2) 551°F

45. Given the following plant conditions:
    - The unit is operating at 91% power
    - A Loss of Main Feedwater Pump 'B' occurs
    - The crew enters AOP-010, Feedwater Malfunctions

    Which ONE of the following describes (1) the plant response AND (2) the action required in accordance with AOP-010?

    A. (1) Automatic turbine runback is initiated

       (2) Isolate Steam Generator Blowdown

    B. (1) Automatic turbine runback is initiated

       (2) Trip the Reactor and go to E-0

    C. (1) Automatic turbine runback is NOT initiated

       (2) Isolate Steam Generator Blowdown

    D. (1) Automatic turbine runback is NOT initiated

       (2) Trip the Reactor and go to E-0

46. Given the following plant conditions:
    - The unit was operating at 100% Reactor power when a station black out occurs
    - The crew is implementing ECA-0.0, Loss of All AC Power
    - The TDAFW has been running in automatic with the controller setpoint at 31% for several minutes
    - NO operator actions have been taken on the AFW system
    - All SG NR levels are approximately 9% and lowering
    - AFW flow is currently 160 kpph

    Which ONE of the following identifies the action(s) required to be taken for these conditions?

    A. Transition to FR-H.1, Response to Loss of Secondary Heat Sink.

    B. Open 1 SG PORV (on the SG with the highest level) to lower SG pressure.

    C. Place Aux FW Turbine SPD PDK-2180.1 in MAN and depress the output RAISE pushbutton.

    D. Depress the RAISE pushbutton(s) on the TDAFW FCV(s).

47. Given the following plant conditions:
- The unit is operating at 100% power
- Annunciator ALB-014, 7-4, SG A, B, C Backleakage High Temp, has alarmed
- An NLO has been dispatched to verify local temperatures

Which ONE of the following completes BOTH of the statement below?

The reason this condition occurred is because a __(1)__ is leaking.

In accordance with the AOP-010, under these conditions with the TDAFW piping local temperature > 212°F, the FIRST action required is __(2)__.

A. (1) TDAFW pump steam supply piping check valve

(2) start the TDAFW pump to flush the line through the exhaust

B. (1) TDAFW pump steam supply piping check valve

(2) isolate the TDAFW pump discharge header

C. (1) AFW feed water piping check valve

(2) start the TDAFW pump to flush the line to the SGs

D. (1) AFW feed water piping check valve

(2) isolate the TDAFW pump discharge header

48. Given the following plant conditions:
    - The unit is operating at 100% power
    - Aux Bus 1E deenergizes and is locked out

    Which ONE of the following describes an effect on the unit?

    A. RCP 'C' is deenergized

    B. CSIP 'A' is momentarily deenergized

    C. CSIP 'B' is momentarily deenergized

    D. CTMU Pump '1X' is deenergized

49. Which ONE of the following completes the statements below in accordance with OP-156.01, DC Electrical Distribution, Section 8.2. Rotation of 125 VDC NNS Battery Chargers?

When placing a 125VDC battery charger in service, its ___(1)___ breaker is closed first.

A Low DC Volt alarm ___(2)___ expected after this first breaker is closed.

A. (1) DC output

(2) is NOT

B. (1) DC output

(2) is

C. (1) AC input

(2) is NOT

D. (1) AC input

(2) is

50. Given the following plant conditions:
   - The unit is currently in MODE 3
   - DP-1A-SA has lost power

   Which ONE of the following completes the statement below for the 'A' MDAFW Pump?

   Breaker control from the MCB ___(1)___ AND the control switch indication on the MCB will ___(2)___ .

   A. (1) remains available

      (2) extinguish

   B. (1) is not available

      (2) extinguish

   C. (1) remains available

      (2) remain illuminated

   D. (1) is not available

      (2) remain illuminated

51. Given the following EDG Fuel Oil Data:
    - Both Fuel Oil Day Tanks Specific gravity: 0.835
    - Fuel Oil Day Tank 'A': 47%
    - Fuel Oil Storage Tank 'A': 90,000 gallons
    - Fuel Oil Day Tank 'B': 42%
    - Fuel Oil Storage Tank 'B': 110,000 gallons

    Which ONE of the following identifies the status of the EDGs in accordance with Technical Specification 3.8.1.1, Electrical Power Systems - AC Sources?

    (Reference provided)

	EDG 'A'	EDG 'B'
A.	OPERABLE	OPERABLE
B.	OPERABLE	INOPERABLE
C.	INOPERABLE	OPERABLE
D.	INOPERABLE	INOPERABLE

52. Which ONE of the following identifies an RAB radiation monitor that requires entry into AOP-032, High RCS Activity, when a valid HIGH alarm condition exists?

    A. RM-1RR-3600, Recycle Evaporator Valve Gallery

    B. RM-21CR-3578A, Recycle Monitor Tank 1A & 2A

    C. RM-1RR-3605A, Sample Room 1A Elev. 236

    D. RM-1RR-3611, Recycle Holdup Tank Area

53. Given the following plant conditions:
   - The unit is in Mode 4
   - 'A' Train safety equipment is in service
   - The 'B' NSW pump is tagged out for maintenance
   - A Loss of Off-site power occurs
   - The 'A' EDG failed to start

   Which ONE of the following completes the statement below?

   ESW is providing flow to __(1)__ CCW Heat Exchanger(s) with ESW return header flow aligned to the __(2)__.

   A. (1) ONLY 'B'

      (2) Auxiliary Reservoir

   B. (1) ONLY 'B'

      (2) Cooling Tower Basin

   C. (1) 'A' AND 'B'

      (2) Auxiliary Reservoir

   D. (1) 'A' AND 'B'

      (2) Cooling Tower Basin

54. Given the following plant conditions:
    - The unit is operating at 100% power
    - An Instrument Air leak is occurring
    - Instrument Air pressure is currently 85 psig and stable

    Which ONE of the following predicts the plant response for the current condition?

    A. All FW flow control valves will CLOSE.

    B. RCS letdown flowpath valves drift to mid-position.

    C. PZR Spray valves drift to mid-position.

    D. Gland Steam Seal Spillover Regulator Valve will OPEN.

55. Which ONE of the following completes the statements below?

There are ___(1)___ Primary Shield Cooling Fans.

The Primary Shield Cooling Fans are located in the Containment Building at elevation ___(2)___.

A. (1) Two

(2) 221'

B. (1) Two

(2) 236'

C. (1) Four

(2) 221'

D. (1) Four

(2) 236'

56. Given the following plant conditions:
    - A Loss of Off-site Power occurs while the unit was operating at 100% power
    - EDG A-SA failed to start
    - Load Block 9 has been verified complete on EDG B-SB
    - RCS pressure is 2180 psig

    Assuming NO operator action has been taken, which ONE of the following identifies the PZR Heaters group(s) that are currently energized, if any?

    A. None

    B. B only

    C. C only

    D. B and C only

57. Given the following plant conditions:
    - The unit is operating at 8% power
    - Intermediate Range (IR) N35 is inoperable
    - N35 Level Trip Switch is in BYPASS in accordance with OWP-RP-21, Reactor Protection

    The following occur:
    - At 12:00 N35 Instrument Power fuses blow
    - At 12:15 N35 Control Power fuses blow

    Which ONE of the following identifies (1) the status of the Reactor Trip Breakers AND (2) the reason for the status of the Reactor Trip Breaker?

    A. (1) OPEN at 12:00

       (2) N35 Instrument Power fuses blew

    B. (1) OPEN at 12:15

       (2) N35 Control Power fuses blew

    C. (1) CLOSED at 12:15

       (2) N35 is BLOCKED in accordance with GP-005, Power Operations

    D. (1) CLOSED at 12:15

       (2) N35 is in BYPASS in accordance with OWP-RP-21

58. Given the following plant conditions:
- The unit is operating at 100% power
- ALB-015-1-5, 7.5 KVA UPS Trouble, alarms

Which ONE of the following identifies (1) the uninterruptible power supply that is potentially affected AND (2) the action taken, if this power supply is lost, per AOP-024, Loss Of Uninterruptible Power Supply?

A. (1) UPP-1B

(2) Locally control Steam Dumps

B. (1) UPP-1B

(2) Locally control Condensate Booster pumps

C. (1) UPP-1

(2) Locally control Steam Dumps

D. (1) UPP-1

(2) Locally control Condensate Booster pumps

59. Given the following plant conditions:
    - At 0200, the unit was operating at 100% power
    - The crew is implementing E-1, Loss of Reactor or Secondary Coolant
    - The hydrogen monitoring system has been aligned

    - At 0400 Containment hydrogen concentration was 0.35% and slowly rising

    - At 0500 Containment hydrogen concentration was 0.52% and the Hydrogen Recombiner 1A was placed in operation in accordance with OP-125, Post Accident Hydrogen Systems

    - At 1800 Containment hydrogen concentration has increased to 3.14%

    Based on these conditions, which ONE of the following actions is required in accordance with OP-125?

    A. Start the 1B recombiner ONLY when Containment hydrogen concentration exceeds 3.5%, then operate both recombiners.

    B. Start the 1B recombiner ONLY when Containment hydrogen concentration exceeds 4%, then operate both recombiners.

    C. Start the 1B recombiner NOW and operate both recombiners.

    D. Do NOT start the 1B recombiner. Increase the 1A recombiner power by 4 KW.

60. Given the following plant conditions:
   - Refueling is in progress.
   - A spent fuel assembly is being moved in the Fuel Handling Building (FHB) when it is damaged.
   - Spent Fuel Pool area radiation monitor RM-1FR-3566A-SA is in HIGH alarm.
   - Spent Fuel Pool area radiation monitor RM-1FR-3567B-SB is in ALERT.

   Which ONE of the following completes BOTH of the statements below?

   __(1)__ train(s) of Fuel Handling Building Ventilation Emergency Exhaust has(have) received an automatic start signal.

   RM-1FR-3566A-SA radiation monitor __(2)__ sound an alarm locally.

   A. (1) ONLY 'A'

      (2) will

   B. (1) BOTH 'A' and 'B'

      (2) will

   C. (1) ONLY 'A'

      (2) will NOT

   D. (1) BOTH 'A' and 'B'

      (2) will NOT

61. Which ONE of the following completes the statement below concerning the Waste Gas System in accordance with Technical Specification 3.11.2.5, Radioactive Effluents - Explosive Gas Mixture?

The concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM downstream of the hydrogen recombiners shall be limited to less than or equal to ___(1)___ by volume whenever the hydrogen concentration exceeds ___(2)___ by volume.

A. (1) 2%

(2) 2%

B. (1) 2%

(2) 4%

C. (1) 4%

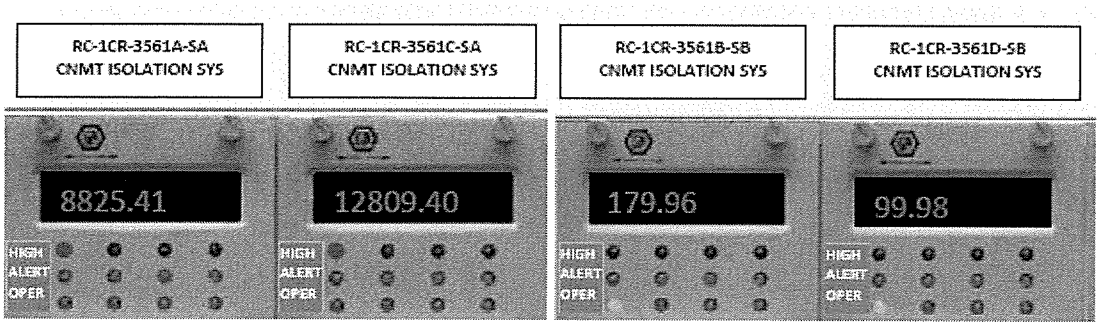
(2) 2%

D. (1) 4%

(2) 4%

62. Given the following plant condition:
   - The unit is operating at 100% Reactor power
   - S-1A, Airborne Radioactivity Removal fan is in AUTO

   - Subsequently, CVI rad monitors indicate as follows:



   - E-5, Containment Pre-entry Purge Fan failed to trip

Which ONE of the following identifies the system response during these conditions?

A. Containment Vacuum Relief dampers (CB-D51 SA and CB-D52 SB) receive a CLOSE signal.

B. Airborne Radioactivity Removal fan S-1A will Auto START.

C. Containment Isolation Phase "A" isolation valves receive a CLOSE signal.

D. Containment Pre-entry Purge Makeup fans AH-81A/B receive a TRIP signal.

63. Given the following plant conditions:
   - The unit is operating in Mode 4 preparing to start up
   - CWP 'A' is running
   - NSW Pump 'A' is running

Which ONE of the following completes the statements below?

Opening 1CW-77, Cooling Tower Bypass Valve #1, will ___(1)___ the back pressure on the NSW system.

To prevent CTMU Pump run out, the MAXIMUM total flow allowed is ___(2)___ gpm.

A. (1) reduce

   (2) 30,000

B. (1) reduce

   (2) 22,000

C. (1) NOT affect

   (2) 30,000

D. (1) NOT affect

   (2) 22,000

64. Given the following plant conditions:
    - The plant is operating at 100% power
    - Air pressure on PI-9751.1, Instrument Air Header Pressure is 80 psig

Which ONE of the following completes the statement below?

1SA-506, Service Air Header Isol. Valve, is __(1)__ AND ALB-002, 8-1, Instrument Air Low Pressure Annunciator, is __(2)__.

A. (1) OPEN

(2) in Alarm

B. (1) OPEN

(2) NOT in Alarm

C. (1) CLOSED

(2) in Alarm

D. (1) CLOSED

(2) NOT in Alarm

65. Given the following plant conditions:
   - The Motor Driven Fire pump has just been stopped per FPT-3001, Motor Driven Main Fire Pump Operability Test Monthly Interval Modes: All
   - A fire occurs
   - Fire header pressure lowers to 90 psig
   - Fire header pressure is now 125 psig and stable

   Which ONE of the following completes the statements below?

   The Motor-Driven Fire Pump is ___(1)___.

   The Diesel Driven Fire Pump is ___(2)___.

   A. (1) OFF

      (2) OFF

   B. (1) RUNNING

      (2) OFF

   C. (1) OFF

      (2) RUNNING

   D. (1) RUNNING

      (2) RUNNING

66. Which ONE of the following completes the statement below describing the location and control of the Security Master Key in the control room that provides access to plant vital areas?

The Security Master Key is located in a ___(1)___ ,

The keys to this Box/Cabinet are controlled by ___(2)___.

A. (1) locked box in the SM desk

(2) the SM

B. (1) 'break-glass' cabinet

(2) the SM

C. (1) locked box in the CRS desk

(2) the CRS

D. (1) 'break-glass' cabinet

(2) Security

67. Which ONE of the following completes BOTH of the statements below in accordance with OPS-NGGC-1314, Communications?

Standing Instructions ___(1)___ contain items of long term significance.

During shift turnover, in accordance OPS-NGGC-1314, it is REQUIRED that the crew review ___(2)___.

A. (1) normally

(2) ONLY the NEW standing instructions since the last watch

B. (1) normally

(2) ALL current standing instructions

C. (1) should NOT

(2) ONLY the NEW standing instructions since the last watch

D. (1) should NOT

(2) ALL current standing instructions

68. Which ERFIS quality code (AND Color) indicates that an in-core thermocouple has failed due to an open circuit?

A. REDU (Red)

B. OPEN (White)

C. LWRN (Yellow)

D. DALM (Green)

69. Which ONE of the following identifies an ACCEPTABLE example of a troubleshooting activity in accordance with AP-929, Troubleshooting Guide?

A. Installing gags on valves

B. Pulling an annunciator card

C. Replacing failed components on circuit boards

D. Temporary M&TE "Test point /jack" connections

70. Given the following plant conditions:
   - The unit is operating at 100% power
   - Makeup to the 'C' SI Accumulator has just been completed
   - 'C' SI Accumulator parameters are as follows:

Boron Concentration	2419 ppm
Pressure	670 psig
Level	68%
1SI-248, Accum 'C' Disch Iso Valve	OPEN
Breaker 1A21-SA-3D, 1SI-248 Accum 'C' Dish	OFF

Based on the current conditions of the 'C' SI Accumulator, which ONE of the following describes the action required in accordance with Technical Specifications 3.5.1, Emergency Core Cooling System - Accumulators?

A. Restore Level to within limits within 1 hour.

B. Restore Boron concentration to within limits within 1 hour.

C. Restore Pressure to within limits within 1 hour.

D. Restore Disch Iso Valve Breaker to ON within 1 hour.

71. Which ONE of the following completes the statement below in accordance with OP-120.07, Waste Gas Processing?

The MAXIMUM allowed total curie content for Two Gas Decay Tanks cross-tied together is less than ______ curies.

A. 10,000

B. 20,000

C. 86,825

D. 105,000

72. Given the following plant conditions:
- A Refueling Outage is in progress
- You have been assigned to hang a clearance in the RCA, have been briefed, and are preparing to sign on to the RWP
- The survey map records the radiation levels as 1750 mRem/hour in the general area

Which ONE of the following completes the statements below?

The classification for this area in accordance with HPS-NGGC-0003, Radiological Posting, Labeling and Surveys, would be a __(1)__ High Radiation Area.

In accordance with OPS-NGGC-1301, Equipment Clearance, independent verification requirements may be waived by the __(2)__ if excessive radiation exposure would result.

A. (1) Very

(2) Control Room Supervisor

B. (1) Very

(2) Radiation Control Supervisor

C. (1) Locked

(2) Control Room Supervisor

D. (1) Locked

(2) Radiation Control Supervisor

73. Given the following plant conditions:
    - The Reactor has tripped and Safety Injection has actuated due to a Large Break Loss of Coolant Accident (LOCA).
    - The crew is implementing E-1, Loss of Reactor Or Secondary Coolant
    - The OAC reports the following for Critical Safety Function Status Trees:
        - Containment - Orange
        - Subcriticality - Orange
        - Heat Sink - Red
        - Integrity - Red
        - All others are Green

    Which ONE of the following identifies the required procedure transition AND what it is based on?

    A. FR-P.1, Response to Imminent Pressurized Thermal Shock, based on a Severe Challenge to the RPV Intergity

    B. FR-Z.1, Response to High Containment Pressure, based on an Severe Challenge to the Containment

    C. FR-S.2, Response to Loss of Core Shutdown, based on an Severe Challenge to the Subcriticality

    D. FR-H.1, Response to Loss of Secondary Heat Sink, based on an Severe Challenge to the Secondary Heat Sink

74. Given the following plant conditions:
    - AOP-036, Safe Shutdown Following a Fire, is being implemented
    - MCB level indicators LI-9010A1 SA & LI-9010B1 SB, CST Level, are not available

    Which ONE of the following completes the statement below?

    In accordance with AOP-036.02, Fire Area 1-A-BAL-A, 1-A-BAL-G, 1-A-BAL-H, the alternate method of checking CST level greater than 10% is to use _______.

    A. the local CST level indicator LI-9011

    B. a graph of AFW Pump suction pressure vs CST level

    C. a graph of Condensate Transfer Pump suction pressure vs CST level

    D. the annunciator ALB-017, 5-5, Condensate Storage Tank Low Minimum Level

75. Which ONE of the following completes the statements below in accordance with PEP-230, Control Room Operations?

During an event including an Alert or higher all NLO watch stations should report to the __(1)__ promptly after putting work in a safe conditions.

The __(2)__ must be informed when assigning additional duties to people who were already dispatched to perform another duty and have not yet returned from the first duty assignment.

A. (1) Operations Support Center

(2) Site Emergency Coordinator

B. (1) Operations Support Center

(2) Plant Operations Director

C. (1) Main Control Room

(2) Site Emergency Coordinator

D. (1) Main Control Room

(2) Plant Operations Director

76. Given the following plant conditions:
   - The unit is in Mode 6
   - Refueling Cavity Level is at 23' 6"
   - Both 'A' and 'B' RHR pumps were in operation in Shutdown Cooling mode when 'B' RHR pump trips on overcurrent

   Which ONE of the following completes the statement below in accordance with Technical Specification 3.9.8, Residual Heat Removal and Coolant Circulation?

   The MINIMUM RHR flowrate for the above conditions is __(1)__ gpm AND the basis for this flow requirement is to __(2)__.

   A. (1) 900

      (2) minimize the effect of a boron dilution incident and prevent boron stratification.

   B. (1) 900

      (2) preclude cavitation during RHR pump operation.

   C. (1) 2500

      (2) minimize the effect of a boron dilution incident and prevent boron stratification.

   D. (1) 2500

      (2) preclude cavitation during RHR pump operation.

77. Which ONE of the following completes BOTH of the statements below?

The loss of feedwater ATWS is the limiting ATWS event for the ___(1)___ fission product barrier.

In accordance with the FR-S.1 Background Document, for the loss of Feedwater ATWS event, the analysis assumes that the turbine is tripped within a MAXIMUM of ___(2)___ seconds.

A. (1) Reactor Coolant System

(2) 30

B. (1) Reactor Coolant System

(2) 60

C. (1) Containment

(2) 30

D. (1) Containment

(2) 60

78. Given the following plant conditions:
   - The plant was operating at 100%
   - 0600, 'C' SG develops a 15 gpm tube leak and CRS directs a plant shutdown in accordance with AOP-016, Excessive Primary Plant Leakage
   - 0610, 1MS-45, MS Line 'C' Safety relief valve, opens and cannot be shut
   - 0630, 'C' SG tube leakage degrades and a Reactor Trip and Safety Injection are initiated
   - 0645, Chemistry confirms an offsite release is in progress

   Which ONE of the following identifies (1) the FIRST required classification for the conditions above AND (2) the EARLIEST required time the State and Counties must be notified?

   (Reference provided)

   A. (1) FU1.1

      (2) 0625

   B. (1) FU1.1

      (2) 0710

   C. (1) SU8.1

      (2) 0615

   D. (1) SU8.1

      (2) 0700

79. Given the following plant conditions:
    - The crew transitioned from E-1, Loss of Reactor or Secondary Coolant to FR-H.1, Loss of Secondary Heat Sink
    - RCS Bleed and Feed was NOT initiated
    - Core exit TCs are stable
    - Containment pressure is 4.5 psig
    - 'A' CT Pump running, 'B' CT Pump is under clearance
    - Aux Feedwater flow has just been established at 250 KPPH
    - SG levels are as follows:
        - 'A' 39% Narrow range
        - 'B' 24% Narrow range
        - 'C' 29% Narrow range

    Which ONE of the following is (1) required in accordance with FR-H.1 AND (2) the reason?

    A. (1) Remain in FR-H.1

       (2) Because NONE of the SG Narrow range levels are greater than the minimum required for heat sink.

    B. (1) Remain in FR-H.1

       (2) Because NOT ALL SG Narrow range levels are greater than the minimum required for heat sink.

    C. (1) Transition to E-1, and step in effect

       (2) Adequate heat sink has been restored.

    D. (1) Transition to FR-Z.1, Response to High Containment Pressure

       (2) Only one CT Pump is running with adverse Containment conditions.

80. Given the following plant conditions:
- The unit was operating at 100% power when a loss of Offsite power occurred
- 6.9 KV Emergency Bus 1B-SB 86 lockout actuates
- EDG 'A' fails to start
- The ASI system is supplying RCP seal injection
- The crew is implementing ECA-0.0, Loss of All AC Power
- ECA-0.0, step 29 to initiate a cooldown to control PZR level using the SG PORVs is in progress

Which ONE of the following completes the statements below in accordance with ECA-0.0?

__(1)__ SG PORV(s) can be operated from the MCB.

The RCS cooldown is required to be stopped when __(2)__.

A. (1) All three

(2) all cold leg temperature are < 400°F

B. (1) All three

(2) the RCS pressure is < 700 psig

C. (1) ONLY the 'C'

(2) all cold leg temperature are < 400°F

D. (1) ONLY the 'C'

(2) the RCS pressure is < 700 psig

81. Given the following plant conditions:
    - The Unit is operating at 100% power.
    - The following PIC-1 loads have lost power
        - TE-413 RCS Hot Leg Temp Loop A
        - TE-423 RCS Hot Leg Temp Loop B
        - TE-433 RCS Hot Leg Temp Loop C

Which ONE of the following completes the statements below?

(consider each statement separately)

Based on the event above, in accordance with OST-1020, Remote Shutdown Monitoring And Accident Monitoring Instrumentation Channel Check Monthly Interval Modes 1-2-3, the RCS Subcooling Margin Monitor ___(1)___.

If at any time the subcooling monitor becomes inoperable, in accordance with Technical Specification, 3.3.3.6, Accident Monitoring Instrumentation, an acceptable backup method of calculating subcooling margin is to calculate it using ___(2)___ within 72 hours.

(Reference provided)

A. (1) remains operable

   (2) the CSFST graph

B. (1) remains operable

   (2) the OSI/PI computer

C. (1) is inoperable

   (2) the CSFST graph

D. (1) is inoperable

   (2) the OSI/PI computer

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82. Given the following plant conditions:
   - The unit is operating at 100% power
   - At 1200, Sept 13, 2013 a load rejection occurs
      - Th~~e OAC reports that~~ one group of control rods in Bank D failed to move and is ~~are~~ misaligned by ~~approximately~~ 20 steps
      - ALB-013-7-1, Rod Control Urgent Alarm, is in alarm
   - At 1215, Sept 13, 2013 all rods have been verified above the Rod Insertion limits

   Which ONE of the following completes the statements below?

   In accordance with Technical Specification 3.1.3.1, Movable Control Assemblies - Group Height, the MOST limiting action required is to place the unit in Hot Standby prior to _______.

   (Reference provided)

   A. 1900, Sept 13, 2013

   B. 1800, Sept 13, 2013

   C. 0000, Sept 15, 2013

   D. 0600, Sept 15, 2013

83. Given the following plant conditions:
   - The unit is operating at 100% power
   - A release of WGDT 'E' is in progress
   - ALB-010-4-5, Rad Monitor System Trouble, alarms
   - The RM-11 status display screens are provided as a reference
   - The actions for AOP-005, Radiation Monitoring System, have been completed
   - The CRS has entered AOP-009, Accidental Release of Waste Gas
   - It is desired to continue with the Waste Gas Decay Tank release

Which ONE of the following (1) describes the status of REM-3546 PIG (4GG793), AND (2) in accordance with ODCM 3.3.3.11, Radioactive Gaseous Effluent Monitoring Instrumentation, what are the MINIMUM actions required?

(Reference provided)

A. (1) Inoperable - equipment failure monitor loss of isokinetic flow is present.

   (2) samples, release rate calcs, and the valve line-up are Independently Verified

B. (1) Inoperable - equipment failure monitor loss of isokinetic flow is present.

   (2) samples, release rate calcs, and the valve line-up are Independently Verified AND once per 12 hours grab samples are analyzed for radioactivity within 24 hours

C. (1) Inoperable - operate failure monitor loss of sample flow is present.

   (2) samples, release rate calcs, and the valve line-up are Independently Verified

D. (1) Inoperable - operate failure monitor loss of sample flow is present.

   (2) samples, release rate calcs, and the valve line-up are Independently Verified AND once per 12 hours grab samples are analyzed for radioactivity within 24 hours

84. Given the following plant conditions:
- The unit is operating at 100% power
- One of the MSL rad monitors becomes inoperable

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Which ONE of the following identifies (1) the normal indication for MSL Radiation Monitors AND (2) the pre-planned alternate method of montioring the Main Stean lines in accordance with Technical Specification, 3.3.3.6, Accident Monitoring Instrumentation, and OWP-RM-09, Radiation, Effluent, And Explosive Gas Monitoring?

(Reference provided)

A. (1) 0.35 mRem/hour

(2) TB Vent Stack and CVPETS Rad monitor

B. (1) 0.35 mRem/hour

(2) SGBD rad monitor

C. (1) 1.05 mRem/hour

(2) TB Vent Stack and CVPETS Rad monitor

D. (1) 1.05 mRem/hour

(2) SGBD rad monitor

85. Given the following plant conditions:
- The plant is operating in Mode 3
- At 0900 on Sept 1st, the Personnel Air Lock (PAL), Inner door seal fails
- At 0800 on Sept 3rd, the Emergency Air Lock (EAL), Outer door seal fails

Which ONE of the following completes the statements below in accordance with Technical Specification 3.6.1.3, Containment Air Locks, and its Bases?

The latest day/time that either of the air locks can be use for entry/exit under administrative controls is __(1)__.

In accordance with the Technical Specification 3.6.1.3, Bases, during this period of time, the use of the air lock to perform ONLY non-Technical Specification required activites or repairs on ONLY non-vital plant equipment is __(2)__ in Containment.

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(Reference provided)

A. (1) 0900 on September 8th

(2) allowed

B. (1) 0900 on September 8th

(2) not allowed

C. (1) 0800 on September 10th

(2) allowed

D. (1) 0800 on September 10th

(2) not allowed

86. Given the following plant conditions:
- 'A' MDAFW pump is unavailable due to a motor problem
- 'B' Main Steam Line radiation monitor is in HIGH alarm
- The crew trips the Reactor and actuates Safety Injection due to lowering PZR level
- After the Reactor Trip, one 'B' SG safety valve stuck open
- An 86 lockout occurs on the 1B-SB 6.9KV Emergency Bus
- MSIV's will not close
- 'B' SG narrow range level is 30% and rising following isolation of feed

Which ONE of the following completes the statement below?

1MS-70, Main Steam B To Aux Fw Turbine, __(1)__, AND __(2)__ will direct this for the given conditions.

A. (1) must remain open

(2) E-2, Faulted Steam Generator Isolation

B. (1) is required to be closed

(2) E-2, Faulted Steam Generator Isolation

C. (1) must remain open

(2) E-3, Steam Generator Tube Rupture

D. (1) is required to be closed

(2) E-3, Steam Generator Tube Rupture

87. Given the following plant conditions:
    - The unit is operating at 100% power
    - A LOCA occurs
    - The crew is implementing E-1, Loss of Reactor or Secondary Coolant
    - RCS pressure is 675 psig and slowly lowering
    - ALB-004, 2-2, Refueling Water Storage Tank Low Level, is in alarm
    - ALB-004, 2-3, Refueling Water Storage Tank Low Low Level Alert, is NOT in alarm
    - Safety Injection has been reset
    - Subsequently, a loss of offsite power occurs

    Which ONE of the following (1) identifies the required transition AND (2) the attachment used to verify proper configuration of safeguards equipment following the loss of offsite power?

    A. (1) ES-1.3, Transfer To Cold Leg Recirculation

       (2) E-0, Attachment 6, Safeguards Equipment Realignment Following A Loss Of Offsite Power

    B. (1) ES-1.2, Post LOCA Cooldown And Depressurization

       (2) E-0, Attachment 6, Safeguards Equipment Realignment Following A Loss Of Offsite Power

    C. (1) ES-1.3, Transfer To Cold Leg Recirculation

       (2) E-0, Attachment 8, Response To Loss of Offsite Power to AC Emergency After SI Actuation

    D. (1) ES-1.2, Post LOCA Cooldown And Depressurization

       (2) E-0, Attachment 8, Response To Loss of Offsite Power to AC Emergency After SI Actuation

88. Given the following plant conditions:
  - The unit is operating at 100% power
  - At 0800 the 1B-SB Emergency Battery has been declared inoperable due to a failure of the 1B-SB battery charger
  - At 0830 an electrician performing the weekly maintenance surveillance test for the 1A-SA Emergency Battery reports following pilot cell indications:
    - electrolyte level is midway between the minimum and maximum marks
    - float voltage is 2.10 volts
    - specific gravity is 1.198

Which ONE of the following completes BOTH of the statements below?

In accordance with Technical Specification 3.8.2.1, D. C. Sources - Operating, the 1A-SA battery is ___(1)___.

Based on the conditions provided above, the MINIMUM required action is to ___(2)___.

(Reference provided)

A. (1) operable

(2) place the 1A-SB battery charger in service prior to 1000

B. (1) operable

(2) place the 1A-SB battery charger in service prior to 1030

C. (1) inoperable

(2) enter Technical Specification 3.0.3 at 0930

D. (1) inoperable

(2) enter Technical Specification 3.0.3 at 1000

89. Given the following plant conditions:
    - The unit is operating at 65% power
    - The following annunciator is received in the Control Room:
        - ALB-002-7-2, Serv Wtr Pumps Discharge Low Press
    - The BOP notes that Cooling Tower Basin Level is lowering rapidly
    - Service Water header pressure is 50 psig and lowering

    One minute later
    - Service Water header pressure is 35 psig and continues to lower
    - CTMU cannot maintain Cooling Tower Basin level
    - The Cooling Tower Basin Level continues to lower
    - The RAB AO reports that a large volume of water is gushing from the downstream flange of 1SW-276, Headers A & B Return To Normal SW Header valve

    Which ONE of the following completes the statements below?

    The leak is located in the ___(1)___ system.

    In accordance with Technical Specification 3.7.4, Emergency Service Water, the bases for the Limiting Condition of Operation is to ensure that sufficient cooling capacity is available for continued operation of safety related equipment during ___(2)___ conditions.

    A. (1) Normal Service Water

       (2) normal AND accident

    B. (1) Normal Service Water

       (2) ONLY accident

    C. (1) Emergency Service Water

       (2) normal AND accident

    D. (1) Emergency Service Water

       (2) ONLY accident

90. Given the following plant conditions:
   - The unit is operating at 100% power
   - A loss of power to Safety Bus 1B-SB occurs
   - The 'B' EDG fails to start

   Which ONE of the following describes (1) the effect on the plant AND (2) the Technical Specification requirements that currently apply?

   (Reference provided)

   A. (1) A Containment Ventilation Isolation Signal will be generated

   (2) Restore the 'B' Train Containment vacuum breaker in 72 hours or be in at least HOT STANDBY within the next 6 hours.

   B. (1) A Containment Ventilation Isolation Signal will be generated

   (2) Be in at least HOT STANDBY within the next 7 hours.

   C. (1) A Containment Ventilation Isolation Signal will NOT be generated

   (2) Restore the 'B' Train Containment vacuum breaker in 72 hours or be in at least HOT STANDBY within the next 6 hours.

   D. (1) A Containment Ventilation Isolation Signal will NOT be generated

   (2) Be in at least HOT STANDBY within the next 7 hours.

91. Given the following plant conditions:
    - The crew is implementing E-1, Loss Of Reactor Or Secondary Coolant
    - Plant conditions are as follows:
        - CNMT pressure - 12.6 psig
        - RCS Hot leg temperature - 650°F
        - The five hottest core exit thermocouples are:
            A08 - 1201°F
            B05 - 1208°F
            G02 - 857°F
            H15 - 753°F
            L14 - 734°F
        - RCS pressure - 200 psig
        - RVLIS Full Range level - 40%
        - The SPTOP and CSFST displays are NOT available on ERFIS

    Which ONE of the following identifies (1) the requirement for FR-C.1, Response to Inadequate Core Cooling AND (2) the status of the Fuel Clad Barrier in accordance with EP-EAL?

    (Reference provided)

    A. (1) required to be implemented

       (1) Loss of Fuel Clad Barrier

    B. (1) required to be implemented

       (2) Potenitial Loss of Fuel Clad Barrier

    C. (1) NOT required to be implemented

       (2) Loss of Fuel Clad Barrier

    D. (1) NOT required to be implemented

       (2) Potential Loss of Fuel Clad Barrier

92. Given the following plant conditions:
- 1000 A General Emergency has been declared due to a LOCA
- 1015 RVLIS Full Range is 35% and lowering
  - RCS Pressure is 100 psig
  - Core Exit Theromcouple temperature is 694°F
- 1115 The hydrogen monitoring system and recombiners were placed in service in accordance with E-1 and OP-125, Post Accident Hydrogen System
- 1200 Due to a malfunction of the recombiners, the containment hydrogen concentration is now 6%
  - RVLIS Full Range is 34%
  - RCS Pressure is 80 psig
  - Core Exit Theromcouple temperature is 712°F

Which ONE of the following completes the statements below regarding the hydrogen in containment?

The containment hydrogen monitoring system is designed with an intermittent cycle of hydrogen indication for ___(1)___ different sample points in containment.

The required Protective Action Recommendation is to evacuate a ___(2)___ mile radius.

(Reference Provided)

A. (1) Three

(2) 2

B. (1) Three

(2) 5

C. (1) Six

(2) 2

D. (1) Six

(2) 5

93. Given the following plant conditions:
  - A batch release of the Secondary Waste Sample Tank is in progress
  - A HIGH ALARM is received on REM-21WS-3542, Secondary Waste Sample Tank Pump discharge radiation monitor; however the release failed to AUTO terminate

Which ONE of the following completes the statements below?

In accordance with ODCM 3.11.1.1, Liquid Effluents - Concentration, the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS shall be limited to __(1)__ times the concentrations specified in 10 CFR Part 20.

In accordance with the ODCM 3.3.3.10, Monitoring Instrumentation - Radioactive Liquid Effluent Monitoring Instrumentation, with REM-21WS-3542 inoperable, this release may continue from this pathway provided that __(2)__ .

(Reference provided)

A. (1) 10

(2) samples, release rate calcs, and the valve line-up are Independently Verified

B. (1) 10

(2) once per 12 hours grab samples are analyzed for radioactivity at a LLD

C. (1) 20

(2) samples, release rate calcs, and the valve line-up are Independently Verified

D. (1) 20

(2) once per 12 hours grab samples are analyzed for radioactivity at a LLD

94. Given the following plant conditions:
    - A core off load is in progress to support a refueling outage in accordance with FHP-014, Fuel and Insert Shuffle Sequence
    - A fuel assembly has just been latched and raised for serial number verification
    - The serial number on Attachment 2, Core Offload/Reload Fuel Transfer Data Sheet, does NOT match the serial number on the fuel assembly

    Which ONE of the following identifies the action(s) required by FHP-014, with regard to the latched fuel assembly?

    A. Lower the fuel assembly in the location it was removed from AND unlatch

    B. Move the fuel assembly to the temporary storage location AND unlatch

    C. Lower the fuel assembly in the location it was removed from but do NOT unlatch

    D. Move the fuel assembly to the temporary storage location but do NOT unlatch

95. Which ONE of the following choices completes the statements below?

OPS NGGC-1301, Equipment Clearance, requires that the ground checklist be authorized by a(the) ___(1)___

___(2)___ verification is required for ground installation.

A. (1) Senior Reactor Operator

(2) Concurrent

B. (1) Senior Reactor Operator

(2) Independent

C. (1) Electrical Maintenance Supervisor

(2) Concurrent

D. (1) Electrical Maintenance Supervisor

(2) Independent

96. Given the following plant conditions:
   - The unit at 100% power
   - At 09:00 on Sept 8, 2013, the the A-SA EDG Fuel Oil Transfer pump was placed under clearance to repair a fuel oil leak
   - At 11:00 on the same day, a fault in the control power circuit for the B-SB Containment Spray pump causes the control power fuses to blow

   Assuming no additional changes to equipment operability which ONE of the following identifies, when the unit must enter Mode 3 in accordance with Technical Specifications?

   (Reference provided)

   A. 1800 on Sept 8, 2013

   B. 2200 on Sept 8, 2013

   C. 1500 on Sept 11, 2013

   D. 1700 on Sept 11, 2013

97. Given the following plant conditions:
- An employee was injured and contaminated
- The employee was transported to Western WakeMed for treatment before he was de-contaminated
- Duke Energy Progress is planning a news release for this event

Which ONE of the following completes the statements below?

In accordance with AP-617, Reportability Determination And Notification, the EARLIEST required NRC notification of this event is within __(1)__ hours.

In accordance with AOP-013 __(2)__ is the primary radiological concern for fuel off-loaded more than 6 months ago because it will NOT be detected by personal dosimetry or area radiation monitors.

(Reference provided)

A. (1) 4

(2) Krypton-85

B. (1) 4

(2) Iodine-131

C. (1) 8

(2) Krypton-85

D. (1) 8

(2) Iodine-131

98. Which ONE of the following completes the statements below in accordance with PEP-330, Radiological Consequences, Attachment 1, Limitations for Lifesaving and Emergency Reentry/Repair Actions?

Emergency worker exposures during life saving missions should be limited to __(1)__ REM TEDE.

Exposures in excess of 5 REM TEDE shall not be permitted unless specifically authorized by the __(2)__ .

A. (1) 15

(2) Emergency Response Manager

B. (1) 15

(2) Site Emergency Coordinator

C. (1) 25

(2) Emergency Response Manager

D. (1) 25

(2) Site Emergency Coordinator

99. Which ONE of the completes the statements below in accordance with PEP-230, Control Room Operations?

The Emergency Response Organization (ERO) accountability process must be completed within a MAXIMUM of ___(1)___ from the time the Site Area Emergency was declared.

The SEC-MCR's task of making Offsite Protective Action Recommendations (PARs) ___(2)___ be delegated to the TSC.

A. (1) 30 minutes

(2) can NOT

B. (1) 30 minutes

(2) can

C. (1) 60 minutes

(2) can NOT

D. (1) 60 minutes

(2) can

100. Given the following plant conditions:
- The plant is operating at 100% power
- The OSI/PI and ERFIS server are Out of Service for a software update

- At 0800 the following occurs:
  - ALB-026 /1-4, Annun Sys 1 Power Supply Failure
  - ALB-003 / 4-5, Annunciator System 2 Power Supply Failures
  - The OAC reports 23 of the 30 Main Control Board ALBs have lost annunciators
  - The AO reports the both Annuciator System 1 and Annunciator System 2 have lost multiple 125 VDC power supplies

- At 0900 the following occurs:
  - ALB-019, 3-2A, HTR DRN Pump B O/C TRIP-GND
  - The HTR DRN Pump B trips

Which ONE of the following completes both statements?

At 0800 the HIGHEST required classification is __(1)__.

At 0900 the HIGHEST required classification is __(2)__.

(Reference provided)

A. (1) Unusual Event

(2) Unusual Event

B. (1) Unusual Event

(2) Alert

C. (1) Alert

(2) Alert

D. (1) Alert

(2) Site Area Emergency

You have completed the test!

2013 ILC NRC Exam

DUKE ENERGY PROGRESS

HARRIS TRAINING SECTION

EXAM NUMBER: 2013 NRC LESSON/COURSE CODE: SO6C03H

SUBJECT/CATEGORY: SRO Written EXAM POINT VALUE: 100

STUDENT NAME (PLEASE PRINT): ____________________

DATE: ____________________ SSN: ____________________

Prepared by: Archie Lucky / J.R. Horton DATE: 9/25/2013

Exam Validation by: Mike Matheny and Kyle Kelly DATE: 9/04/2013

APPROVED BY: Simon Schwindt DATE: 9/06/2013
SUPERVISOR OR DESIGNEE

ALL WORK DONE ON THIS EXAM (INCLUDING CORRECTIONS) IS MY OWN. I HAVE NEITHER GIVEN NOR RECEIVED AID.

I AGREE THAT I WILL NOT DIVULGE ANY INFORMATION WITH REGARDS TO THE CONTENT OF THIS EXAMINATION TO ANY UNAUTHORIZED PERSONNEL.

SIGNATURE: ____________________ DATE ________

GRADE: ______ GRADED BY: ____________________ DATE ________

GRADE VERIFICATION: ____________________ DATE ________

References and/or tools provided for use with the examination include: (list below)

- Calculator
- Steam Tables / Mollier Diagram
- Curve No. F-18-1, Rev. 0
- Curve No. F-X-20, Rev. 1
- AP-617, Rev. 33
- EAL Matrix, Rev. 10
- ECA-1.1 step 7.b, pg 6, Rev. 0
- ODCM 3.3.3.10 pg D-2 thru D-4
- ODCM 3.3.3.11 pg D-7 thru D-9
- PEP-110, Rev. 22
- RM-11 Status Display Screen print (2)
- T.S. 3.1.3.1 pg 3/4 1-14 thru 1-16
- T.S. 3.3.3.6 pg 3/4 3-66 thru 3-69
- T.S. 3.6.1.3 pg 3/4 6-4 thru 6-5
- T.S. 3.6.2.1 pg 3/4 6-11
- T.S. 3.6.5 pg 3/4 6-32
- T.S. 3.8.1.1 pg 3/4 8-1 thru 8-4
- T.S. 3.8.2.1 pg 3/4 8-12 thru 8-14

QA/VITAL RECORD

2013 NRC SRO Question 81 (6) Reference

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels, except In Core Thermocouples and Reactor Vessel Level, less than the Total Required Number of Channels requirements shown in Table 3.3-10 restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the Pressurizer Safety Valve Position Indicator, the Reactor Coolant System Subcooling Margin Monitor, In Core Thermocouples or Reactor Vessel Level, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE accident monitoring instrumentation channels for the radiation monitor(s), the Pressurizer Safety Valve Position Indicator*, or the Reactor Coolant System Subcooling Margin Monitor#, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within the next 14 days, that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status.

d. With the number of OPERABLE accident monitoring instrumentation channels for In Core Thermocouples or Reactor Vessel Level less than the total required number of channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report, pursuant to specification 6.9.2, within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

e. With the number of OPERABLE accident monitoring instrument channels for In Core Thermocouples or Reactor Vessel Level less than the minimum channels OPERABLE requirement of Table 3.3-10, either restore one channel to OPERABLE status within 7 days or be in at

LIMITING CONDITION FOR OPERATION

least HOT STANDBY in the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

---

* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

\# The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Narrow Range	2	1
b. Wide Range	2	1
2. Reactor Coolant Hot-Leg Temperature--Wide Range	2	1
3. Reactor Coolant Cold-Leg Temperature--Wide Range	2	1
4. Reactor Coolant Pressure--Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level--Narrow Range	N.A.	1/steam generator
8. Steam Generator Water Level--Wide Range	N.A.	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Auxiliary Feedwater Flow Rate	N.A.	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	N.A.	1
12. PORV Position Indicator*	N.A.	1/valve
13. PORV Block Valve Position Indicator**	N.A.	1/valve
14. Pressurizer Safety Valve Position Indicator	N.A.	1/valve
15. Containment Water Level (ECCS Sump)--Narrow Range	2	1
16. Containment Water Level--Wide Range	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
17. In Core Thermocouples	4/core quadrant	2/core quadrant
18. Plant Vent Stack--High Range Noble Gas Radiation Monitor	N.A.	1
19. Main Steam Line Radiation Monitors	N.A.	1/steam line
20. Containment -High Range Radiation Monitor	N.A.	1
21. Reactor Vessel Level	2	1
22. Containment Spray NaOH Tank Level	2	1
23. Turbine Building Vent Stack High Range Noble Gas Radiation Monitor	N.A.	1
24. Waste Processing Building Vent Stack High Range Noble Gas Radiation Monitors		
a. Vent Stack 5	N.A.	1
b. Vent Stack 5A	N.A.	1
25. Condensate Storage Tank Level	2	1

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power is removed.

2013 NRC SRO Question 82 (7) Reference

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With more than one rod inoperable, due to a rod control urgent failure alarm or obvious electrical problem in the rod control system existing for greater than 36 hours, be in HOT STANDBY within the following 6 hours.

d. With one rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

2013 NRC SRO Question 82 (7) Reference

ACTION (Continued):

remain valid for the duration of operation under these conditions;

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

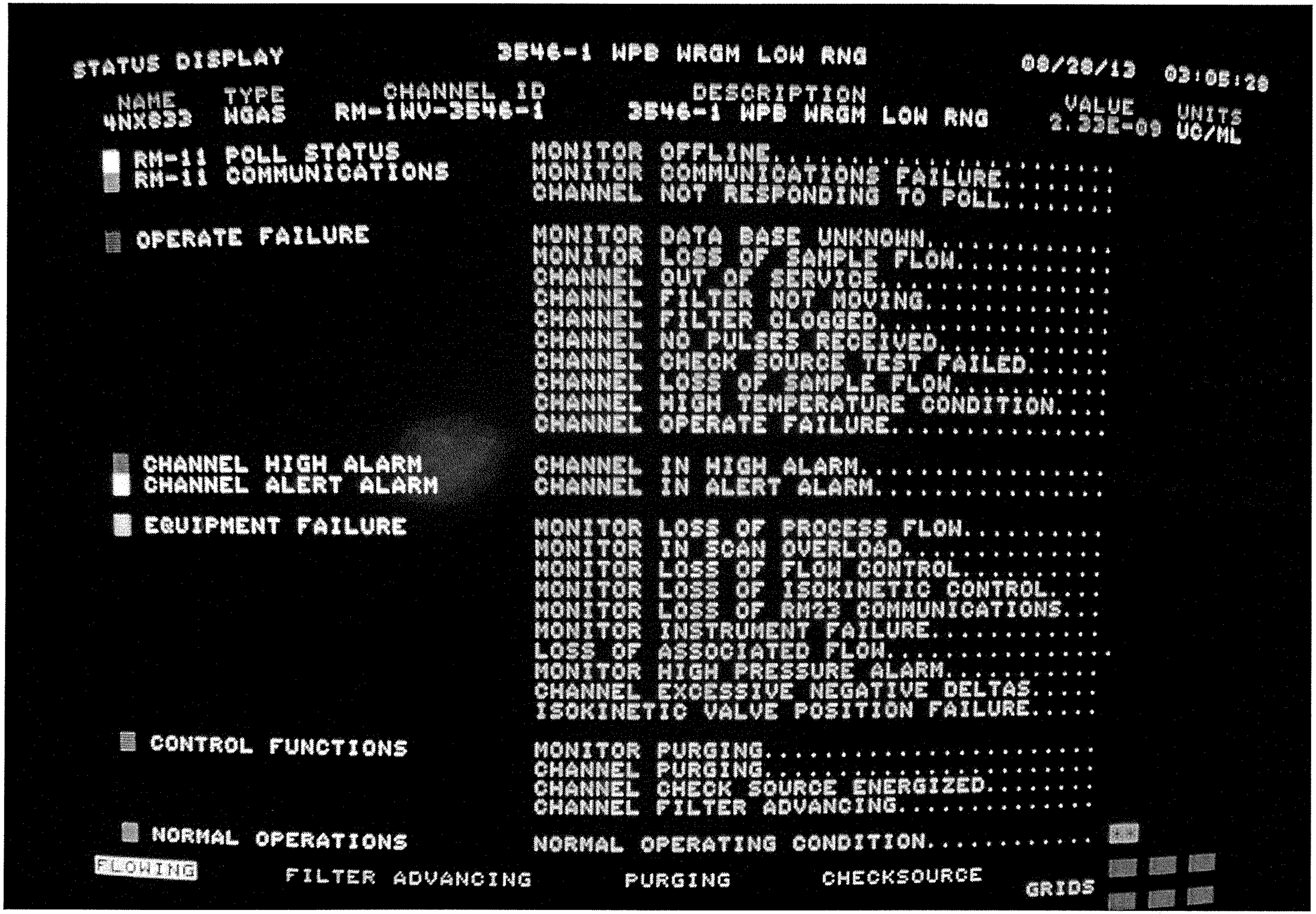
Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

2013 NRC SRO Question 83 (8) Reference

2013 NRC SRO Question 83 (8) Reference



3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.11 Radioactive Gaseous Effluent Monitoring Instrumentation

OPERATIONAL REQUIREMENT

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Operational Requirements 3.11.2.1 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Operational Requirement 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above Operational Requirement, immediately (1) suspend the release of radioactive gaseous effluents monitored by the affected channel or (2) declare the channel inoperable and take ACTION as directed by b. below.

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instrument to OPERABLE status within 30 days. If unsuccessful, explain in the next Annual Radioactive Effluent Release Report pursuant to ODCM, Appendix F, Section F.2 why this inoperability was not corrected in a timely manner.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and a DIGITAL CHANNEL OPERATIONAL TEST or an ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

2013 NRC SRO Question 83 (8) Reference

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MIN. CHANNELS OPERABLE	APPLICABILITY	ACTION
1.	GASEOUS WASTE PROCESSING SYSTEM HYDROGEN AND OXYGEN ANALYZERS			
	Specification is not used in ODCM			
2.	TURBINE BUILDING VENT STACK			
a.	Noble Gas Activity Monitor	1	*	47
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46
3.	PLANT VENT STACK			
a.	Noble Gas Activity Monitor	1	*	47
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46
4.	WASTE PROCESSING BUILDING VENT STACK 5			
a.1	Noble Gas Activity Monitor (PIG)	1	*	45, 51
a.2	Noble Gas Activity Monitor (WRGM)	1	MODES 1, 2, 3	52
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46
5.	WASTE PROCESSING BUILDING STACK 5A			
a.	Noble Gas Activity Monitor	1	*	47
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46

TABLE NOTATIONS

* At all times.

TABLE 3.3-13 (Continued)

ACTION STATEMENTS

ACTION 45 - With the number channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the waste gas decay tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - Not Used in the ODCM.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 50 - Not used in the ODCM.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement for both the PIG and WRGM, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 52 - With the number of OPERABLE accident monitoring instrumentation channels for the radiation monitor(s) less than the Minimum Channels OPERABLE requirements of Technical Specification Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission, pursuant to Technical Specification 6.9.2, within the next 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels, except In Core Thermocouples and Reactor Vessel Level, less than the Total Required Number of Channels requirements shown in Table 3.3-10 restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the Pressurizer Safety Valve Position Indicator, the Reactor Coolant System Subcooling Margin Monitor, In Core Thermocouples or Reactor Vessel Level, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE accident monitoring instrumentation channels for the radiation monitor(s), the Pressurizer Safety Valve Position Indicator*, or the Reactor Coolant System Subcooling Margin Monitor#, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within the next 14 days, that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status.

d. With the number of OPERABLE accident monitoring instrumentation channels for In Core Thermocouples or Reactor Vessel Level less than the total required number of channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report, pursuant to specification 6.9.2, within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

e. With the number of OPERABLE accident monitoring instrument channels for In Core Thermocouples or Reactor Vessel Level less than the minimum channels OPERABLE requirement of Table 3.3-10, either restore one channel to OPERABLE status within 7 days or be in at

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

least HOT STANDBY in the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

---

* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Narrow Range	2	1
b. Wide Range	2	1
2. Reactor Coolant Hot-Leg Temperature--Wide Range	2	1
3. Reactor Coolant Cold-Leg Temperature--Wide Range	2	1
4. Reactor Coolant Pressure--Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level--Narrow Range	N.A.	1/steam generator
8. Steam Generator Water Level--Wide Range	N.A.	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Auxiliary Feedwater Flow Rate	N.A.	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	N.A.	1
12. PORV Position Indicator*	N.A.	1/valve
13. PORV Block Valve Position Indicator**	N.A.	1/valve
14. Pressurizer Safety Valve Position Indicator	N.A.	1/valve
15. Containment Water Level (ECCS Sump)--Narrow Range	2	1
16. Containment Water Level--Wide Range	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
17. In Core Thermocouples	4/core quadrant	2/core quadrant
18. Plant Vent Stack--High Range Noble Gas Radiation Monitor	N.A.	1
19. Main Steam Line Radiation Monitors	N.A.	1/steam line
20. Containment -High Range Radiation Monitor	N.A.	1
21. Reactor Vessel Level	2	1
22. Containment Spray NaOH Tank Level	2	1
23. Turbine Building Vent Stack High Range Noble Gas Radiation Monitor	N.A.	1
24. Waste Processing Building Vent Stack High Range Noble Gas Radiation Monitors		
a. Vent Stack 5	N.A.	1
b. Vent Stack 5A	N.A.	1
25. Condensate Storage Tank Level	2	1

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power is removed.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Two containment air locks shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

. . . . Notes . . . .

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. A separate ACTION is allowed for each air lock.

3. Enter 3.6.1.1 LCO for "Containment Integrity" when the air lock leakage results in exceeding the overall containment leakage rate, Specification 3.6.1.2.a.

4. Locking a Personnel Air Lock door shut consists of locking the associated manual pumping stations and deactivating the electronic mechanisms used to open a Personnel Air Lock door once the associated air lock door is shut. Locking an Emergency Air Lock door shut consists of locking the mechanical operator.

. . . .

a. One or more containment air locks with one containment air lock door inoperable:#

   1. Within one hour, verify the OPERABLE door is closed in the affected air lock, and

   2. Within 24 hours, lock the OPERABLE door closed in the affected air lock, and

   3. Once per 31 days, verify the OPERABLE door is locked closed in the affected air lock*, or

   4. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

\# 1. ACTIONS 3.6.1.3.a.1, 3.6.1.3.a.2, 3.6.1.3.a.3, and 3.6.1.3.a.4 are not applicable if both doors in the same air lock are inoperable and ACTION 3.6.1.3.c is entered.

2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable.

* Air lock doors in high radiation areas may be verified closed by administrative means.

2013 NRC SRO Question 85 (10) Reference

LIMITING CONDITION FOR OPERATION

b. One or more containment air locks with containment air lock interlock mechanism inoperable.##

1. Within one hour, verify an OPERABLE door is closed in the affected air lock, and
2. Within 24 hours, lock an OPERABLE door closed in the affected air lock, and
3. Once per 31 days, verify the OPERABLE door is locked closed in the affected air lock*, or
4. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. One or more containment air locks inoperable for reasons other than 3.6.1.3.a or 3.6.1.3.b.

1. Immediately initiate action to evaluate overall containment leakage rate per LCO 3.6.1.2, and
2. Within one hour, verify a door is closed in the affected air lock, and
3. Within 24 hours, restore air lock to OPERABLE status, or
4. Otherwise be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

\## 1. ACTIONS 3.6.1.3.b.1, 3.6.1.3.b.2, 3.6.1.3.b.3, and 3.6.1.3.b.4 are not applicable if both doors in the same air lock are inoperable and ACTION 3.6.1.3.c is entered.

2. Entry and exit of containment is permissible under the control of a dedicated individual.

* Air lock doors in high radiation areas may be verified closed by administrative means.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE by:

a. Performing required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions###. The acceptance criteria for air lock testing are:

1. Overall air lock leakage rate is $\leq .05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq .01\ L_a$ when tested at $\geq P_a$.

b. At least once per 6 months by verifying that only one door in the air lock can be opened at a time**.

1. An inoperable air lock door does not invalidate the previous successful performance of the overall airlock leakage test.

2. Results shall be evaluated against Specification 3.6.1.2.a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

** Only required to be performed upon entry or exit through the containment air lock. (If Surveillance Requirement 4.6.1.3.b has not been performed in the last 6 months, then perform Surveillance Requirement 4.6.1.3.b during the next containment entry through the associated air lock.)

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Emergency Battery Bank 1A-SA and either full capacity charger, 1A-SA or 1B-SA, and,

b. 125-volt Emergency Battery Bank 1B-SB and either full capacity charger, 1A-SB or 1B-SB.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt Emergency Battery and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8-2 meet the Category A limits, and

   2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1. The parameters in Table 4.8-2 meet the Category B limits,

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and

   3. The average electrolyte temperature of 10 connected cells is above 70° F.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

4. The battery charger will supply at least 150 amperes at greater than or equal to 125 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

2013 NRC SRO Question 88 (13) Reference

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts (6)	> 2.07 volts
Specific Gravity(4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

3/4.6.5 VACUUM RELIEF SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5 The containment vacuum relief system shall be OPERABLE with an Actuation Setpoint of equal to or less negative than -2.5 inches water gauge differential pressure (containment pressure less atmospheric pressure).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one containment vacuum relief system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 No additional requirements other than those required by the Inservice Testing Program.

Progress Energy

R
REFERENCE
USE

HARRIS NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME 2

PART 5

PROCEDURE TYPE: PLANT EMERGENCY PROCEDURE

NUMBER: **PEP-110**

TITLE: **Emergency Classification and Protective Action Recommendations**

Table of Contents

1.0 PURPOSE

1. The purpose of this procedure is to provide guidance on the use of Emergency Action Levels (EALs) for classifying an emergency. This implements Section 4.1 of PLP-201.

2. This procedure provides guidelines for determining Protective Action Recommendations (PARs) to be made to offsite authorities during a General Emergency. This implements Section 4.5 of PLP-201.

3. This procedure provides guidance for summarizing events and actions taken during an event for use during facility turnover and facility briefings. This implements Section 2.3 of PLP-201.

4. This procedure provides guidance for event termination and entry into Recovery. This implements Section 6.7 of PLP-201.

2.0 INITIATING CONDITIONS

1. Entry into the Emergency Action Level (EAL) Matrix has been directed by any of the Emergency Operating Procedures, Fire Protection Procedures, Abnormal Operating Procedures, or any other procedure.

2. A Critical Safety Function Status Tree (CSFST) on the Safety Parameter Display System has produced a valid red or orange output and monitoring of the CSFSTs has been authorized in accordance with an approved procedure.

3. Notification has been received from a member of the Security Organization that a Security Condition, Threat, or Hostile Action has occurred.

4. Conditions exist which, in the judgment of the Shift Manager (SM), could be classified as an emergency.

3.0 GENERAL

NOTE:	The Current revision of the EAL Matrix is located in EP-EAL. Large print versions of the EAL Matrix are located in the Main Control Room, Technical Support Center and Emergency Operations Facility.

3.1. General Guidelines for Use of the EAL Matrix

1. All emergency classifications shall be made within 15 minutes following indications that conditions have reached an EAL threshold, based upon valid indications, reports or conditions. An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

2. Dose projections are used during the evaluation of EALs. When the dose assessment is complete – the clock starts. This is the **Run date/time** on the Dose Assessment Summary Report. It is up to the Dose Assessment Team Leader, Dose Assessment Team, and the RCM to validate assumptions and results, and report those results to the ERM within this 15-minute period, so they may communicate to the SEC for emergency event classification.

3. The primary tool for determining the emergency classification level is the EAL Matrix. EP-EAL, Emergency Action Levels is used in conjunction with this procedure and the EAL Matrix when classifying events. EP-EAL provides the technical basis and additional explanatory material to correctly classify events.

4. Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier.

    a. "Loss" means the barrier no longer assures containment of radioactive materials.

    b. "Potential loss" infers an increased probability of barrier loss and decreased certainty of maintaining the barrier.

5. To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

3.1 General Guidelines for Use of the EAL Matrix (continued)

6. The requirement is that emergency classifications are to be made as soon as conditions are present for the classification, but within 15 minutes in all cases of conditions present.

7. Where possible, the EALs have been made consistent with and utilize the conditions defined in the HNP Emergency Operating Procedure (EOP) network. While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened.

8. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

9. Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

10. There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

11. The highest emergency class for which an Emergency Action Level was exceeded shall be declared.

    a. Only one Emergency Action Level (EAL) classification shall be made at a time.

    b. If two EALs are clearly met, then choose the EAL of highest classification level as determined by the EAL matrix.

3.1 General Guidelines for Use of the EAL Matrix (continued)

12. If the plant condition degrades and a higher classification emergency is declared before the notifications are made for the lesser emergency declaration, update the notification to reflect the higher emergency classification and complete the updated notifications within the 15 minutes of the *lesser* emergency declaration. [RIS 2007-02]

13. If the notification cannot be updated and completed within 15 minutes of the lesser emergency declaration, make the notification for the lesser emergency declaration within 15 minutes of its declaration with a caveat that explains a change in classification is forthcoming. [RIS 2007-02]

14. In parallel, prepare the notification for the higher emergency classification and make the notification for the higher emergency classification within 15 minutes of the classification time of the higher emergency declaration. [RIS 2007-02]

3.2. Specific Rules for Use of the EAL Matrix

1. The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

2. For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

3. Although the majority of the EALs provide very specific thresholds, the Site Emergency Coordinator (SEC) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SEC, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

4. The EAL Matrix should be read from left to right and top to bottom.

3.3. Protective Action Recommendations (PARs) General Guidance

1. PARs are made by HNP personnel whenever a General Emergency is declared. Additionally, if in the opinion of the Emergency Response Manager, or the SEC-CR if the EOF is not yet activated, conditions warrant the issuance of PARs, a General Emergency will be declared (HNP will not issue PARs for any accident classified below a General Emergency).

2. PARs provided in response to a radioactive release include evacuation, taking shelter and consideration of the use of KI.

   a. Evacuation is the preferred action unless external conditions impose a greater risk from the evacuation than from the dose received.

   b. HNP personnel do not have the necessary information on external factors to determine whether offsite conditions would require sheltering instead of an evacuation. Therefore, an effort to base PARs on external factors (such as road conditions, traffic/traffic control, weather or offsite emergency worker response) should not be attempted.

   c. Sheltering may be an appropriate action for controlled releases of radioactive material from the containment, if there is assurance that the release is short term (puff release) and the area near the plant cannot be evacuated before the plume arrives.

   d. KI should be a recommendation if dose assessment or projection results indicate offsite radioactive iodine dose ≥ 5 Rem CDE to the adult thyroid.

3. At a minimum, a plant condition driven PAR to evacuate a 2 mile radius and 5 miles downwind, and shelter all other Subzones, is issued at the declaration of a General Emergency. Depending on plant conditions, evacuation of a 5 mile radius and 10 miles downwind, and shelter all other Subzones, may be issued instead of the minimum PAR.

   a. PARs are included with the initial and follow-up notifications issued at a General Emergency.

   b. The PAR must be provided to the State within 15 minutes of (1) the classification of the General Emergency or (2) any change in recommended actions.

   c. The PAR must be provided to the NRC as soon as possible and within 60 minutes of (1) the classification of the General Emergency or (2) any change in recommended actions.

3.3 Protective Action Recommendations (PARs) General Guidance (continued)

4. The Emergency Response Manager, or the SEC-MCR if the EOF is not yet activated, may elect to specify PARs for any combinations of Subzones or the entire EPZ (or beyond) regardless of plant and dose based guidance.

5. Evacuation and shelter PARs should not be extended based on the results of dose projections unless the postulated release is likely to occur within a short period of time. Plant-based PARs are inherently conservative such that expanding the evacuation zone as an added precaution may result in a greater risk from the evacuation than from the radiological consequences of a release. It also would dilute the effectiveness of the offsite resources used to accommodate the evacuation.

6. Protective actions taken in areas affected by plume deposition following the release are determined and controlled by offsite governmental agencies.

   a. HNP is not expected to develop offsite recommendations involving ingestion or relocation issues following plume passage.

   b. HNP may be requested to provide resources to support the determination of post plume protective actions.

7. Throughout the duration of a General Emergency, assess plant conditions and effluent release status to ensure the established PARs are adequate.

8. The Site Emergency Coordinator (SEC) is the decision maker on determining if a radiological emergency release is in progress. An emergency release is defined as any unplanned quantifiable discharge of radioactive material to the environment that causes, or is due to, a declared emergency event. A radiological emergency release is in progress if:

   a. Any radiation monitor listed in Table R-1 of the EAL Matrix shows an increase in activity.

   b. Primary-to-secondary leakage causes an emergency declaration.

   c. A known unmonitored release path exists from an area that contains radioactive material.

   d. Environmental Monitoring Team surveys detect an increase in background radiation levels outside the site boundary

   e. Any alternate methods is used to determine a release is in progress.

EXAMPLE
The Plant Vent Stack radiation monitor (RM-21AV-3509-1SA) is out of service and compensatory measures indicate a release is in progress.

3.3 Protective Action Recommendations (PARs) General Guidance (continued)

9. Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline states: (1) Protective Action Recommendations (PARs) are made consistent with the goal of 15 minutes once data is available, and (2) Dose assessment and PAR development are expected to be made promptly following indications that the conditions have reached a threshold in accordance with the licensee's PAR scheme. The 15 minute goal from data availability is a reasonable period of time to develop or expand a PAR. Plant conditions, meteorological data, field monitoring data, and/or radiation monitor data should provide sufficient information to determine the need to change PARs. If radiation monitor readings provide sufficient data for assessments, it is not appropriate to wait for field monitoring to become available to confirm the need to expand the PAR. The 15 minute goal should not be interpreted as providing a grace period in which the licensee may attempt to restore conditions and avoid making the PAR recommendation.

    a. Time is of the essence when conducting and approving dose projections. Dose projection results may escalate or preclude emergency declarations.

    b. The clock starts when you have indications that a PAR **threshold is exceeded**. This could be radiation level readings via installed instrumentation (e.g., ERFIS, OSI/PI, local monitors, etc.), radiation level readings from field teams, or when you complete a dose assessment.

    c. When the dose assessment is complete – the clock starts. This is the **Run date/time** on the Dose Assessment Summary Report. It is up to the Dose Assessment Team Leader, Dose Assessment Team, and the RCM to validate assumptions, results, and recommend PARs for approval by the ERM within this 15-minute period.

4.0 PROCEDURE STEPS

4.1. Emergency Classification

NOTE:	The expectation is that emergency classifications are to be made as soon as conditions are present for the classification, but within 15 minutes in all cases of conditions being present.
NOTE:	The All Conditions EAL matrix must be evaluated for all plant conditions (hot or cold).
NOTE:	Use a marker on the EAL matrix to aid in place-keeping and EAL applicability.

CAUTION

The highest emergency classification for which an Emergency Action Level (EAL) was exceeded shall be declared.

1. **EVALUATE** the "All Conditions" EAL Matrix.
    a. **READ** the EAL Matrix from left to right and top to bottom
    b. **READ** the EAL Category
    c. **READ** the EAL subcategory
    d. **READ** the Initiating Condition
    e. **READ** the Mode Applicability bar
    f. **READ** the category number criterion
    g. **READ** any applicable notes or tables
    h. **DETERMINE** EAL classification threshold applicability

2. **IF** the Reactor Coolant System temperature is greater than 200^{0}F, **THEN EVALUATE** the "Hot Conditions" EAL Matrix.
    a. **READ** the EAL Matrix from left to right and top to bottom
    b. **READ** the EAL Category
    c. **READ** the EAL subcategory
    d. **READ** the Initiating Condition
    e. **READ** the Mode Applicability bar
    f. **READ** the category number criterion
    g. **READ** any applicable notes or tables
    h. **DETERMINE** EAL classification threshold applicability

4.1 Emergency Classification (continued)

3. **IF** the Reactor Coolant System temperature is less than or equal to 200^{0}F, **THEN EVALUATE** the Cold Conditions EAL Matrix.

   a. **READ** the EAL Matrix from left to right and top to bottom

   b. **READ** the EAL Category

   c. **READ** the EAL subcategory

   d. **READ** the Initiating Condition

   e. **READ** the Mode Applicability bar

   f. **READ** the category number criterion

   g. **READ** any applicable notes or tables

   h. **DETERMINE** EAL classification threshold applicability

4. **IDENTIFY** the highest applicable emergency classification level.

5. **ANNOUNCE** to the MCR or TSC personnel the emergency event **AND** the time of classification.

6. **IMPLEMENT** requirements in PEP-230 and/or PEP-240, as appropriate.

4.2. Plant-based Protective Action Recommendations (PARs)

1. Use Attachment 3, "Protective Action Recommendation Process" as an aid in determining the proper PAR.

2. At a minimum, evacuation of a 2 mile radius and 5 miles downwind (with sheltering of all other Subzones) will be recommended for a General Emergency declaration.

3. Evacuation of a 5 mile radius and 10 miles downwind (with sheltering of all other Subzones) will be recommended for plant conditions in which damage is imminent or has occurred for all three fission product barriers as indicated by all three conditions below (a., b. and c.):

   a. Substantial core damage is imminent or has occurred as indicated by any of the following conditions:

      (1) Core damage estimations >1% melt.

      (2) Core Exit Thermocouple readings ≥ 2300° F.

      (3) Core uncovered > 30 minutes.

4.2 Plant-based Protective Action Recommendations (PARs) (continued)

b. A significant loss of reactor coolant is imminent or has occurred are indicated by any of the following conditions:

(1) Containment Radiation Monitors reading:

- >10,000 R/Hr with no containment spray.
- >4,000 R/Hr with containment spray on.

(2) Containment hydrogen gas concentration >1%.

(3) Rapid vessel depressurization.

(4) A large break loss of coolant accident.

c. Containment Barrier failure (primary or S/G) is imminent or has occurred as indicated by:

(1) A release of radioactivity greater than the projected dose of either:

- 1000 mRem TEDE at or beyond the site boundary.
- 5000 mRem Thyroid CDE at or beyond the site boundary.

OR a measured dose rate of either:

- >1000 mRem/hr at or beyond the site boundary.
- I-131 equivalent concentration > 3.9 E-6 μCi/cc at or beyond the site boundary.

(2) Primary containment pressure can not be maintained below design basis pressure of 45 psig.

(3) Primary containment H_2 gas concentration can not be maintained below combustible limit of 4% by volume.

(4) Faulted/Ruptured S/G with a relief valve open.

4. Containment monitors may provide indication of both core damage and loss of RCS. Monitor values used to determine a specific amount of core damage are dependent on plant conditions, power history, and time after shutdown. Monitor readings used to quantify an amount of damage or coolant leakage should be complimented by other indications and engineering judgment.

4.2 Plant-based Protective Action Recommendations (PARs) (continued)

5. Acceptable changes in initial PARs includes expanding evacuation but does not allow a change from evacuation of zones to sheltering of those zones.

NOTE:	A direct release is defined as a pathway from the containment to any environment outside the containment when containment or system isolation is required due to a safety injection signal, containment pressure greater than 3 psig, or a valid containment ventilation isolation signal and the pathway cannot be isolated from the Main Control Room.

6. If a release is in progress:

   a. Perform dose assessment as soon as possible to determine if PAGs are exceeded and if additional Subzones require evacuation. Add any Subzones requiring evacuation as determined by dose assessment to the plant-based PARs.

7. If no release is in progress:

   a. Perform dose projections on possible conditions as time permits to determine if PAGs could be exceeded. Consider adding any Subzones requiring evacuation as determined by dose projection to the plant-based PARs.

8. If either the dose assessment or dose projection indicate that the KI PAG (5 REM CDE to the adult thyroid) is or could be exceeded, then the KI consideration PAR should be added (line 5D on ENF).

4.3. Dose Assessment Based Protective Action Recommendations (PARs)

NOTE:	Dose projections are not required to support the decision process in Attachment 3, "Protective Action Recommendation Process."
NOTE:	Many assumptions exist in dose assessment calculations, involving both source term and meteorological factors, which make computer predictions over long distances highly questionable.

1. IF dose assessment results exceed PAGs at the outer boundary of the 10 mile EPZ, THEN:

   a. Issue an initial ENF to state and Counties that include a statement similar to the following:

      "Dose assessment results indicate PAGs are exceeded 'X' miles from the Harris Nuclear Plant. Environmental Monitoring Teams have been dispatched to verify dose assessment results."

   b. Dispatch Environmental Teams to downwind areas to verify the calculated exposure rates.

4.3 Dose Assessment Based Protective Action Recommendations (PARs) (continued)

c. IF the dose assessment data is verified, THEN issue an initial ENF to State and Counties that includes a statement similar to the following:

"Environmental Monitoring Teams have verified PAGs are exceeded 'X' miles from the Harris Nuclear Plant. Recommend expanding evacuation zones 'X' miles downwind from the plant."

d. IF dose assessment data is NOT verified, THEN issue a follow up ENF to State and Counties that includes a statement similar to the following:

"Environmental Monitoring Teams were unable to verify PAGs are exceeded beyond the 10 mile Emergency Planning Zone. No additional protective actions are recommended at this time."

NOTE:	Refer to Attachment 6, "**DOSE-ASSESSMENT-BASED PROTECTIVE ACTION RECOMMENDATIONS BACKGROUND**" for background information on Dose Assessment Based PARs.

2. From the Main Control Room: If a release is in progress and time permits, perform offsite dose assessment in accordance with PEP-340 to determine whether the plant-based protective actions of Attachment 3 are adequate.

3. From the Emergency Operations Facility: Conduct offsite dose assessment in accordance with EMG-NGGC-0002 to determine whether the plant-based protective actions of Attachment 3 are adequate using the following methods as applicable:

   a. Monitored Release:

      (1) If a release is in progress, assess the calculated impact to determine whether the plant-based PARs of Attachment 3 are adequate.

      (2) If a release is not in progress, use current meteorological and core damage data to project effluent monitor threshold values which would require 2, 5, and 10 mile evacuations (Attachment 3). Reestablish threshold values whenever meteorological conditions or core damage assessment values change.

4.3 Dose Assessment Based Protective Action Recommendations (PARs) (continued)

b. Containment Leakage/Failure:

(1) If a release is in progress, assess the calculated impact to determine whether the plant-based PARs of Attachment 3 are adequate.

(2) If a release is not in progress, use current meteorological and core damage data on various scenarios (design leakage, failure to isolate, catastrophic failure) to project the dose consequences.

(3) Determine whether the plant-based PARs of Attachment 3 are adequate.

(4) Reestablish scenario values whenever meteorological conditions or core damage assessment values change.

c. Field Survey Analysis: Actual field readings from Environmental Teams should be compared to dose assessment results and used as a dose projection method to validate calculated PARs and to determine whether the plant or release based protective actions of Attachment 3 are adequate.

d. Release Point Analysis: Actual sample data from monitored or unmonitored release points should be utilized in conjunction with other dose assessment and projection methods to validate calculated PARs and to determine whether the plant-based protective actions of Attachment 3 are adequate.

4. The Emergency Response Manager and the Radiological Control Manager shall discuss dose assessment and projection analysis results and evaluate their applicability prior to issuing PARs to the State if possible.

4.4. Downgrading the Emergency Classification Level

NOTE: The preferred method during plant recovery concerning EALs is to terminate the declared event when the plant has recovered from the effects of the initiating events rather than reducing the EAL level as recovery is completed. It is not required that emergency declarations be reduced and lower EALs declared as plant conditions improve.

1. If the action level currently has abated to a lower declaration or the situation has been resolved prior to completion of off-site reporting:

a. Declare the highest classification for which an Emergency Action Level was exceeded, if not already done, and

b. Evaluate downgrading to the emergency classification appropriate for the present conditions.

4.4 Downgrading the Emergency Classification Level (continued)

2. Downgrading of an emergency is performed by issuing a notification to a lower emergency classification level whenever plant conditions improve to satisfy the affected Emergency Action Levels. However, the following guidelines apply:

   a. If the Emergency Response Manager (ERM) position is activated, they shall be consulted before downgrading occurs.

   b. If the NRC Director of Site Operations position is activated, they should be consulted before downgrading occurs.

   c. If offsite Protective Action Recommendations have been made, the SEC-TSC shall consult with the ERM and with State and County authorities, prior to downgrading. It is recommended that any off-site Protective Action Recommendations be completed prior to downgrading of a General Emergency.

   d. Where lasting damage has occurred to the fission product barriers or to safety systems, the ERM should transition to PEP-500 rather than a simple downgrade of the emergency.

   e. For Alert or higher classifications, unless the conditions causing emergency action levels are very quickly resolved (less than approximately 30 minutes), downgrading should not occur until after the TSC and EOF are activated.

4.5. Emergency Termination and Transition to Recovery

1. If entering Recovery from an Unusual Event, determine the need for a Recovery Plan and support organization.

   a. Generally, the activities following an Unusual Event will not require the formation of a Recovery Organization or a transition period prior to event termination and entry into Recovery.

   b. Refer to PEP-500 for further guidance if recovery efforts following an Unusual Event extend beyond offsite notification and the generation of required reports.

2. Complete the Termination Checklist (Attachment 5).

   a. If conditions will allow for the termination of the emergency and entry into Recovery, exit this procedure and enter PEP-500, "Recovery."

   b. If conditions do no support termination of the emergency and entry into Recovery, continue following the guidance provided in Section 3.1.

5.0 REFERENCES

5.1. PLP-201, "Emergency Plan"

1. Section 4.1, "Emergency Classification"
2. Section 4.5.1, "Protective Action Guides"

5.2. Referenced Plant Emergency Procedures

1. PEP-230, Control Room Operations
2. PEP-240, Activation and Operation of the Technical Support Center
3. PEP-270, Activation and Operation of the emergency Operations Facility
4. PEP-310, Notifications and Communications
5. PEP-344, HNP Offsite Dose Assessment Based on Monitored Releases
6. PEP-500, Recovery

5.3. Other References

1. EMG-NGGC-0002, "Off-site Dose Assessment"
2. State of North Carolina Radiological Emergency Response Plan for Nuclear Power Facilities
3. EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents"
4. NUREG-0654 Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents"
5. NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73"
6. NUREG/BR-0150, Vol. 4, Rev.4, US NRC, RTM-96 Response Technical Manual
7. Regulatory Guide 1.101 "Emergency Planning and Preparedness for Nuclear Power Plants"
8. EPPOS No.1 "Emergency Preparedness Position (EPPOS) on Acceptable Deviations to Appendix 1 to NUREG-0654/FEMA-REP-1"
9. Harris Nuclear Plant Development of Evacuation Time Estimates, KLD Associates Final Report August 23, 2007

5.3 Other References (continued)

10. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events"

11. EP-EAL, "Emergency Action Levels"

12. NEI 1999-02, "Regulatory Assessment Performance Indicator Guideline"

13. NRC Regulatory Issue Summary 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

6.0 SPECIAL TOOLS AND EQUIPMENT

1. EAL Matrix: Matrix are maintained in the Main Control Room, TSC and EOF

2. PAR Boards: PAR boards, based on Attachment 3, are maintained in the Main Control Room, TSC and EOF

3. EP-EAL: Copies of the Emergency Action Levels are maintained in the Main Control Room, TSC and EOF

7.0 DIAGRAMS AND ATTACHMENTS

See Table of Contents.

Attachment 1 – Intentionally blank

Sheet 1 of 1

Attachment 2 – Intentionally blank

Sheet 1 of 1

Attachment 3 – Protective Action Recommendation Process

Sheet 1 of 3

General Emergency Declared?

- No → No PARs Required
- Yes ↓

Does an approved dose projection or assessment indicate ≥ 5 REM Adult Thyroid CDE?

- Yes → Recommend consideration of the use of KI
- No ↓

1, 4a

Substantial core damage is imminent or has occurred?

- No → Evacuate 2 Mile Radius and 5 Miles Downwind
- Yes ↓

2, 4b

A significant loss of reactor coolant is imminent or has occurred?

- No → Evacuate 2 Mile Radius and 5 Miles Downwind
- Yes ↓

3

Containment barrier failure (Primary or S/G) is imminent or has occurred?

- No → Evacuate 2 Mile Radius and 5 Miles Downwind
- Yes ↓

Evacuate 5 Mile Radius and 10 Miles Downwind.
Shelter all other Subzones.
Refer to PEP-110, Section 3.3 if a short "puff" release is anticipated or external conditions impose a greater risk from the evacuation than from the dose received.

Evacuate 2 Mile Radius and 5 Miles Downwind.
Shelter all other Subzones.
Refer to PEP-110, Section 3.3 if a short "puff" release is anticipated or external conditions impose a greater risk from the evacuation than from the dose received.

5 Mile Radius and 10 Miles Downwind

Wind Direction (From °)	Evacuate Subzones	Shelter Subzones
348° - 010°	A,B,C,D,H,I,K,L	E,F,G,J,M,N
011° - 034°	A,B,C,D,H,I,K,L	E,F,G,J,M,N
035° - 079°	A,B,C,D,I,J,K,L,M	E,F,G,H,N
080° - 101°	A,B,C,D,J,K,L,M	E,F,G,H,I,N
102° - 124°	A,B,C,D,K,L,M	E,F,G,H,I,J,N
125° - 146°	A,B,C,D,K,L,M,N	E,F,G,H,I,J
147° - 191°	A,B,C,D,E,K,L,M,N	F,G,H,I,J
192° - 214°	A,B,C,D,E,K,L	F,G,H,I,J,M,N
215° - 236°	A,B,C,D,E,K,L	F,G,H,I,J,M,N
237° - 259°	A,B,C,D,E,F,K,L	G,H,I,J,M,N
260° - 326°	A,B,C,D,F,G,H,K,L	E,I,J,M,N
327° - 347°	A,B,C,D,H,K,L	E,F,G,I,J,M,N

2 Mile Radius and 5 Miles Downwind.

Wind Direction (From °)	Evacuate Subzones	Shelter Subzones
327° - 010°	A,D,K	B,C,E,F,G,H,I,J,L,M,N
011° - 056°	A,K	B,C,D,E,F,G,H,I,J,L,M,N
057° - 124°	A,K,L	B,C,D,E,F,G,H,I,J,M,N
125° - 191°	A,B,L	C,D,E,F,G,H,I,J,K,M,N
192° - 214°	A,B	C,D,E,F,G,H,I,J,K,L,M,N
215° - 259°	A,B,C	D,E,F,G,H,I,J,K,L,M,N
260° - 281°	A,C	B,D,E,F,G,H,I,J,K,L,M,N
282° - 304°	A,C,D	B,E,F,G,H,I,J,K,L,M,N
305° - 326°	A,D	B,C,E,F,G,H,I,J,K,L,M,N

Attachment 3 – Protective Action Recommendation Process

Sheet 2 of 3

Acceptable changes in initial PARS would include expanding evacuation but would not allow a change from evacuation of zones to sheltering of those zones.

1. Indications that substantial core damage is imminent or has occurred include:
   a) Core damage > 1% melt.
   b) Core Exit Thermocouple readings ≥ 2300° F.
   c) Core uncovered > 30 minutes.
2. Indications that a significant loss of reactor coolant is imminent or has occurred include:
   a) Containment radiation reading > 10,000 R/Hr without spray or > 4,000 R/Hr with spray.
   b) Containment hydrogen gas concentration > 1%.
   c) Rapid vessel depressurization.
   d) A large break loss of coolant accident.
3. Indications that containment barrier failure (primary or S/G) is imminent or has occurred are indicated by:
   a) A release of radioactivity greater than the projected dose of either:
      - 1000 mRem TEDE at or beyond the site boundary.
      - 5000 mRem Thyroid CDE at or beyond the site boundary.

      Or a measured dose rate of either:
      - >1000 mRem/hr at or beyond the site boundary.
      - I-131 equivalent concentration > 3.9 E-6 µCi/cc at or beyond the site boundary.
   b) Primary containment pressure can not be maintained below design basis pressure of 45 psig.
   c) Primary containment H_2 gas concentration can not be maintained below combustible limit of 4% by volume.
   d) Faulted/Ruptured S/G with a relief valve open.

NOTE:	A direct release is defined as a pathway from the containment to any environment outside the containment when containment or system isolation is required due to a safety injection signal, containment pressure greater than 3 psig, or a valid containment ventilation isolation signal and the pathway cannot be isolated from the Main Control Room.

4. Accidents which result in a direct release pathway to the environment will most likely be thyroid dose limiting. For a faulted and ruptured S/G, water level must be below the tube bundles (S/G Narrow Range < 25% normal containment conditions or < 40% adverse containment conditions) with a relief valve open before it is considered a direct release pathway to the environment. For circumstances involving a direct release pathway to the environment:
   a) Consider **any** loss of Fuel sufficient to warrant the determination that substantial core damage has occurred.
   b) Consider **any** loss of RCS sufficient to warrant the determination that a significant loss of reactor coolant has occurred.
5. PARs due to Spent Fuel Pool releases are determined using Attachment 6, Dose Assessment Based Protective Action Recommendations.
6. Containment monitors can provide indication of a loss or potential loss of both core damage and loss of RCS. Monitor readings used to quantify an amount of damage or coolant leakage should be complimented by other indications and engineering judgment.

Attachment 3 – Protective Action Recommendation Process

Sheet 3 of 3

If a release is in progress:

- Perform dose assessment as soon as possible to determine if PAGs are exceeded and if additional Subzones require evacuation.
- Add any Subzones requiring evacuation as determined by dose assessment to the plant-based PARs.

If no release is in progress:

- Perform dose projection on possible conditions as time permits to determine if PAGs could be exceeded. Consider adding any Subzones requiring evacuation as determined by dose projection to the plant-based PARs.

Attachment 4 - Event Information Worksheet
Sheet 1 of 2

Date/Time:____________ (Use ERFIS time)

EVENT INFORMATION WORKSHEET

A) Emergency Classification

Time Declared: ____________ (24 hr)

❑ Unusual Event ❑ Alert

❑ Site Area ❑ General

Provide a brief summary of the event and mitigating actions in progress:

EAL: ____________________

B) Fission Product Barrier Status

	Fuel	RCS	Cnmt
Intact:	❑	❑	❑
Potential Loss:	❑	❑	❑
Loss:	❑	❑	❑

C) Plant Conditions

❑ On-Line ❑ At Power: ________ %

❑ Off-Line ❑ Cooling Down

❑ Cold Shutdown

Time of Rx Shutdown: ____________ (24 hr)

❑ Stable ❑ Improving

❑ Degrading

Describe plant and recent activities ________

Describe equipment, instrument, or other problems: ________________

ERDS Status: ❑ On-Line ❑ Off-Line

ERFIS Status: ❑ On-Line ❑ Off-Line

OSI/PI Status: ❑ On-Line ❑ Off-Line

D) Radiological Release

❑ None ❑ Controlled

❑ Is Occurring ❑ Uncontrolled

❑ Has Occurred ❑ Below PAGs

❑ Above PAGs

Time Started: ____________ (24 hr)

Noble Gas: ____________ Ci/sec

Iodines: ____________ Ci/sec

Projected Duration: ____________ hours

Environmental Monitoring Team activities:

E) Personnel Status

Missions in plant: ❑ No ❑ Yes

Location of in-plant teams/personnel: ________

Injuries (No._____): ❑ No ❑ Yes

Contamination(s): ❑ No ❑ Yes

Over Exposure(s): ❑ No ❑ Yes

❑ Minor ❑ Major

Details (names of injured, status of family notification):

F) Facility Activation Status

❑ TSC: ____________ (24 hr)

❑ OSC: ____________ (24 hr)

❑ EOF: ____________ (24 hr)

❑ JIC: ____________ (24 hr)

If TSC is not ready for activation can the TSC accept responsibility for:

Notification to NRC: ❑ N/A ❑ No ❑ Yes

If EOF is not yet ready for activation can the EOF accept responsibility for:

Emergency Communicator Communications to State and Counties (ENF must still be approved by SEC) ❑ No ❑ Yes

Dose Assessment ❑ No ❑ Yes

Attachment 4 - Event Information Worksheet

Sheet 2 of 2

EVENT INFORMATION WORKSHEET

G) Offsite Assistance Requested

- ☐ None
- ☐ Medical ____________ (24 hr)
  - ○ Ambulance ○ Helicopter
- ☐ Fire Department ____________ (24 hr)
  - ○ Holly Springs ○ Apex
- ☐ Law Enforcement ____________ (24 hr)
  - ○ Local ○ State

H) Onsite Protective Actions

- ☐ None
- ☐ Assembly/Accountability
- ☐ Local Area(s) Evacuated
- ☐ Protected Area Evacuated
- ☐ Exclusion Area Evacuated
- ☐ Potassium Iodide Issued
- ☐ Employee Info Phone #: ________________

I) Offsite Notifications (last issued)

State/County	Time: ____________	(24 hr)
NRC	Time: ____________	(24 hr)
News Release	Time: ____________	(24 hr)
Hospital	Time: ____________	(24 hr)
INPO	Time: ____________	(24 hr)
ANI	Time: ____________	(24 hr)

J) PARs

- ☐ None Issued, or
  - ○ Evac: A B C D E F G H I J K L M N
  - ○ Shelter: A B C D E F G H I J K L M N

(Circle the affected subzones)

  - ○ Consideration of the use of KI

K) Offsite Facility Activation Status

- ☐ Chatham County EOC: ________ (24 hr)
- ☐ Harnett County EOC: ________ (24 hr)
- ☐ Lee County EOC: ________ (24 hr)
- ☐ Wake County EOC: ________ (24 hr)
- ☐ State EOC: ________ (24 hr)
- ☐ NRC Incident Response Center: ________ (24 hr)

L) Offsite Actions/Response

- ☐ None Issued, or
  - ○ Schools ○ Daycare
  - ○ Hospitals ○ Rest Homes
  - ○ Lake Evacuations
  - ○ Other: ______________________________

____

  - ○ Evac: A B C D E F G H I J K L M N
  - ○ Shelter: A B C D E F G H I J K L M N

(Circle the affected subzones)

  - ○ KI administered to the General Public
- ☐ Sirens Activated: ________ (24 hr)
- ☐ Tone Alerts Activated: ________ (24 hr)
- ☐ EAS Activated: ________ (24 hr)

Any applicable incomplete items from previous pages of PEP-230, Attachment 1 - SITE EMERGENCY COORDINATOR – CR checklist? ______________________________

Any assistance needed? ______________________________

Comments ______________________________

Attachment 5 - Termination Checklist

Sheet 1 of 2

TERMINATION CHECKLIST

		True	False
1.	Conditions no longer meet an Emergency Action Level and it appears unlikely that conditions will deteriorate. List any EAL(s) which is/are still exceeded and a justification as to why a state of emergency is no longer applicable: ______________________________ ______________________________ ______________________________	☐	☐
2.	Plant releases of radioactive materials to the environment are under control (within Tech Specs) or have ceased and the potential for an uncontrolled radioactive release is acceptably low.	☐	☐
3.	The radioactive plume has dissipated and plume tracking is no longer required. The only environmental assessment activities in progress are those necessary to determine the extent of deposition resulting from passage of the plume.	☐	☐
4.	In-plant radiation levels are stable or decreasing, and acceptable given the plant conditions.	☐	☐
5.	The reactor is in a stable shutdown condition and long-term core cooling is available.	☐	☐
6.	The integrity of the Reactor Containment Building is within Technical Specification limits.	☐	☐
7.	The operability and integrity of radioactive waste systems, decontamination facilities, power supplies, electrical equipment and plant instrumentation including radiation monitoring equipment is acceptable.	☐	☐
8.	Any fire, flood, earthquake or similar emergency condition or threat to security no longer exists.	☐	☐

Attachment 5 - Termination Checklist
Sheet 2 of 2

TERMINATION CHECKLIST

		True	False
9.	Any contaminated injured person has been treated and/or transported to a medical care facility.	❑	❑
10.	All required notifications have been made.	❑	❑
11.	The NRC Senior Resident Inspector has been notified that the event will be terminated.	❑	❑
12.	Offsite conditions do not unreasonably limit access of outside support to the station and qualified personnel and support services are available.	❑	❑
13.	Discussions have been held with Federal, State and County agencies and agreement has been reached and coordination established to terminate the emergency.	❑	❑

It is not necessary that all responses listed above be 'TRUE'; however, all items must be considered prior to event termination and entry into Recovery.. For example, it is possible that some conditions remain which exceed an Emergency Action Level following a severe accident but entry into Recovery is appropriate. Additionally, other significant items not included on this list may warrant consideration such as severe weather.

Comments: ______________________________

Approved: ______________________ Site Emergency Coordinator ____________ Date ____________ Time

Attachment 6 – Dose-Assessment-Based Protective Action Recommendations Background

Sheet 1 of 2

1. Protective Action Guides

   - The evacuation of the general public will usually be justified when the projected TEDE dose to an individual is one Rem or greater or the projected CDE thyroid dose is five Rem or greater.

2. EPZ Subzones

   - The objective of the dose assessment calculations is to allow for the determination of protective actions. Protective actions may affect any portion of or the entire Emergency Planning Zone (EPZ).

   a. The EPZ extends out to ten miles from the plant. The EPZ is then divided radially into three rings (0-2, 2-5, and 5-10 miles) and axially into sixteen 22.5° sectors. This allows for the implementation of protective actions within specifically affected areas, or 'Key Holes' rather than across the entire EPZ.

   b. Subzones are used to define the areas within the Harris EPZ. Radial distances are maintained approximately equal to the standard EPZ, but axial sector based areas have been abandoned. The fourteen Subzones which make up the HNP EPZ are divided using geopolitical, natural, and man-made boundaries.

   c. The Subzones are divided into three groups.

      (1) Subzone A encompasses the inner ring which extends out to approximately two miles from the plant.

      (2) Subzones B, C, D, K, and L compose the middle ring, at about two to five miles from the plant.

      (3) Subzones E, F, G, H, I, J, M, and N make up the outer ring, five to ten miles from the plant.

3. Subzone Evacuation Groups

   a. The combination of Subzones which compose a group was determined as follows:

Attachment 6 – Dose-Assessment-Based Protective Action Recommendations Background
Sheet 2 of 2

b. For any given wind direction a combination of Subzones will be affected by the plume. By defining the maximum horizontal dispersion at ten miles in the crosswind axis for a plume of stability class 'A' (most unstable) and solving for the angle a plume with a 32 ° footprint is created.

$$\sigma_y = (-0.0234 * \ln(x) + 0.350)x = 2.96 \text{ miles}$$

$$\tan(x) = \sigma_y / x = 16^\circ$$

c. The wind direction band that will affect each Subzone is ascertained by transcription of worse case plume onto a United States Geological Survey map. By accounting for the overlapping of wind directions, fifteen distinct Subzone combinations (groups) are established.

d. Taking into account that not all Subzones out to the EPZ boundary need be evacuated in all cases, additional subarea groups are generated. From this, twenty five possible evacuation combinations exist for all possible wind directions. A further adjustment was included to align the sub-area groups to agree with the ETE data. The combinations of subarea groups are as follows:

Evacuation Subzone/Groups

W.D. (° From)	0-2 Miles		0-5 Miles		0-10 Miles	
	Sub-zone	Group #	Sub-zones	Group #	Sub-zones	Group #
011°-034°	A	1	A, K	2	A, K, H, I, J	11
035°-056°	A	1	A, K	2	A, K, I, J, M	12
057°-079°	A	1	A, K, L	3	A, K, I, J, L, M	13
080°-101°	A	1	A, K, L	3	A, K, J, L, M	14
102°-124°	A	1	A, K, L	3	A, K, J, L, M, N	15
125°-146°	A	1	A, B, L	4	A, B, L, M, N	16
147°-191°	A	1	A, B, L	4	A, B, E, L, M, N	17
192°-214°	A	1	A, B	5	A, B, E, N	18
215°-236°	A	1	A, B, C	6	A, B, C, E, F	19
237°-259°	A	1	A, B, C	6	A, B, C, E, F, G	20
260°-281°	A	1	A, B, C, D	7	A, B, C, D, F, G, H	21
282°-304°	A	1	A, C, D	8	A, C, D, F, G, H	22
305°-326°	A	1	A, C, D, K	9	A, C, D, F, G, H, K	23
327°-347°	A	1	A, D, K	10	A, D, G, H, I, K	24
348°-010°	A	1	A, D, K	10	A, D, H, I, K	25

Revision Summary

<table>
<tr><th colspan="2">Revision 22 Summary</th></tr>
<tr><td colspan="2">Rev. 22 processed with PRR: 409023
PRRs Incorporated: 409023
CRs Incorporated: 501711-05 (CORR), 569001-18 (ENHN), CR 589380-05 (CORR)
ECs Incorporated: NA</td></tr>
<tr><td>3.3 Step 8</td><td>[CR 501711-05, CR 569001-18]
From: The Site Emergency Coordinator (SEC) is the decision maker on determining if an emergency release (radioactive) is in progress. An emergency release is defined as any unplanned quantifiable discharge to the environment of radioactive effluent attributable to a declared emergency event. To assist in this determination, the following are gaseous and liquid release in-progress definitions:
a. A gaseous (airborne) emergency release (radioactive) is in progress if any of the following conditions exist
(1) An approved monitored release was occurring AND the reading on the radiation monitor designated to monitor this release increases due to the emergency event.
(2) Any release due to the emergency event that was not previously approved.
(3) Any primary-to-secondary leak which causes an emergency declaration.
b. A liquid emergency release (radioactive) is in progress if any of the following conditions exist:
(1) An approved monitored release was occurring AND the reading on the radiation monitor designated to monitor this release increases but does not isolate on an alarm signal
(2) The rupture of a system, that releases radioactive liquids into an area which affects or has the potential to affect an offsite environment.
c. A direct release is defined as a pathway from the containment to any environment outside the containment when containment or system isolation is required due to a safety injection signal, containment pressure greater than 3 psig, or a valid containment ventilation isolation signal and the pathway cannot be isolated from the Main Control Room.
To: The Site Emergency Coordinator (SEC) is the decision maker on determining if a radiological emergency release is in progress. An emergency release is defined as any unplanned quantifiable discharge of radioactive material to the environment that causes, or is due to, a declared emergency event. A radiological emergency release is in progress if:
a. Any radiation monitor listed in Table R-1 of the EAL Matrix shows an increase in activity.
b. Primary-to-secondary leakage causes an emergency declaration.
c. A known unmonitored release path exists from an area that contains radioactive material from commercial nuclear power plant operations.
d. Environmental Monitoring Team surveys detect an increase in background radiation levels outside the site boundary
e. Any alternate methods is used to determine a release is in progress.
EXAMPLE
The Plant Vent Stack radiation monitor (RM-21AV-3509-1SA) is out of service and compensatory measures indicate a release is in progress.</td></tr>
<tr><td>4.2 step 3.c, Attachment 3 sh 1 flowchart, and Attachment 3 sh 2 step 3</td><td>From: "Containment failure (primary or S/G) ..."
To: "Containment barrier failure (primary or S/G) ..."</td></tr>
</table>

2013 NRC SRO Question 92 (17) Reference

Revision 22 Summary	
4.2 step 4, and Attachment 3 sh 2 step 6	[PRR 409023, CR 501711-05] Changed first sentence From: "Containment monitors may provide indication of both core damage and RCS breach." To: "Containment monitors may provide indication of both core damage and loss of RCS."
4.3 step 1	[CR 589380-05] From: "In the event dose assessment results indicate the need to recommend actions beyond the outer EPZ boundaries that is past 10 miles: Dispatch Environmental Teams to downwind areas to verify the calculated exposure rates prior to issuing PARs outside the EPZ. Many assumptions exist in dose assessment calculations, involving both source term and meteorological factors, which make computer predictions over long distances highly questionable. To: **NOTE:** Many assumptions exist in dose assessment calculations, involving both source term and meteorological factors, which make computer predictions over long distances highly questionable. "IF dose assessment results exceed PAGs at the outer boundary of the 10 mile EPZ, THEN: a. Issue an initial ENF to state and Counties that include a statement similar to the following: "Dose assessment results indicate PAGs are exceeded 'X' miles from the Harris Nuclear Plant. Environmental Monitoring Teams have been dispatched to verify dose assessment results." b. Dispatch Environmental Teams to downwind areas to verify the calculated exposure rates. c. IF the dose assessment data is verified, THEN issue an initial ENF to State and Counties that includes a statement similar to the following: "Environmental Monitoring Teams have verified PAGs are exceeded 'X' miles from the Harris Nuclear Plant. Recommend expanding evacuation zones 'X' miles downwind from the plant." d. IF dose assessment data is NOT verified, THEN issue a follow up ENF to State and Counties that includes a statement similar to the following: "Environmental Monitoring Teams were unable to verify PAGs are exceeded beyond the 10 mile Emergency Planning Zone. No additional protective actions are recommended at this time."
Attachment 3 sh 2 step 4.a	[PRR 409023, CR 501711-05] From: "Consider **any** Fuel Breach sufficient..." To: "Consider **any** loss of Fuel sufficient..."
Attachment 3 sh 2 step 4.b	[PRR 409023, CR 501711-05] From: "Consider **any** RCS Breach sufficient..." To: "Consider **any** loss of RCS sufficient ..."
Attachment 3 sh 2 step 5	[PRR CR 501711-05] new step "PARs due to Spent Fuel Pool releases are determined using Attachment 6, Dose Assessment Based Protective Action Recommendations."
Throughout	Incorporated formatting and word processing features, such as consistent use of auto step numbering, indentations, note boxes and cross referencing. These are editorial corrections per PRO-NGGC-0204 and do not need to be addressed further.

2013 NRC SRO Question 93 (18) Reference

D.1 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.10 Radioactive Liquid Effluent Monitoring Instrumentation

OPERATIONAL REQUIREMENT

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Operational Requirement 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above Operational Requirement, immediately (1) suspend the release of radioactive liquid effluents monitored by the affected channel or (2) declare the channel inoperable and take ACTION as directed by b. below.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best effort to restore to the minimum number of radioactive liquid effluent channels within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report pursuant to ODCM, Appendix F, Section F.2 why this inoperability was not corrected in a timely manner.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

2013 NRC SRO Question 93 (18) Reference

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Lines		
1) Treated Laundry and Hot Shower Tanks Discharge Monitor	1	35
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	1	35
3) Secondary Waste Sample Tank Discharge Monitor	1	35, 36*
b. Turbine Building Floor Drains Effluent Line	1	36
2. Radioactivity Monitor Providing Alarm and Automatic Stop Signal to Discharge Pump		
a. Outdoor Tank Area Drain Transfer Pump Monitor	1	37
3. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Normal Service Water System Return From Waste Processing Building to the Circulating Water System	1	39
b. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	1	39
4. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Lines		
1) Treated Laundry and Hot Shower Tanks Discharge	1	38
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge	1	38
3) Secondary Waste Sample Tank	1	38
b. Cooling Tower Blowdown	1	38

* When the Secondary Waste System is in the continuous release mode and releases are occurring, Action 36 shall be taken when the monitor is inoperable. In the batch release mode, Action 35 is applicable.

2013 NRC SRO Question 93 (18) Reference

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Operational Requirement 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 1E-07 µCi/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 µCi/gram DOSE EQUIVALENT I-131 or,

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 µCi/gram DOSE EQUIVALENT I-131.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 1E-07 µCi/ml.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the weekly Cooling Tower Blowdown weir surveillance is performed as required by Operational Requirement 4.11.1.1.1. Otherwise, follow the ACTION specified in ACTION 37 above.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. A separate day tank containing a minimum of 1457 gallons of fuel,

   2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

   3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable:

   1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

   2. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

   3. Verify required feature(s) powered from the OPERABLE offsite A.C. source are OPERABLE. If required feature(s) powered from the OPERABLE offsite circuit are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 24 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

ACTION (Continued):

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

*2. Within 24 hours, determine the OPERABLE diesel generator is not inoperable due to a common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.4#; and

3. Restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

c. With one offsite circuit and one diesel generator of 3.8.1.1 inoperable:

NOTE: Enter applicable Condition(s) and Required Action(s) of LCO 3/4.8.3, ONSITE POWER DISTRIBUTION - OPERATING, when this condition is entered with no A.C. power to one train.

1. Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Following restoration of one A.C. source (offsite circuit or diesel generator), restore the remaining inoperable A.C. source to OPERABLE status pursuant to requirements of either ACTION a or b, based on the time of initial loss of the remaining A.C. source.

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*This ACTION is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

ACTION (Continued):

d. With two of the required offsite A.C. sources inoperable:

1. Restore one offsite circuit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

2. Verify required feature(s) are OPERABLE. If required feature(s) are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 12 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) inoperable.

3. Following restoration of one offsite A.C. source, restore the remaining offsite A.C. source in accordance with the provisions of ACTION a with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. source.

e. With two of the required diesel generators inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

#2. Restore one of the diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one diesel generator, restore the remaining diesel generator in accordance with the provisions of ACTION b with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator.

f. With three or more of the required A.C. sources inoperable:

1. Immediately enter Technical Specification 3.0.3.

2. Following restoration of one or more A.C. sources, restore the remaining inoperable A.C. sources in accordance with the provisions of ACTION a,b,c,d and/or e as applicable with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. sources.

g. With contiguous events of either an offsite or onsite A.C. source becoming inoperable and resulting in failure to meet the LCO:

1. Within 6 days, restore all A.C. sources required by 3.8.1.1 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

h. With one automatic load sequencer inoperable:

1. Restore the automatic load sequencer to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

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HARRIS NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME 1

PART 1

PROCEDURE TYPE: ADMINISTRATIVE PROCEDURE (AP)

NUMBER: **AP-617**

TITLE: **REPORTABILITY DETERMINATION AND NOTIFICATION**

Table of Contents

1.0 PURPOSE

R

1. This procedure provides guidance in determining NRC Reportability in the following areas:

   a. Events requiring verbal notification to the NRC via Emergency Telecommunication System (ETS) within one, four, eight, or twenty-four hours.

   b. Events requiring a written follow-up report to the NRC as a Licensee Event Report (LER) or as a Special Report.

   c. Scheduled Routine Reports required by Title 10 of the Code of Federal Regulations (CFR) or by the Operating License Technical Specifications; PLP-114, Relocated Technical Specifications and Design Basis Requirements; and the Offsite Dose Calculation Manual (ODCM).

   d. Event Reports (other than LERs) that are prepared on an as needed basis.

   e. Reportability evaluation for non-routine reports will be based on Condition Reports per Reference 2.3.

2. The reports listed in this procedure are regulatory requirements. The specific reference for each report is identified with each report on the applicable attachment.

3. Several specific reporting requirements are also addressed by other procedures:

   a. Immediate notifications of safeguards (security) related events related to the Physical Security of Special Nuclear Material as required by §73.71 (Reporting of Safeguards Events) will be classified per Reference 2.7.

   b. Reporting of events which result in the declaration of an emergency classification shall be in accordance with Emergency Plan and implementing procedures.

   c. Reporting of events regarding fish kills, hazardous substance releases, and oil spills shall be in accordance with Reference 2.17 for notification to appropriate Corporate, Local, State and Federal (non-NRC) agencies.

   d. Reporting of events regarding environmental violations shall be in accordance with this procedure and References 2.30 and 2.31 for notification to appropriate Corporate, Local, State and Federal (non-NRC) agencies.

   e. Reporting of events to insurers regarding certain fires or other losses shall be in accordance with Reference 2.32.

   f. Reporting of events regarding non-routine radioactive releases shall be in accordance with Reference 2.37 for notification to appropriate Corporate, Local, State and Federal (non-NRC) agencies.

1.0 PURPOSE (continued)

4. This procedure provides instructions for the immediate notification to the NRC using the Emergency Telecommunication System (ETS) phone for non-emergency events that require such reporting according to §50.72, Technical Specifications and other § requirements.

2.0 REFERENCES

1. SEC-NGGC-2120, Use Storage and Protection of Safeguards and Other Limited Access Information
2. AP-611, Regulatory Correspondence (superseded by REG-NGGC-0016)
3. CAP-NGGC-0200, Condition Identification and Screening Process
4. REG-NGGC-0013, Evaluating and Reporting of Defects and Noncompliance in Accordance With 10 CFR 21
5. PEP-310, Notifications and Communications
6. AP-620, Licensee Event Report Development and Approval (superseded by REG-NGGC-0016)
7. SEC-NGGC-2147, Reporting of Safeguards and Fitness for Duty Events
8. SHNPP Operating License and Technical Specifications
9. NUREG 1022, Licensee Event Report System
10. NRC Inspection Procedure 61706, Core Thermal Power Evaluation
11. SP-014, Additional Surveillance/Compensatory Security Measures
12. ADM-NGGC-0201, Nuclear Task Management
13. RDC-NGGC-0001, NGG Standard Records Management Program
14. Offsite Dose Calculation Manual (ODCM)
15. PLP-114, Relocated Technical Specifications and Design Basis Requirements
16. FSAR, TMI Appendix Item II.K.3.3, Report on Safety and Relief Valve Failures and Challenges
17. PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, and Oil Spill Notification
18. NUREG 1460, Guide to NRC Reporting and Recordkeeping Requirements
19. Regulatory Guide 1.16, Reporting of Operating Information - Appendix A Technical Specifications

2.0 REFERENCES (continued)

20. Regulatory Guide 10.1, Compilation of Reporting Requirements for Persons Subject to NRC Regulations
21. IE Information Notice No. 83-34, Event Notification Information Worksheet
22. IE Information Notice No. 85-62, Backup Telephone Numbers to the NRC Operations Center
23. IE Information Notice No. 85-78, Event Notification
24. Letter HELD-H-278, Zimmerman to Beatty et al., March 31, 1987
25. SAF-SUBS-00033, Employee Incident Investigations
26. PLP-201, Emergency Plan
27. EPM-400, Public Notification and Alerting System
28. ESR 95-00745, Deletion of “Fail indicator circuits for CT Beacons”
29. U.S. Department of Transportation, Federal Aviation Administration Advisory Circular AC 70/7460-1K
30. EMP-001, NPDES Permit Monitoring
31. National Pollutant Discharge Elimination System (NPDES) Permit Number NC0039586, North Carolina Department of Environment and Natural Resources, Division of Water Quality
32. PLP-105, Insurance Programs at Harris Nuclear Plant
33. NEI Position Statement: Guidance to Licensees on Complying with the Licensed Power Limit (NRC ADAMS Accession No. ML081750537)
34. NRC Memorandum titled “Discussion of ‘Licensed Power Level’”, Jordan, E.L., Division of Reactor Operations Inspection, Aug. 22, 1980
35. PLP-300, Process Control Program
36. SEC-NGGC-2140, Fitness For Duty Program
37. CHE-NGGC-0057, Groundwater Protection Program
38. PLP-717, Equipment Important To Emergency Preparedness and ERO Response
39. REG-NGGC-0016, Regulatory Correspondence & LER Development
40. CR 580945, Oct. 3, 2012 Notice of Violation for EOF
41. FPP-013, Fire Protection-Minimum Requirements, Mitigating Actions and Surveillance Requirements

3.0 DEFINITIONS/ABBREVIATIONS

1. Code of Federal Regulations - CFR
2. Equipment Inoperable Record - EIR
3. Emergency Response Facility Information System – ERFIS

3.0 DEFINITIONS/ABBREVIATIONS (continued)

4. Emergency Telecommunication System - ETS
5. Engineered Safety Feature - ESF
6. Licensee Event Report (LER) - A written report conforming to the format and content requirements of §50.73 and NUREG 1022.
7. National Oceanic and Aeronautic Administration - NOAA
8. Nuclear Regulatory Commission - NRC
9. Offsite Dose Calculation Manual - ODCM
10. Operating License - OL
11. Reactor Protection System - RPS
12. Safety Parameter Display System – SPDS
13. Solid State Protection System – SSPS
14. Emergency Notification System – ENS
15. Health Physics Network - HPN

4.0 RESPONSIBILITIES

1. The Shift Manager (SM):
   a. Determining immediate NRC reportability, and
   b. Making appropriate notifications.
2. The Supervisor - Licensing/Regulatory Programs:
   a. Confirming the correctness of immediate reportability determinations.
   b. Determining need for reports to other outside agencies.
   c. Generating related reports as required.
3. The Superintendent - Security (or On-duty Security Supervisor):
   a. Evaluating security related events in accordance with Reference 2.7
   b. Informing the SM when a security related event must be reported to the NRC.

5.0 PROCEDURE

5.1 Immediate Reportability

1. The Shift Manager (SM) determines that an event requires immediate notification (per Attachment 1), or the Superintendent - Security (or On-duty Security Supervisor) informs the SM that an event requires notification to the NRC.

2. The SM prepares or assigns an individual to prepare the Reactor Plant Event Notification Worksheet (Attachment 7). Event Notification Worksheets for Safeguard/Security events are normally prepared by the Security Organization.

3. The Event Description narrative should be as short and concise as possible while conveying a clear description of the event. Attachment 5 may be used for developing one-hour notifications. Attachment 6 is a completed sample Worksheet.

4. The initial part of the Event Description should state:

    a. The initial conditions of the plant or affected systems prior to event occurrence.

    b. The actual event and direct cause, if known.

    c. The current conditions of the plant or affected systems

    Example - Plant was in Mode 1 at 50% reactor power and increasing load. At 1000, the reactor was manually tripped following a loss of both main feedwater pumps caused by feedwater regulating valve oscillations. The plant is stable in Mode 3 at normal temperature and pressure.

5. The balance of the Event Description should contain known specific details of precursor events which led to the reportable event, including the time of each event. Report only known facts; do not speculate.

    Example - Feedwater regulating valve oscillations occurred when placing valves into automatic control. The A and B feedwater regulating valves had been successfully placed in automatic at 0950 and were controlling normally. When the C valve was placed in automatic at 0958, large oscillations were noted in the "C" valve followed by oscillations in the "A" and "B" valves. During the oscillation, the condensate booster pump tripped on low flow resulting in tripping of the feed pump. The reactor was manually tripped prior to receipt of a steam generator low water level signal.

5.1 Immediate Reportability (continued)

6. The Event Description should include a statement of the proper functioning or failure to function of safety systems and the safety significance of an event, if such a determination is possible. If possible, also include two or three compensatory actions taken to assure safety. The Shift Technical Advisor may assist the SM in making such a determination.

   Example - All safety systems functioned as expected (or list equipment which failed to function as expected). "AFW automatically actuated to provide continued decay heat removal. Compensatory actions to assure safety include..."

7. The SM:

   a. Reviews the Event Notification Worksheet.

   b. Makes changes if necessary.

   c. Approves it for release.

8. If time permits, the SM shall contact the General Manager - Harris Plant and the NRC Resident Inspector, and Licensing/Regulatory Programs and provide them the information contained in the notification.

9. The SM notifies the NRC by giving the approved Event Notification Worksheet to an available individual to telecopy the Worksheet to the NRC via the fax number (301-816-5151, may be confirmed via EPL-001).

NOTE: The NRC electronically records notifications.

10. When the approved Event Notification Worksheet has been sent, contact the NRC Operations Center Duty Officer by performing either of the following:

    a. Pick up the receiver on the Emergency Telecommunication System Telephone and dial the NRC Operations Center Duty Officer via one of the numbers located on the phone label, in EPL-001, or on the Event Notification Worksheet.

    OR

    b. If desired, use a normal telephone line to call the NRC Operations Center Duty Officer via one of the numbers located on the phone label, in EPL-001, or on the Event Notification Worksheet.

5.1 Immediate Reportability (continued)

11. When the Duty Officer responds:

    a. Caller says, "THIS IS THE HARRIS NUCLEAR PLANT. THIS IS A NOTIFICATION OF (appropriate event classification from worksheet). HAVE YOU RECEIVED A TELECOPY OF THIS NOTIFICATION?"

    b. If response is "No", have an available individual perform Step 5.1.9 again while continuing with this step.

    c. The caller gives the information on the Event Notification Worksheet and repeats information when requested.

    d. The notification should be read in its entirety.

    e. The caller should respond to any requests for additional information that can be answered accurately, or if the caller is not able to accurately respond to the Duty Officer's requests, the caller shall write down the request and inform the Duty Officer that this information will be delivered in a follow up notification.

    f. The caller should record questions asked, responses provided and if follow up is necessary on a separate sheet of paper and attach it to the Event Notification Worksheet.

12. If the Duty Officer has not received a telecopy after the notification has been completed, the caller shall request the Duty Officer to read back the notification and, if necessary, correct any errors.

13. The caller records the Event Notification Number, name of the individual contacted and time of contact on the Event Notification Worksheet.

14. The caller informs the Duty Officer that the caller is signing off. The Duty Officer may request to stay on and leave the line open. If this occurs, the caller should comply. A replacement caller may be necessary to stay on the phone.

15. If additional information is provided to the Duty Officer beyond the initial Event Notification Worksheet, notify the General Manager - Harris Plant, the NRC Resident Inspector, and Licensing/Regulatory Programs of the additional information provided.

5.1 Immediate Reportability (continued)

16. Follow up Notifications.

    In addition to making the required initial notifications, during the course of the event IMMEDIATELY report:

    a. Any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made.

    b. The results of ensuing evaluations or assessments of plant conditions.

    c. The effectiveness of response or protective measures taken.

    d. Information related to plant behavior that is not understood.

17. Notify Site Communications, or if there is no response, Corporate Communications Media Line of this Reactor Plant Event Notification Worksheet. If after hours leave a message for the on call person. (See EPL-001 for contact numbers).

18. Event Notification Worksheets which have been designated "Safeguards Information" in accordance with the provision of References 2.1 and 2.7 shall be returned to Security after the notification has been made with no further dissemination.

19. The Event Notification Worksheet should be sent to Licensing/Regulatory Programs; this does not apply to "Security" Notifications.

20. Licensing/Regulatory Programs will also evaluate if follow up notification is required for clarification, retraction or other. The below criteria should be considered:

    a. Clarity for public docket (not extremely technical)

    b. Minimize inflammatory jargon – be precise and factual

    c. Provides perspective and mitigating conditions

21. Licensing/Regulatory Programs will initiate ARs to track generation of follow up reports.

22. Licensing/Regulatory Programs will forward a copy of the Event Notification Worksheet to the ICES Coordinator within five working days for entry into the ICES database.

5.2 Other Reports

Licensing/Regulatory Programs shall perform the following:

1. Reportability determinations for Steps 2 through 6 below shall be completed expeditiously. Reports should be confirmed as tracked by an AR.

2. Evaluate the condition for reportability as a Special Report under Technical Specification Section 6.9.2 per Attachment 2.

3. Evaluate the condition for reportability as an LER using Reference 2.9. Development of the LER is per Reference 2.6. As indicated in §50.73(a)(1), invalid actuations, other than Reactor Protection System actuations when the reactor is critical, may be reported by telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of a written LER.

CAUTION
Evaluation for §21 Reportability may not be substituted for reporting pursuant to §50.73. Actual reporting per §21 may be performed using an LER per §50.73 and Ref. 2.6.

4. Evaluate the condition for potential reportability under §21 per Reference 2.4.

5. Evaluate the condition for reportability via a Routine Report per Attachment 3.

6. Evaluate the condition for reportability via an Event Report (other than LER) per Attachment 4.

6.0 DIAGRAMS/ATTACHMENTS

Attachment 1 - Immediate Notification Requirements

Attachment 2 - Technical Specification and ODCM Special Reports

Attachment 3 - Routine Reports

Attachment 4 - Event Reports (Other than LERs)

Attachment 5 - One Hour Notifications - Sample Wording

Attachment 6 – SAMPLE Reactor Plant Event Notification Worksheet

Attachment 7 - Reactor Plant Event Notification Worksheet

Attachment 8 - Reportability Evaluation (REW) Worksheet

IMMEDIATE NOTIFICATION REQUIREMENTS

The following tables are divided into sections based upon the time allowed for reporting the respective events as follows:

I One Hour Notifications
II Four Hour Notifications
III Eight Hour Notifications
IV Twenty-four Hour Notifications

NOTE: The events listed in this attachment may be concurrent with conditions that result in a declared emergency. In the case of a declared emergency, the notification made under the Emergency Plan and implementing procedures satisfies the notifications required by this procedure (10 CFR 50.72(a)). Written reports will be based on §50.73 and Technical Specifications regardless of whether the initial notification is made under the Emergency Plan or this procedure.

I. ONE HOUR NOTIFICATIONS

I.A. OPERATIONAL EVENTS -10 CFR 50.72 (b) (1)

1. Technical Specification Deviations (10 CFR 50.54x)
2. Safety Limit Violation (TS 6.7.1)

I.B. RADIOLOGICAL EVENTS

1. Radioactive Shipments (Note 1)
2. Loss or Theft of Licensed Material/Radiological Sabotage (Note 2)
3. Exposure to Individuals or Releases (Note 3)
4. Accidental Criticality (Note 4)

I.C. SECURITY EVENTS (Note 5)

1. Security Events per SEC-NGGC-2147.
2. International Atomic Energy Agency (IAEA) Representative

I.D. FITNESS FOR DUTY (Note 6)

1. FFD - NRC Employee

IMMEDIATE NOTIFICATION REQUIREMENTS

II. FOUR HOUR NOTIFICATIONS

OPERATIONAL EVENTS 10 CFR 50.72 (b) (2)

1. Initiation of any Nuclear Plant Shutdown required by Technical Specifications.

2. Unplanned Actuation of the reactor protection system (scram) when the reactor was critical and any event that results or should have resulted in ECCS discharge into the RCS.

3. Off-Site Notification Has Been or Will Be Made (Note 12)

III. EIGHT HOUR NOTIFICATIONS

1. Degraded or Unanalyzed Condition

2. Loss of Emergency Response Capability (Note 7)

3. Unplanned Actuation of selected ESF Systems
   Refer to NUREG 1022 System Actuation to identify applicable system actuations.

4. Loss of a Safety Function

5. Transport of a Potentially Contaminated Individual

6. Fatality or Hospitalization (Note 13)

IV. TWENTY-FOUR HOUR NOTIFICATIONS

1. EXPOSURE TO INDIVIDUALS OR RELEASES

   a. Radiological Exposure/Release (Note 8)

   b. Unusual or Important Environmental Events (Note 9)

2. VIOLATION OF OPERATING LICENSE CONDITIONS (Note 10)

3. FITNESS FOR DUTY PROGRAM EVENTS (Note 11)

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

	NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
1.	RADIOACTIVE SHIPMENTS		
	a) Removable contamination from a received package containing radioactive material in excess of the limits specified in §71.87(i). The involved RP Supervisor shall immediately notify the final delivery carrier.	§20.1906(d)(1) §71.87(i)	
	b) Radiation levels from a received package of radioactive material in excess of the limits specified in §71.47. The involved RP Supervisor shall immediately notify the final delivery carrier.	§20.1906(d)(2) §71.47	
	c) Security related events with respect to the transport of special nuclear material are handled via SEC-NGGC-2147. Security threats or theft of licensed material shall be reported to site Security personnel.	§73.71(b)(2) §73 Appendix G §73.71(a)(5)	
2.	LOSS OR THEFT OF LICENSED MATERIAL/ RADIOLOGICAL SABOTAGE		
	Note: Theft and Sabotage are Security Events handled by SEC-NGGC-2147.		
	Any loss or theft or attempted theft of:		
	a) Licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in Appendix C to §20.1000-§20.2401 under such circumstances that it appears that an exposure could result to persons in unrestricted areas,	§20.2201(a)(1)(i) §20.2201(d)	30-Day Written Report required per §20.2201(b)
	b) Any Special Nuclear Material or spent fuel,	(theft – See SEC-NGGC-2147) §74.11 §150.16(b) §73.71(a)	60-Day Written Report required per §73.71(a) or (b) – See SEC-NGGC-2147 15-Day Written Report may be required per §150
	c) Recovery of or accounting for loss of any shipment of Special Nuclear Material or spent fuel	§73.71(a)	
	d) Greater than 10 curies of tritium at any one time or 100 curies in one calendar year, or	§30.55(c)	15-Day Written Report required
	e) More than 15 pounds of uranium or thorium at any one time or more than 150 pounds in one calendar year.	§40.64(c) §150.17(c)	15-Day Written Report required

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

	NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
3.	EXPOSURE TO INDIVIDUALS OR RELEASES		
	Any event involving by-product, source or Special Nuclear Material that may have caused or threatens to cause:		
	a) An individual to receive: 1) A total effective dose equivalent of ≥25 Rem 2) An eye dose equivalent of ≥75 Rem 3) A shallow-dose equivalent to the skin or extremities of ≥250 Rad 4) An intake of 5 ALI in 24 hours	§20.2202(a)(1)	LER required by §50.73(a)(2)(viii), (a)(2)(ix) and §20.2203
	b) Release of radioactive material in excess of Technical Specification Instantaneous Limits shall be declared an emergency in accordance with PEP-310. The reporting requirements of PEP-310 shall take precedence over the less restrictive times for reporting requirements of §20.2202 and §50.72(b)(2) for releases.	§20.2202(a)(2) §50.72(b)(2)(iv)	LER required by §50.73(a)(2)(viii), (a)(2)(ix) and §20.2203
4.	ACCIDENTAL CRITICALITY		
	Accidental criticality of special nuclear material.	§70.52(a)	None
5.	SECURITY EVENTS **Note:** Reporting of Security Events (Including Safeguards and Fitness-For-Duty Events) is per SEC-NGGC-2147. Notify site Security personnel.		
	INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA) REPRESENTATIVE	§75.7	None
	Individual claiming to be an IAEA representative who is not accompanied by an NRC employee and has no prior confirmation of credentials in writing.		
	Notification is by telephone to Director, Office of Nuclear Reactor Regulation	§75.6 and §75.7	
6.	FITNESS FOR DUTY - NRC EMPLOYEE Notification of NRC employee's unfitness for duty. Per §26.27(d), the appropriate Regional Administrator must be notified immediately by telephone. During other than normal working hours, the NRC Operations Center must be notified.	§26.27(d)	None

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

NOTE :	PLP-717, Equipment Important To Emergency Preparedness and ERO Response, Attachment 2, Essential ERO Equipment and Compensatory Measures, contains guidance on determining immediate reportability. Other Technical Specification and ODCM reporting requirements may apply and are located in Attachment 2 of this (AP-617) procedure. **[CR 580945 CAPR]**

	NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
7.	LOSS OF EMERGENCY RESPONSE CAPABILITY	§50.72(b)(3)(xiii)	None

Any event that results in a major loss of assessment capability, offsite response capability, or communications capability (e.g., significant portion of Control Room indication, Emergency Telecommunication System, or offsite notification system).

This may include loss of any of the following:

a) Emergency Response Facilities
b) Radiation Monitors and Plant Equipment used in identification of Emergency Action Levels
c) Computers and Telecommunications including:
   1. Selective Signaling
   2. NRC Emergency Telecommunication System
   3. Emergency Response Data System
   4. PABX telephone system
   5. Plant PA System
   6. Corporate Telephone Communication System (Voicenet) and the Commercial Telephone System
   7. Satellite Phones
   8. Sound Powered Phone System
   9. HNP Emergency Notification (Everbridge) System
   10. Emergency Response Facility Information System (ERFIS)
   11. Safety Parameter Display System (SPDS)
   12. Dose Assessment Software (RASCAL)
d) Sirens and Tone Alert Radios

Use PLP-717, Attachment 2 in determination of immediate reportability.

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

	NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
8.	RADIOLOGICAL EXPOSURE/RELEASE	§20.2202(b)	30-Day Written Report Required per §20.2203
	Any event involving licensed material possessed by the licensee that may have caused or threatens to cause an individual to receive, in a period of 24 hours: a) A total effective dose equivalent > 5 Rem; or b) An eye dose equivalent > 15 Rem; or c) A shallow-dose equivalent to the skin or extremities> 50 Rem; or d) An intake of > 1 ALI.		
9.	UNUSUAL OR IMPORTANT ENVIRONMENTAL EVENTS		
	Any event that indicates or could result in significant environmental impact causally related to plant operation. Examples are:	Env. Prot. Plan, (Operating License Appendix B) Section 4.1, and PLP-500	Local, State & Federal Agency Notifications defined in PLP-500 require NRC notification within 4 hours per 50.72 (See Note 12)
	a) Excessive bird impaction	ESS determines threshold	If event is significant and not reportable to a Local, State, or Federal agency, a 24 hour NRC notification may still be required per Env. Prot. Plan.
	b) Onsite plant or animal disease outbreak	ESS determines threshold	
	c) Mortality or unusual occurrence of Endangered Species	ESS determines threshold	
	d) Fish Kills	Local and State Notifications defined in PLP-500	30-Day Follow-up written report required per Env. Prot. Plan Subsection 5.4.2
	e) Increase in nuisance organisms or conditions	ESS determines threshold	
	f) Unanticipated or emergency discharge of waste water or chemical substances	Local and State Notifications defined in PLP-500	
	g) Damage to vegetation resulting from cooling tower drift deposition	ESS determines threshold	
	h) Station outage or failure of any cooling water intake or service water system components due to bio-fouling by Corbicula (Asiatic Clam)	ESS determines threshold	

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
10. VIOLATION OF OPERATING LICENSE CONDITIONS		
a) Any event resulting in the plant operating in a manner which violates the SHNPP Facility Operating License, Section 2.C:	OL Section 2.G	60-day LER required per 10 CFR 50.73 and 10 CFR 50.4(e)
(1) Intentionally raising power above 2948 MWt (100%) for any period of time. (2) Failure to reduce thermal power to less than or equal to 2948 MWt when the 2-hour average exceeds 2948 MWt. (3) Permitting the core thermal power 8-hour average to exceed 2948 MWt. (4) Failure to take prudent action prior to a pre-planned evolution that could cause a power increase to exceed 2948 MWt (example: Scheduled securing of a Heater Drain Pump without first reducing power to accommodate the expected power increase. A short term increase in transient power above 2948 MWt following a boron dilution is not included if actions to reduce power are taken in a reasonable time following the dilution reactivity transient). Note: No actions are allowed that would intentionally raise core thermal power above 2948 MWt for any period of time. Small, short-term fluctuations in power that are not under the direct control of a license reactor operator or result from actions taken for a different purpose (example: temperature control) are not considered intentional.	NEI Position Statement: Guidance to Licensees on Complying with the Licensed Power Limit (NRC ADAMS Accession No. ML081750537)	
b) A failure to comply with the following administrative requirements (See Note 1):	OL Section 2.G	60-day LER required per 10 CFR 50.73 and 10 CFR 50.4(e)
1) Deviation from the requirements of the Environmental Protection Plan;	OL Section 2.C.2	
2) Failure to comply with anti-trust conditions of Appendix C to OL;	OL Section 2.C.3	
3) Failure to comply with new fuel storage requirements.	OL Section 2.C.10	

NOTES
IMMEDIATE NOTIFICATION REQUIREMENTS

	NOTIFICATION	REFERENCE	WRITTEN FOLLOW-UP
11.	FITNESS FOR DUTY PROGRAM EVENTS		
Note:	Reporting of Security Events (Including Safeguards and Fitness-For-Duty Events) is per SEC-NGGC-2147. Notify site Security personnel.		
	a) Sale, use, or possession of illegal drugs within the protected area.	§26.73(a)(1)	None
	b) Any acts by any person licensed under §55, or by any supervisory personnel assigned to perform duties within the scope of §26 1) Involving the sale, use, or possession of a controlled substance, 2) Resulting in a confirmed positive test on such persons, 3) Involving use of alcohol within the protected area, or 4) Resulting in a determination of unfitness for scheduled work due to the consumption of alcohol.	§26.73(a)(2)	None
	c) False positive error on a blind performance test specimen when error is determined to be administrative.	App. A to Part 26 B.2.8(e)(5) Reference 2.35	Non-docketed correspondence to NRC FFD coordinator
12.	OFF SITE NOTIFICATION HAS BEEN OR WILL BE MADE		
	Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.	§50.72(b)(2)(xi)	
13.	FATALITY OR HOSPITALIZATION		
	a) See SAF-SUBS-00033 and contact the Site Safety Representative	SAF-SUBS-00033	See Note 12 for NRC required 4-hour notification
	b) OSHA must be notified within 8 hours of: 1) Workplace Fatality 2) Workplace incident with 3 or more personnel hospitalized		
	c) North Carolina requires a call to the Dept. of Labor, Elevator Division, within 24 hours of an injury or fatality related to elevators.		

TECHNICAL SPECIFICATION AND ODCM SPECIAL REPORTS

A Special Report may be identified from CRs, EIRs, declared emergencies or as the result of equipment inspections. The following steps shall be performed when it appears that a Special Report is required.

1. The Supervisor - Licensing/Regulatory Programs shall be notified of the event if this has not already occurred via a CR or EIR.
2. The Supervisor - Licensing/Regulatory Programs shall inform the General Manager - Harris Plant and applicable unit manager(s) of the need for a special report.
3. The Supervisor - Licensing/Regulatory Programs shall assign action items to the responsible units per Reference 2.12 to provide input for the required reports.
4. Completed reports shall be routed for approval per Reference 2.2.
5. A copy of the completed special report shall be provided to the Secretary - PNSC for review at a subsequent PNSC meeting.
6. The special report shall be transmitted as a QA Record.

REPORTING REQUIREMENTS

SUBJECT	TS [ODCM] REFERENCE	RESP UNIT	TIMING OF RESPONSE
Leak or boron deposit found during inspection	O.L. NRC Order dated 2/11/03	HESS	60 days after returning the plant to operation
Moderator Temperature Coefficient more positive than specified limits	3.1.1.3	HESS	10 Days
Remote Shutdown Monitoring Instrument inoperable for greater than 60 days	3.3.3.5.a	Maint	14 Days
Radiation Monitors, Pressurizer Safety Valve Position Indicators or Subcooling Margin Monitors inoperable for greater than 7 days. (Also see OWP-ERFIS)	3.3.3.6	Maint or IT (and Engineering for Rad. Monitors)	14 Days
PORV/RCS Vents used to mitigate RCS pressure transient at low temperature	3.4.9.4	Operations	30 Days
An ECCS actuation and injection of water into the Reactor Coolant System (See Note 2)	3.5.2, 3.5.3	Operations	90 Days (See Note 3)
Change to Sample Plan Used for Snubber Functional Testing	3.7.8 PLP-106 Att. 4	HESS	Before Implementation

TECHNICAL SPECIFICATION AND ODCM SPECIAL REPORTS

SUBJECT	TS [ODCM] REFERENCE	RESP UNIT	TIMING OF RESPONSE
Radioactive Material in Liquid Holdup Tanks exceeding limits	3.11.1.4	Environmental and Chemistry	Include in Annual Radioactive Effluent Release Report
Use of non-preferred Incore Detectors when evaluating QPTR	4.2.4.2 (Bases)	HESS	30 Days
Abnormal degradation of Containment Vessel structure detected during required inspections	4.6.1.6.2	HESS	15 Days
Sealed Source leakage test results	4.7.9.3	Radiation Protection	Annually if removable contamination greater than 0.005 µCi detected
Greater than 30 total rods or 10 rods per fuel assembly replaced with filler rods or vacancies during any single refueling.	5.3.1	HESS	30 Days Following Start-up
Safety Limit Violation.	6.7.1	Operations	14 Days
Startup Report following: 1) License amendment to increase power level. 2) Installation of fuel of different design or manufacturer. 3) Modifications that significantly alter the nuclear, thermal or hydraulic characteristics of the unit.	6.9.1.1	HESS	90 Days Following Resumption of Commercial Operations or Completion of Startup Test Program, or 9 months after Initial Criticality, whichever is earliest. Supplementary reports required each 3 months.
Change to Core Operating Limits Report	6.9.1.6.4	HESS	Upon issuance; must be submitted no later than the date of implementation.
Steam Generator Tube Inspection Report	6.9.1.7	HESS	Within 180 days after initial entry into Hot Shutdown following completion of an inspection performed in accordance with TS 6.8.4.l.
Special Reports	6.9.2	As Assigned	As Requested
Unreviewed Environmental Question	T.S. Appendix B EPP Section 3.1	Environmental and Chemistry	Before implementation of change

TECHNICAL SPECIFICATION AND ODCM SPECIAL REPORTS

SUBJECT	TS [ODCM] REFERENCE	RESP UNIT	TIMING OF RESPONSE
Proposed changes/renewal application for NPDES Permit	T.S. Appendix B EPP Section 3.2	Environmental and Chemistry	At time of submittal to the permitting agency
Changes to/renewal of NPDES Permit or State Certification	T.S. Appendix B EPP Section 3.2	Environmental and Chemistry	30 Days after change/renewal
Stay of NPDES Permit or State Certification	T.S. Appendix B EPP Section 3.2	Environmental and Chemistry	30 Days following stay
Unusual or Important Environmental Events	T.S. Appendix B EPP Sections 4.1 and 5.4.2	Environmental and Chemistry	30 Days after event. (Note 4) See also 24 hour notifications
Seismic Monitoring Instrument inoperable for greater than 30 days	PLP-114	Maint	10 Days
Actuation of Seismic Monitoring Instruments during seismic event greater than or equal to 0.01g (See Note 1)	PLP-114	HESS	14 Days
Meteorological Monitoring Instrument inoperable for greater than 7 days	PLP-114	Maint	10 Days
Metal Impact Monitoring System Channel(s) inoperable for greater than 30 days.	PLP-114	Maint	10 Days
Explosive gas monitoring instrument inoperable for greater than 30 days.	PLP-114	Maint	30 Days (Note: No time specified in PLP-114)
Area Temperatures exceeding PLP-114, Attachment 4 limits by more than 30°F, or for greater than 8 hours	PLP-114	HESS & Operations	30 Days
A calculated dose to a member of the public from the release of radioactive materials in liquid effluents to an unrestricted area exceeding limits	[3.11.1.2]	Environmental and Chemistry	30 Days
Radioactive liquid waste being discharged without treatment and in excess of limits and any portion of the liquid radwaste treatment system not in operation	[3.11.1.3]	Environmental and Chemistry & Operations	30 Days
Calculated air dose in gaseous effluent exceeding limits in areas at or beyond site boundary	[3.11.2.2]	Environmental and Chemistry	30 Days

TECHNICAL SPECIFICATION AND ODCM SPECIAL REPORTS

SUBJECT	TS [ODCM] REFERENCE	RESP UNIT	TIMING OF RESPONSE
Calculated dose to a member of the public from a release of gaseous effluents containing Iodine 131, Iodine 133, tritium and radionuclides in particulate form with half-lifes greater than eight days exceeding the limits.	[3.11.2.3]	Environmental and Chemistry	30 Days
Untreated radioactive gaseous waste discharged in excess of limits and any portion of the gaseous radwaste treatment system not in operation.	[3.11.2.4]	Environmental and Chemistry & Operations	30 Days
Calculated dose from release of radioactive material in liquid or gaseous effluents, to a member of the public in excess of limits.	[3.11.4]	Environmental and Chemistry	30 Days
Level of radioactivity, as a result of plant effluent in a specified location, exceeding the reporting levels of ODCM O.R. Table 3.12-2 when averaged over any calendar quarter.	[3.12.1]	Environmental and Chemistry	30 Days

NOTES:

1. Refer to the following TS Surveillance Requirements following any seismic event: 4.3.3.3.2 and 4.4.5.3.c.2
2. Refer to the following TS Surveillance Requirements following any safety injection actuation: 4.4.5.3.c.1, 4.4.5.3.c.3, 4.4.5.3.c.4, 4.4.6.2.2.d, and 4.5.2.g.1
3. LER also required (see Reference 2.6). The information required by the Special Report exceeds the requirements of the LER.
4. Events requiring reports to other government agencies shall be reported per those requirements in lieu of the Environmental Protection Plan. A copy of the report shall be sent to the NRC.

Attachment 3
Sheet 1 of 3

ROUTINE REPORTS

The SHNPP Technical Specifications, ODCM, §20 and §50 require that reports be provided to the NRC at routine intervals. The following steps apply to routine reports:

1. The Supervisor - Licensing/Regulatory Programs shall establish action items per Reference 2.12 for these reports.
2. As applicable, the responsible unit and Licensing/Regulatory Programs shall establish standard formats for a routine report.
3. Routine reports shall be routed for approval per Reference 2.2.
4. Routine reports shall be transmitted as a QA record.

NOTE: TS = Technical Specification
OR = ODCM Operational Requirement

SUBJECT	REGULATORY REFERENCE	RESP UNIT	TIMING OF RESPONSE
Annual Operating Report	T.S. 6.9.1.2	Lic / Reg Prog	Before 3/1
Annual Radiological Environmental Operating Report	T.S. 6.9.1.3 O.R. 3.12.1 O.R. Table 3.12-1 O.R. Table 4.12-1 O.R. 4.12.2 O.R. 3.12.3 O.R. 4.12.3	Environmental and Chemistry	Before 5/1
Annual Environmental Operating Report	Env. Prot. Plan 5.4.1	Environmental and Chemistry	Before 5/1
Annual Radioactive Effluent Release Report	T.S. 3.11.1.4 T.S. 6.9.1.4 T.S. 6.14.c ODCM App. F.3 O.R. Table 4.11-1 O.R. 3.12.1 O.R. Table 3.12-1 O.R. 3.12.2 O.R. 3.3.3.10 O.R. 3.3.3.11 §50.36a(a)(2)	Environmental and Chemistry	Before 5/1

ROUTINE REPORTS

NOTE 1: This report does not need to be routed for approval per Reference 2.2 since it is not correspondence to a regulatory agency.

SUBJECT	REGULATORY REFERENCE	RESP UNIT	TIMING OF RESPONSE
Consolidated Data Entry (CDE) Report	FSAR Section 1.8 (Reg. Guide 1.16)	Lic / Reg Prog	By the end of the 1st month following each quarter
Individual Worker Radiation Dose Report (To employee) (See Note 1 above)	§19.13(b)	ESS Rad Services (Dosimetry)	Annually
Annual Exposure Report for Individual Monitoring	§20.2206(b)	Lic / Reg Prog	Annually, by April 30th
Semiannual Fitness for Duty Program Performance Data Analysis Report	§26.71(d)	Nuclear Operations	Within 60 days of the end of each 6-month reporting period (Jan-June and July-Dec)
ECCS evaluation model changes or errors where sum of absolute magnitudes of changes results in PCT change ≤50°F	§50.46(a)(3)	NFM&SA	Annually
Changes to QA Program which do not reduce commitments	§50.54(a)(3)	NAS	In accordance with §50.71(e)
Insurance and Financial Security Annual Report	§50.54(w)(3)	Nuclear Operations	Annually, on April 1
In-service Inspection Summary	§50.55a (ASME Section XI IWA-6230)	HESS	90 days after completion of inspections
FSAR Update, facility changes, tests, and experiments conducted without prior approval	§50.59(b) §50.71(e)	Lic / Reg Prog	Six months following each refueling outage. Interval not to exceed 24 months between updates

ROUTINE REPORTS

SUBJECT	REGULATORY REFERENCE	RESP UNIT	TIMING OF RESPONSE
Annual Financial Report, including certified financial statements	§50.71(b)	Nuclear Operations	Upon issuance of the report (Normally April 30)
Status of Decommissioning Funding	§50.75(f)	Nuclear Operations	March 31, 1999 and at least once every two years thereafter (frequency becomes annual when the plant is within 5 years of projected end of operation or when the plant is involved in mergers or acquisitions).
Simulator - Report of uncorrected performance test failures and schedule for correction	§55.45(b)(5)(ii)	Training	Every 4 years on anniversary of certification
Material Status Reports (old Forms DOE/NRC-742 and 742(c))	§74.13(a)(1) §70.53(a)(1) §40.64(b) §150.17(b)	HESS	30 days after March 31 and September 30 NOTE: 40.64 and 150.17 require statement of foreign origin source material.
QA Program for Transportation of Radioactive Material Packages	§71.101	Nuclear Operations	Every 5 Years. Docket 71-0345
Financial Protection - Guarantee of payment of deferred premiums	§140.21	Nuclear Operations	Annually, on anniversary date on which indemnity agreement is effective (Normally April 30)

EVENT REPORTS (Other than LERs)

Title 10 of the Code of Federal Regulations and other requirements require that reports be provided to the NRC and other regulatory agencies based on the occurrence of specific events. The following instructions apply to these events (unless otherwise noted):

1. The responsible unit shall determine if written procedures should be prepared to implement the reporting requirement.
2. When written reports are required to be submitted to the NRC, the General Manager - Harris Plant and the Supervisor – Licensing/Regulatory Programs shall be informed.
3. Completed reports shall be routed for approval per Reference 2.2.
4. The event report shall be transmitted as a QA Record.

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Report to former radiation worker of worker's exposure to radiation (Note 7)	19.13(c)	Radiation Protection	Upon request; within 30 days from request or 30 days after exposure has been determined
Radiation Exposure Data to Individual-Overexposure (Note 7)	19.13(d)	Radiation Protection	Upon overexposure (Note 5)
Radiation Exposure Data to Terminating Employees (Note 7)	19.13(e)	ESS Rad Services (Dosimetry)	Upon request at termination
Bioassay Services to Determine Exposure	20.1204(c)	Radiation Protection	As requested by NRC
Report of planned special exposure	20.1206(f) 20.2204	Radiation Protection	Within 30 days following planned special exposure
Report to individual of planned special exposure (Note 7)	20.1206(g)	Radiation Protection	Within 30 days from planned special exposure
Respiratory Protection Program Equipment not certified by NIOSH/MSA	20.1703(d)	Radiation Protection	30 Days prior to equipment usage
Theft or loss of licensed material greater than 10 times Appendix C quantities	20.2201(a)(1)(ii) 20.2201(b)	Radiation Protection Radiation Protection	30 Days (by ETS) 30 Days (written follow-up)
Additional information on theft or loss	20.2201(d)	Radiation Protection	30 Days

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Reports of Overexposures/ Excessive Levels and Concentrations	20.2203 40CFR190 ODCM O.R. 3.11.4	Radiation Protection	30 Days (Note 5)
Missing Waste Shipment Trace Investigation	20 App. G, Sec. III.E	Radiation Protection	2 weeks after completion of Investigation
Interim evaluation report of identified deviation or failure to comply (when evaluation cannot be completed within 60 days of discovery)	21.21(a)(2)	Lic / Reg Prog	Within 60 days of discovery
Failure to Comply or Existence of a Defect (Refer to AP-616)	21.21(c)(1) & (d)(3)	Lic / Reg Prog	2 days, written follow-up within 30 days
FFD testing more conservative than §26 requirements	26 App. A, Sec. A.1.1(2)	Human Res	Within 60 days of implementing change
FFD - Unsatisfactory performance testing of a certified laboratory	26 App. A, Sec. B.2.8 (e)(4)	Human Res	30 days
Reports required as condition of Parts 30-35 licenses (By-product Material)	30.34(e)(4)	Radiation Protection	As specified in license
Renewal or non-renewal of Parts 30-35 licenses	30.36	Radiation Protection	30 Days prior to expiration
Notification of byproduct (Part 30) or SNM (Part 70) license expiration or cessation of principal activities	30.36(d) 70.38(d)	Radiation Protection	Within 60 days
Amendment to Part 30-35 License	30.38	Radiation Protection	As Required
Failure of or damage to shielding, on-off mechanism or indicator; detection of removable radioactive material	31.5(c)(5)	Radiation Protection	5 Days (5 Day report required per Byproduct Materials License in lieu of 30 day requirement)
Transfer of device to specific or general licensee	31.5(c)(8) 31.5(c)(9)(i)	Radiation Protection	30 Days
Leaking of sealed radiographic source	34.27(d)	Licensed Radiographer	5 Days

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Incidents involving radiographic equipment: • unintentional disconnection of the source assembly from the control cable. • inability to retract the source assembly to its fully shielded position and secure it in this position. • failure of any component (critical to safe operation of the device) to properly perform its intended function.	34.101(a)	Licensed Radiographer	Within 30 days of occurrence
Licensee identified information which has significant implications for public health and safety or the common defense and security	50.9(b) 30.9(b) 40.9(b) 70.9(b) 71.7(b)	Lic / Reg Prog	As necessary (Note 4); §70.9(b) specifies within two working days of identifying the information
Change to the LOCA analysis which results in a change to the Peak Clad Temperature	50.46(a)(3)	NFM&SA	If ≤50°F, then include in the annual report. If >50°F, then 30 days.
Changes in QA Program which reduce commitments	50.54(a)	NAS	Before implementation
Request for Written Information	50.54(f)	As Specified	As Requested
Changes to Operator Requalification Program which decreases scope, time allotted or frequency of conducting portions of the program	50.54(i-1)	Training	Before Implementation
Changes in Security Plan, Guard Training and Qualification Plan, or Safeguards Contingency Plan made without prior approval	50.54(p) 70.32(g)	Security	Before implementation if changes reduce effectiveness of plan; otherwise, within two months after change. **NOTE:** §70.32(g) specifies within 60 days for safeguards contingency plan.

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Changes in emergency plan or implementing procedures made without prior approval	50.54(q) 50 App E(V)	Emerg Prep	Before implementation if changes reduce effectiveness of plan, otherwise within 30 days after change
Notification of safe and stable condition of reactor and no significant risk to public health and safety	50.54(w)(4)(ii)	Emerg Prep	After attaining safe and stable condition (following an accident with costs >$100 million)
Cleanup plan for decontaminating reactor to permit resumption of operation or commencement of decommissioning	50.54(w)(4)(ii)	Emerg Prep	Within 30 days of notification that reactor is in safe and stable condition (following an accident with costs >$100 million)
Plan for management of and notification of significant change in the proposed waste management program (of irradiated fuel at the reactor, after expiration of license and until transferred to DOE)	50.54(bb)	Lic / Reg Prog	Within 2 years after cessation of operation or 5 years before expiration of OL and after making change in program
Filing of petition for bankruptcy (Title 11 of US Code)	50.54(cc)(1) 30.34(h) 40.41(f) 70.32(a)(9)(i)	Lic / Reg Prog	Immediately following filing
Notification that conformance to a certain Code required by Section XI of the ASME B&PV Code and Addenda for inservice test is impractical	50.55a(f)(5) (iii)	HESS	After identifying problem
Determination that a pump or valve test required by Section XI of the ASME B&PV Code and Addenda is impractical and not included in the revised inservice test program	50.55a(f)(5) (iv)	HESS	No later than 12 months after expiration of initial 120-month period of operation, and each subsequent 120-mo. period of operation during which the test is determined to be impractical

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Notification that conformance to a certain Code required by Section XI of the ASME B&PV Code and Addenda for inservice inspection is impractical	50.55a(g)(5)(iii)	HESS	After identifying problem
Determination that a pump or valve test required by Section XI of the ASME B&PV Code and Addenda is impractical and not included in the revised inservice inspection program	50.55a(g)(5)(iv)	HESS	No later than 12 months after expiration of initial 120-month period of operation, and each subsequent 120-mo. period of operation during which the test is determined to be impractical
Updated assessment of projected value for RT_{PTS} for reactor vessel beltline materials - after significant change in projected value of RT_{PTS} or change in facility's operating expiration date	50.61(b)(1)	HESS	After change
Plan for thermal annealing of reactor vessel	50.66	HESS	3 yrs prior to date when fracture toughness criteria would be exceeded
Reports required as a condition of license	50.71(a)	As Specified	As Specified in License
Telephone report in lieu of LER for an invalid actuation of a system listed in 50.73(a)(2)(iv)(B), other than RPS actuation when critical	50.73(a)(1)	Operations	60 Days
Reassignment of Licensed Operator to position not requiring license	50.74(a)	Operations	30 Days
Termination of Licensed Operator	50.74(b)	Operations	30 Days
Hardware and software changes that affect transmitted data points identified in ERDS Data Point Library	50 App. E, Sec. VI.3.a	Nuclear Info Systems	Within 30 days after changes are completed
Hardware and software changes (except data point modifications) that could affect transmission format and computer communication protocol to the ERDS	50 App. E, Sec. VI.3.b	Nuclear Info Systems	As soon as practicable and at least 30 days prior to making modification

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Test methods for supplemental fracture toughness tests	50 App. G Sec. III.B	HESS	Submitted and approved prior to testing
Fracture Toughness	50 App G, Sec. IV.A.1	HESS	3 years before date when predicted fracture toughness will no longer satisfy App G
Report of test results of specimens withdrawn from capsules (fracture toughness tests)	50 App H, Sec. IV	HESS	Within One Year of Withdrawal
Report of effluents released in excess of one-half design objective exposure	50 App I, Sec. IV.A	Environmental and Chemistry	Within 30 Days from end of quarter (Special Report required by T.S. 3.11.4 satisfies this requirement)
Reactor Building ILRT	50 App J, Option A, Sec. V.B TS 4.6.1.2	HESS	3 Months After Test (No time specified in regulation or TS)
Certification of Medical Fitness	55.23 55.31	Training	Upon Application
Incapacitation Because of Disability or Illness	55.25 50.74(c)	Operations	30 Days after learning of diagnosis
Application for Operator License	55.31	Training	As necessary
Reapplication for Operator License	55.35	Training	Two months after first denial, six months after second denial, two years after third and subsequent denials
Conviction of a Felony for Licensed Operator	55.53(g) 73.71(b)	Operations/ Security	30 Days
Application for Operator License Renewal	55.57	Training	As necessary
Reports of Conditions of Part 70 License	70.32(b)(5) 75.36	Security	As specified in license
Changes in plan for Physical Protection of SNM in transit made without prior approval	70.32(d)	Security	2 Months

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Accident notification report required by DOT on transportation of licensed material	71.5(a)(1)(iv) 49CFR171.15 & 16	Radiation Protection	Carrier is to provide notice to DOT at the earliest practicable moment and written follow-up within 30 days
Transportation Package Information	71.12(c)(3) 71.101(f)	Radiation Protection	Before first use (Note 1)
Deviations related to Type B package for transport of radioactive material, specifically: • significant reduction in effectiveness of Type B packaging during use • defects with safety significance in Type B packaging after first use, or the means employed to repair the defects and prevent recurrence • conditions of approval of certificate of compliance not observed in making a shipment	71.95	Radiation Protection	Within 30 days (NOTE: requirement is for licensee to report)
Advance notification of shipment of Irradiated Fuel, Nuclear Waste, or certain shipments of SNM	71.97(a) 73.37(b)(1) 73.72(a) 73.73(a) 73.74(a)	Radiation Protection	Before shipment (Note 2)
Revision notice or cancellation notice for shipment of irradiated fuel, nuclear waste, or certain shipments of SNM	71.97(e) 71.97(f) 73.37(f) 73.72(a)(5) 73.73(b) 73.74(b)	Radiation Protection	Upon change/ cancellation (Note 2)
Advance Notice and Approval of Routes for Shipment of Irradiated Fuel	73.37(b)(7) 73.37(f)	Radiation Protection & Nuclear Operations	Before shipment (Note 2)
Theft or unlawful diversion or attempted theft or unlawful diversion of SNM or spent fuel	73.71 74.11 150.16(b)	Security	As required (See also 1 hour notifications and SEC-NGGC-2147)

EVENT REPORTS (Other than LERs)

SUBJECT	10 CFR Reference or (other)	RESP UNIT	TIMING OF RESPONSE
Results of Trace Investigation of Lost or Unaccounted for Shipment of SNM	73.71(a) 74.11	Security	30 Days
Threat to or reduced effectiveness of physical security	73.71(d)	Security	30 Days (See also 1 hour notifications and SEC-NGGC-2147)
Nuclear Material Transfer (old Form DOE/NRC-741)	74.15(a) 40.64(a) 70.54 150.16(a) 150.17(a)	HESS	Whenever transfer occurs (Note 6)
Earthquake exceeding Operating Basis Earthquake values	100 App. A, Sec. V(A)(2)	HESS	Prior to resuming operations
Import or export of nuclear equipment or material	110	Radiation Protection	Varies; see Part 110
Bodily injury or property damage from possession or use of radioactive material resulting in an indemnity claim	140.6(a)	Radiation Protection	As promptly as practical
Change in proof of financial protection or other financial information filed with the NRC	140.15(e)	Treasury	Promptly
Termination of liability insurance policy used for financial protection (notification of renewal or other proof of financial protection)	140.17(b)	Treasury	At least 30 days prior to termination
Failure of High Integrity Container or Notification of Misuse of a High Integrity Container	PLP-300	Radiation Protection	Within 30 days of knowledge of the incident
Cooling Tower Beacon Outage greater than 30 minutes or restoration from an outage greater than 30 minutes. (Note 8)	FAA Advisory Circular AC 70/7460-1K	Operations	Upon Discovery
Level of radioactivity in onsite groundwater, exceeding the reporting levels of ODCM O.R. Table 3.12-2 for drinking water.	CHE-NGGC-0057	Environmental and Chemistry	Within 30 days of discovery

EVENT REPORTS (Other than LERs)

NOTES:

1. Development and use of a package will require special reporting requirements per §71.5, 71.95 and 71.101(f).
2. Notification to NRC received at least 10 days before transport of the shipment commences; see §73.72(a), 73.73(a), or 73.74(a) for additional details. State Governor(s) of states through which material is to be shipped shall also be notified by mail postmarked at least 7 days before shipment or by messenger 4 days prior to shipment. Notification of any subsequent schedule changes of greater than six (6) hours or cancellation of shipment shall also be made before the change (See §71.97(c) for additional notification to the Regional NRC Administrator).
3. Not used
4. Report not required if such reporting would duplicate information already submitted per other NRC reporting requirements.
5. When reporting exposure of an individual, the individual shall also be notified not later than the transmittal to the NRC.
6. §40.64(a) specifies next working day for transfers and 10 days for receipt of foreign origin source material. §150.16(a) and 150.17(a) specify within 10 days after material is received.
7. This report does not need to be routed for approval per Reference 2.2 since it is not correspondence to a regulatory agency.
8. Notify the DOT-FAA Flight Service Station at either 1-877-487-6867 or 1-800-992-7433. The following information will be required:
    a. Harris Nuclear Plant, caller name and telephone number
    b. Which Hyperbolic Cooling Tower Beacon Warning Lights (by compass orientation) are inoperable
    c. Location - Plant location is latitude 35°38'01"N, longitude 78°57'23"W and a distance of 16 miles Southwest of Raleigh
    d. Height - The Cooling Tower is 523 feet above ground level. The height of the Cooling Tower above sea level is 784 feet.
    e. Estimated return to service date

    Determine from the Flight Service Station the name of the individual contacted and when a follow-up call should be made. Document the notification and any required follow-up in an NCR with Licensing as the responsible organization.

One Hour Notifications - Sample Wording for the Description Field
(Licensing will review notifications for follow-up clarification as needed.)

I.A. OPERATIONAL EVENTS -10 CFR 50.72 (b) (1)

- Technical Specification Deviations (10 CFR 50.54x)

DEVIATION FROM TECHNICAL SPECIFICATIONS PER 10 CFR 50.54(x)

At ________ hrs license condition _______________ was deviated from per 10 CFR 50.54(x). This condition requires _________________________________. *Discussion as to why the condition was not met, affect on the plant and when compliance was/will be restored.*

The NRC Resident Inspector was notified.

- Safety Limit Violation (TS 6.7.1)

VIOLATION OF SAFETY LIMIT

At _________hrs Technical Specification Safety Limit _______ was violated when ____. Immediate corrective actions were _________________________________. *Discussion as to the affect on the plant, additional planned actions, and any compensatory actions taken to assure safety.*

The NRC Resident Inspector was notified.

I.B. RADIOLOGICAL EVENTS

- Radioactive Shipments

Note: The time and date of the spent fuel shipment is safeguards information. Date and time of discovery is not safeguards but care should be taken not to link the event to arrival of the cask.

Example: SURFACE CONTAMINATION ABOVE LIMITS

Smearable contamination on a radioactive materials package, a used fuel shipping cask transported by rail, exceeded the limits of 10 CFR 71.47. There is no evidence of personnel contamination or spread of contamination beyond the rail car. There is no indication of increased exposure to the public as a result of this event.

The NRC Resident Inspector was notified.

- Loss or Theft of Licensed Material/Radiological Sabotage

(This example is for Loss only. Theft or Sabotage is reported using SEC-NGGC-2147.)

LOCATION OF __________ IS UNKNOWN

How was loss of SNM discovered and what efforts to relocate are underway? What assurance is there that the lost SNM is under the control of a licensee (vs. the public) and that no personnel have been overexposed?

The NRC Resident Inspector was notified. Region II (*name)* and the State of North Carolina have also been notified.

- Exposure to Individuals or Releases

PERSONNEL OVEREXPOSURE

A worker received (*internal/external*) contamination resulting in an estimated total effective dose equivalent (TEDE) of _____ Rem (________mSv). The individual had been *doing what, where. How detected? What location on the body. What decontamination was performed to what result?*

What immediate corrective actions are being taken?

The NRC Resident Inspector was notified

- Accidental Criticality in the Fuel Handling Building

No example. This event is extremely rare.

I.C. SECURITY EVENTS

- Security Events Reported per SEC-NGGC-2147.

Security and Safeguards events are prepared by the Security Organization per SEC-NGGC-2147.

- International Atomic Energy Agency (IAEA) Representative

No example. This event is extremely rare.

I.D. FITNESS FOR DUTY

- FFD - NRC Employee

No example. This event is extremely rare.

SAMPLE WORKSHEET

NRC FORM 361 U.S. NUCLEAR REGULATORY COMMISSION OPERATIONS CENTER
(12-2000)

REACTOR PLANT EVENT NOTIFICATION WORKSHEET EN

NRC OPERATION TELEPHONE NUMBER: PRIMARY -- 301-816-5100 or 800-532-3469*, BACKUPS -- [1st] 301-951-0550 or 800-449-3694*, [2nd] 301-415-0550 and [3rd] 301-415-0553 *Licensees who maintain their own ETS are provided these telephone numbers.

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #
15:43 EDT	Harris Nuclear Plant	1	John Caves	(919) 362-3636

EVENT TIME & ZONE	EVENT DATE	POWER/MODE BEFORE	POWER/MODE AFTER
14:33 EDT	09/18/2003	100% Power, Mode 1	100% Power, Mode 1

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

DESCRIPTION

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

As of 2:33 PM, EDT, more than 20% of the offsite emergency sirens were inoperable for greater than one hour due to loss of power caused by Hurricane Isabel. Currently 27 of 81 sirens are out of service. The State of North Carolina and all four counties within the 10-mile emergency planning zone were notified and are in stand-by to implement mobile route alerting if needed. At this time, Harris cannot estimate the time of siren recovery. This requires an 8-hour non-emergency notification per 10CFR 50.72(b)(3)(xiii) due to the loss of a significant portion of the offsite notification system. The NRC Senior Resident Inspector was informed.

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT	X		
STATE(s)	X		
LOCAL	X		
OTHER GOV AGENCIES	X		
MEDIA/PRESS RELEASE		X	

ANYTHING UNUSUAL OR NOT UNDERSTOOD? ☐ YES (EXPLAIN ABOVE) X NO

DID ALL SYSTEMS FUNCTION AS REQUIRED? X YES ☐ NO

MODE OF OPERATION UNTIL CORRECTED: 1	ESTIMATE FOR RESTART DATE:	ADDITIONAL INFO ON BACK ☐YES X NO

Attachment 7
Sheet 1 of 2

NRC FORM 361 COMMISSION (12-2000) U.S. NUCLEAR REGULATORY OPERATIONS CENTER

REACTOR PLANT EVENT NOTIFICATION WORKSHEET EN

NRC OPERATION TELEPHONE NUMBER: PRIMARY -- 301-816-5100 or 800-532-3469*, BACKUPS -- [1st] 301-951-0550 or 800-449-3694*, [2nd] 301-415-0550 and [3rd] 301-415-0553 *Licensees who maintain their own ETS are provided these telephone numbers.

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #
	Harris Nuclear Plant	1		919 -

EVENT TIME & ZONE	EVENT DATE	POWER/MODE BEFORE	POWER/MODE AFTER

	EVENT CLASSIFICATIONS			1-Hr. Non-Emergency 10 CFR 50.72(b)(1)				
	GENERAL EMERGENCY	GEN/AAEC			TS Deviation	ADEV		
	SITE AREA EMERGENCY	SIT/AAEC		4-Hr. Non-Emergency 10 CFR 50.72(b)(2)				
	ALERT	ALE/AAEC		(i)	TS Required S/D	ASHU		
	UNUSUAL EVENT	UNU/AAEC		(iv)(A)	ECCS Discharge to RCS	ACCS		
	50.72 NON-EMERGENCY (see next columns)			(iv)(B)	RPS Actuation (scram)	ARPS		
	PHYSICAL SECURITY (73.71)	DDDD		(xi)	Offsite Notification	APRE		
	MATERIAL/EXPOSURE	B???		8-Hr. Non-Emergency 10CFR 50.72(b)(3)				
	FITNESS FOR DUTY	HFIT		(ii)(A)	Degraded Condition	ADEG		
	OTHER UNSPECIFIED REQMT. (see last column)			(ii)(B)	Unanalyzed Condition	AUNA		
	INFORMATION ONLY	NNF		(iv)(A)	Specified System Actuation	AESF		

	(v)(A)	Safe S/D Capability	AINA
	(v)(B)	RHR Capability	AINB
	(v)(C)	Control of Rad Release	AINC
	(v)(D)	Accident Mitigation	AIND
	(xii)	Offsite Medical	AMED
	(xiii)	Loss Comm/Asmt/Resp	ACOM
60-Day Optional 10 CFR 50.73(a)(1)			
		Invalid Specified System Actuation	AINV
Other Unspecified Requirement (identify)			
			NONR
			NONR

DESCRIPTION

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

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT			
STATE(s)			
LOCAL			
OTHER GOV AGENCIES			
MEDIA/PRESS RELEASE			

ANYTHING UNUSUAL OR NOT UNDERSTOOD? ☐ YES (EXPLAIN ABOVE) ☐ NO

DID ALL SYSTEMS FUNCTION AS REQUIRED? ☐ YES ☐ NO

MODE OF OPERATION UNTIL CORRECTED:	ESTIMATE FOR RESTART DATE:		ADDITIONAL INFO ON BACK ☐ YES ☐ NO

Attachment 7
Sheet 2 of 2

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

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

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

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

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

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

EVENT DESCRIPTION (continued from front)

NRC HEADQUARTERS
DUTY OFFICER CONTACTED:________________ ___/___/___ ____:___ AM/PM
NAME DATE TIME

Reportability Evaluation (REW) Worksheet

Notification Requirements Per 10CFR50.73 Reports NCR-______
(Ref. NUREG-1022)

A Licensee Event Report (LER) is generally required for any event of the type described in this Attachment within 60 days after the discovery of the event. HNP shall report any applicable event 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.

I. Description of the Event

II. For each Reportability Evaluation, perform the following:

- Consult with or perform a pre-job briefing with Licensing.
- Using the details of the event, determine if the reporting category applies to this event.
- **Mark** the appropriate block Yes or No.
- If uncertain, gather more information to make the determination. Consult Licensing as needed.
- Reference NUREG-1022 as needed since it contains many examples that may aid in the determination.
- Complete the Reportable Evaluation Section to justify the conclusion that was reached based on known facts.

Situation meets this condition?	**Reportable Event**
Yes ☐ **No** ☐	**Plant Shutdown Required by Technical Specifications?**
The completion of any nuclear plant shutdown required by the HNP Technical Specifications.	
Yes ☐ **No** ☐	**Operation or Condition Prohibited by Technical Specifications?**
Any operation or condition which was prohibited by the HNP 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.	
Yes ☐ **No** ☐	**Deviation from Technical Specifications?**
Any deviation from the HNP Technical Specifications authorized pursuant to section 50.54(x).	

Situation meets this condition?	Reportable Event
Yes ☐ No ☐	**System Actuation?**

Any event or condition that resulted in a manual or automatic actuation of any of the systems listed 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;**

(1) Occurred while the system was properly removed from service;

or

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

The systems to which the requirements above apply are:

a. Reactor protection system (RPS) including reactor trip.
b. General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).
c. Emergency core cooling systems (ECCS) including: high-head and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.
d. Auxiliary or Feedwater system.
e. Containment heat removal and depressurization systems, including containment spray and fan cooler systems.
f. Emergency ac electrical power systems, including emergency diesel generators (EDGs).
g. Emergency service water systems.

Yes ☐ No ☐	**Common Cause Inoperability of Independent Trains or Channels?**

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.

Situation meets this condition?	Reportable Event
Yes ☐ No ☐	**Event or Condition that Could Have Prevented Fulfillment of a Safety Function?**

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.

Events covered 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 if redundant equipment in the same system was operable and available to perform the required safety function.**

Yes ☐ No ☐	**Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems?**

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.

Events covered above may include cases of procedural error, equipment failure, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacy. However, HNP is **not** required to report an event above if the event results from:

a. A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or
b. Normal and expected wear or degradation.

Yes ☐ No ☐	**Degraded or Unanalyzed Condition?**

Any event or condition that resulted in:

a. The condition of the nuclear power plant including its principal safety barriers, being seriously degraded; or
b. The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

Situation meets this condition?	Reportable Event
Yes ☐ No ☐	**External Threat or Hampering?**
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 plant.	
Yes ☐ No ☐	**Internal Threat or Hampering?**
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.	
Yes ☐ No ☐	**Radioactive Release?**
Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in appendix B to 10CFR20, table 2, column 1. 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 10CFR20 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.	

III. Reportable Evaluation

IV. Conclusion

Based on the information above, this event:
(check one)

☐ Does **NOT** meet the reportability requirements of 10 CFR 50.73.

☐ **IS** reportable to the NRC per the requirements of 10.CFR 50.73.

Revision Summary
(PRR-609830)

General

This revision adds a reference for CR 580945. This change is an editorial correction per PRO NGGC-0204.

Description of Changes

Page	Section	Change Description
5	2.0	Added new reference # 41 for CR 580945