1. 000000000007EK3.01 2

Unit 1 Initial Conditions:
- 15% Power.
- An automatic Reactor Trip occurs.

Current conditions:
- Reactor is verified tripped.
- Main Turbine DID NOT automatically trip.
- Operators attempted to manually trip the Main Turbine from the control room.
- All Turbine Stop and Governor valves remain OPEN.
- Generator output breakers are CLOSED.
- Turbine speed is NOT DECREASING.
- Generator motoring is NOT initiated.
- It is now 45 seconds after the Reactor Trip.

Based on the current conditions, which one of the following is (1) the NEXT action required by 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," and (2) the reason for this action?

A. (1) Close MSTVs.
(2) Prevent an uncontrolled cooldown of the Reactor Coolant System (RCS).

B. (1) Close MSTVs.
(2) Prevent a Loss of Heat Sink condition from occurring.

C. (1) Open generator output breakers AND place Excitation control switch in OFF.
(2) Prevent motoring the Main Generator.

D. (1) Open generator output breakers AND place Excitation control switch in OFF.
(2) Actuate an alternate Turbine trip signal.

<u>K/A</u>

Reactor Trip
007EK3.01 Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip.
(CFR 41.5 / 41.10 / 45.6 / 45.13) (RO – 4.0)

<u>K/A Match Analysis</u>

This question matches the K/A statement by requiring the applicants to correctly identify the next required action in step 2 (immediate operator actions) of 1-E-0, given an operationally valid scenario, as well as identify the correct reason for this action. The reason for the next action is given in the Westinghouse EOP Background document/detailed step description; however, it is NOT SRO-only because it is dealing with immediate operator actions which are required to be known from memory by all licensed operators.

Designated as memory because steps were taken right from E-0.

<u>Answer Choice Analysis</u>

A. CORRECT. In accordance with 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," step [2] b) RNO, which states "IF Turbine will NOT Trip, THEN close MSTVs." Therefore, the correct NEXT ACTION is to close the MSTVs. The reason for this step is given in the Westinghouse EOP Background document for E-0 as follows: "The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require."

B. INCORRECT. In accordance with 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," step [2] b) RNO, which states "IF Turbine will NOT Trip, THEN close MSTVs." Therefore, the correct NEXT ACTION is to close the MSTVs and part (1) of this distractor is correct. However, part (2) is incorrect; the correct reason for this step is given in the Westinghouse EOP Background document for E-0 as follows: "The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require." Part (2) is plausible because if the turbine remains on line, it will continue to draw steam and lower steam generator levels, which may result in a loss of heat sink.

C. INCORRECT. Part (1) is incorrect, but plausible, because it is another action contained in step [2] of 1-E-0 to be taken after 30 seconds have passed. In accordance with 1-E-0 step [2] d) RNO, which states "IF Generator Output Breakers do NOT open within 30 seconds, THEN manually open output breakers AND place the EXCITATION control switch in OFF." Part (2) of distractor C. is plausible because it is one of the reasons for taking the action in part (1).

D. INCORRECT. Part (1) is incorrect as detailed in the analysis for distractor C. Part (2) of distractor D. is plausible because tripping the main generator will cause a turbine trip signal, and the goal of step [2] of 1-E-0 is to trip the turbine.

<u>Supporting References</u>

1. Surry Procedure 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," rev. 61.

2. Westinghouse E-0 Background Document HP-Rev. 2, dtd 04/30/2005. Especially section 4.1 "Detailed Description of Steps, Notes, and Cautions."

3. This question is modified from Braidwood 1 007EK1.03 from 04/01/1996 to apply to the Surry Power Station and to enhance plausibility.

4. This question is also modified from Beaver Valley 1 007EK1.03 from 04/28/1997 to apply to the Surry Power Station and to enhance plausibility.

References Provided to Applicant

None.

Approved

Answer: A

2. 000000000008AK2.02 2

Initial conditions:

- Unit 1 is at 100% power.
- 1-RC-LT-461 (PRZR LEVEL PROTECT CH 3) selected for “Upper Control Channel.”
- 1-RC-LT-460 (PRZR LEVEL PROTECT CH 2) selected for “Lower Control Channel.”

Current conditions:

- 1C-B8 “PZR LO PRESS” is lit
- 1C-C8 “PRZR HI LVL HTRS ON” is lit
- PRZR Master pressure controller **output** is 0%.
- PRZR level controller **output** is 0%.
- Pressure trends are observed on the following indicators:
  - 1-RC-PI-455 (PRZR PRESS PROTECT CH 1) = 2245 psig and dropping.
  - 1-RC-PI-456 (PRZR PRESS PROTECT CH 2) = 2243 psig and dropping.
  - 1-RC-PI-457 (PRZR PRESS PROTECT CH 3) = 2135 psig and dropping.
- Level trends are observed on the following indicators:
  - 1-RC-LT-459 (PRZR LEVEL PROTECT CH 1) = 48% and dropping.
  - 1-RC-LT-460 (PRZR LEVEL PROTECT CH 2) = 47% and dropping.
  - 1-RC-LT-461 (PRZR LEVEL PROTECT CH 3) = 68% and rising.

Based on the current conditions, which one of the following explains the above event?

A. PRZR Master pressure controller output has failed low.

B. A leak has developed at the instrument tap to the reference leg for level transmitter 1-RC-LT-461.

C. PRZR level controller output has failed low.

D. A leak has developed at the instrument tap to the variable leg for level transmitter 1-RC-LT-461.

K/A
Pressurizer Vapor Space Accident
Knowledge of the interrelations between the Pressurizer Vapor Space Accident and **Sensors and detectors**.
(CFR: 41.7/45.7) (RO – 2.7)

K/A Match Analysis

The RO applicant is required to recognize how a leak from the Pressurizer vapor space (i.e., reference leg instrument tap) will effect level and pressure indications and control on the pressurizer.

Answer Choice Analysis

A. INCORRECT. *Plausible because an output failure would cause the controller output to indicate zero and actual pressure would be trending up due to heaters coming on. However, the failure would not explain the contradictory trends in pressurizer level on the available pressurizer level indications.*

B. CORRECT. With a leak at the instrument tap to the reference leg of RC-LT-461, water in the reference leg will boil off causing indicated water level to increase and pressure in the reference leg to drop. In addition, pressure transmitters PT-457, PT-445 (input to pressure recorder PR-1441), and PT-444 (input to Pressurizer Master Pressure Controller) all tap off the same reference leg piping that supplies RC-LT-461 so indicated pressure would decrease rapidly on these pressure transmitters. PT-456 and PT-457 would also decrease. As the heaters come on in response to low pressure sensed by the Pressurizer Master Pressure Controller and an increase in sensed level of more than 5% from program level (53.7%).

C. INCORRECT. *Plausible because an output failure would cause the controller output to indicate zero and actual level would be trending down due to less charging flow*. *However, the failure would not explain the trends on pressurizer pressure or the zero output on the Pressurizer Master Controller.*

D. INCORRECT. *Plausible if the applicant believed a leak on the variable leg caused indicated water level to increase*.

Supporting References

1. Surry lesson plan ND-93.3-LP-7, "Pressurizer Level Control System," rev. 10, Obj D, pp. 5-6.

2. Surry lesson plan ND-93.3-LP-5, "Pressurizer Pressure Control System," rev. 14,

Obj D, pp. 6-7.

References Provided to Applicant

None.
Approved

Answer: B
3. 000000000009EK1.01 2

Unit 1 plant conditions:
- SBLOCA has occurred
- All charging pumps are inoperable
- RCS hot legs are voided
- FR-C.1 RESPONSE TO INADEQUATE CORE COOLING in progress
- RCPs are off
- Total AFW flow = 100 gpm
- The SRO directs the intact SGs to be depressurized to 200 psig

Based on the above conditions, which one of the following: (1) states what limits (if any) are placed on SG depressurization and (2) the reason for this action?

A. (1) Limited such that RCS cooldown rate does not exceed 100 ºF/Hr
(2) Maintain the SG as a heat sink via reflux boiling

B. (1) Limited such that RCS cooldown rate does not exceed 100 ºF/Hr
(2) To establish RCP support conditions

C. (1) There is no limit on how fast you can depressurize the SGs
(2) Maintain the SG as a heat sink via reflux boiling

D. (1) There is no limit on how fast you can depressurize the SGs
(2) To establish RCP support conditions

K/A
Small Break LOCA. **Knowledge of the operational implications of the following concepts as they apply to the small break LOCA:** Natural circulation and cooling, including reflux boiling.

K/A Match Analysis
Requires applicant to know operational method to depressurize SGs in order to establish reflux boiling during a SBLOCA.

Answer Choice Analysis

A. Incorrect: FR-C.1 Step 23 directs SGs to be depressurized at the maximum rate. 1st part is plausible because when not in loss of cooling scenarios, RCS cooldown is limited to prevent PTS concerns. 2nd part is correct.

B. Incorrect: Step 23 directs SGs to be depressurized at the maximum rate. 1st part is plausible because when not in loss of cooling scenarios, RCS cooldown is limited to prevent PTS concerns. 2nd part is plausible because a heat sink needs to be established if forced cooling is to be established.

C. Correct: FR-C.1 Step 23 directs SGs to be depressurized at the maximum rate. Depressurizing the SGs allows more feedwater flow into the SGs. This will condense some of the steam on the primary side of the U-Tubes which will flow back down the hot leg into the core (Reflux Boiling).

D. Incorrect: 1st part is correct. 2nd part is plausible because a heat sink needs to be established if forced cooling is to be established.

Supporting References
ND-95.3-LP-38 Obj: B
FR-C.1 RESPONSE TO INADEQUATE CORE COOLING Step 23

References Provided to Applicant
none

- Added charging pumps inoperable to improve validididty
- Change to RCP support conditions to clarify intent of part 2
-

Answer: C

4. 000000000011EG2.4.6 2

Unit 1 Initial Conditions:
- Unit 1 experienced a design-basis Large Break Loss of Coolant Accident (LBLOCA) coincident with a loss of offsite power.

Current Conditions:
- You are arriving at the plant to assume the Operator at the Controls (OATC) position.
- It is now ten (10) hours since the LOCA occurred.
- Containment Pressure has returned to sub-atmospheric conditions.

Based upon the current conditions, (1) what is the method of core cooling in use and (2) which spray system is in operation?

A. (1) Recirculation flow from the Containment sump through the SI system with discharge into the cold legs.

(2) The ISRS pumps are operated as required to maintain containment pressure.

B. (1) Recirculation flow from the Containment sump through the SI system with discharge into the cold legs.
(2) The OSRS pumps are operated as required to maintain containment pressure.

C. (1) Recirculation flow from the Containment sump through the SI system with discharge into the hot legs.
(2) The ISRS pumps are operated as required to maintain containment pressure.

D. (1) Recirculation flow from the Containment sump through the SI system with discharge into the hot legs.
(2) The OSRS pumps are operated as required to maintain containment pressure.

K/A

Large Break LOCA
011EG2.4.6 Large Break LOCA: Knowledge of EOP mitigation strategies.
(CFR 41.10 / 43.5 / 45.13) (RO – 3.7)

K/A Match Analysis

This question matches the K/A statement by requiring the applicant to correctly determine the proper recirculation alignment (EOP mitigation strategy) that would be in use 10 hours after a large-break LOCA occurred, including the ISRS alignment. The applicant needs to recall that a transfer to hot leg recirculation takes place 8 hours after a LBLOCA, and needs to be completed before 9 hours have elapsed. The Surry alignment closes all discharge valves into the cold legs and aligns all recirculation flow to the hot legs. 1-E-1 Directs the operator to operate ISRS pumps with SW aligned to repsective heat exchangers to maintain containment pressure less than 13 psia.

Answer Choice Analysis

A. INCORRECT. The first part of this distractor is plausible because it is the method of core cooling from the swapover from injection flow to recirculation flow until 8 hours post-LOCA; it should be the most familiar method of recirculation core cooling to the applicant. This choice could be selected if the applicant forgets about the switch to hot leg recirculation, or if the applicant does not remember the correct time frame for swapping to hot leg recirculation. The second part of this distractor is correct; 1-E-1 Step 13, will direct the operator to operate ISRS as necessary to maintain containment pressure less than 13 psia.

B. INCORRECT. The first part of this distractor is plausible because it is the method of core cooling from the swapover from injection flow to recirculation flow until 8 hours post-LOCA; it should be the most familiar method of recirculation core cooling to the applicant. This choice could be selected if the applicant forgets about the switch to hot leg recirculation, or if the applicant does not remember the correct time frame for swapping to hot leg recirculation. The second part of this distractor is incorrect; it is plausible as during a LBLOCA the OSRS is needed to reduce containment pressure to less that 13 psia. Additionally, the plausibility is further enhanced by the fact that with a HI HI CLS signal in, only the OSRS pumps can be operated from the MCR.

C. CORRECT. The transfer to hot leg recirculation is required to occur 8 hours post-LOCA, and be completed after 9 hours have elapsed. Therefore, the operator arriving at the plant to relieve the watch 10 hours post-LBLOCA should expect the plant to be on recirculation to the hot legs ONLY. Because no other conditions are given in the stem, the applicant must consider that all systems have functioned as they were designed. TThe second part of this distractor is correct; 1-E-1 Step 13, will direct the operator to operate ISRS as necessary to maintain containment pressure less than 13 psia.

D. INCORRECT. The transfer to hot leg recirculation is required to occur 8 hours post-LOCA, and be completed after 9 hours have elapsed. Therefore, the operator arriving at the plant to relieve the watch 10 hours post-LBLOCA should expect the plant to be on recirculation to the hot legs ONLY. Because no other conditions are given in the stem, the applicant must consider that all systems have functioned as they were designed. he second part of this distractor is incorrect; it is plausible as during a LBLOCA the OSRS is needed to reduce containment pressure to less that 13 psia. Additionally, the plausibility is further enhanced by the fact that with a HI HI CLS signal in, only the OSRS pumps can be operated from the MCR.

<u>Supporting References</u>

1. Surry Lesson Plan ND-95.2-LP-7, "Loss of Reactor Coolant Accident," rev. 10 dtd 02/13/08. Reflux boiling is described on p. 28.

2. Surry Procedure 1-E-1, "LOSS OF REACTOR OR SECONDAY COOLANT," rev. 31. Note before step 28 states that hot leg recirc must be established by 9 hours after the event.

3. Surry Lesson Plan ND-95.3-LP-7, "E-1, Loss of Reactor or Secondary Coolant," rev 16 dtd 07.28/08.

4. Surry Procedure 1-ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION," rev. 17.

5. Surry Lesson Plan ND-95.3-LP-10, "ES-1.3, TRANSFER TO COLD LEG RECIRCULATION," rev 11 dtd 07/16/07.

6. Surry Procedure 1-ES-1.4, "TRANSFER TO HOT LEG RECIRCULATION," rev. 5.

7. Surry Lesson Plan ND-95.3-LP-11, "ES-1.4, TRANSFER TO HOT LEG RECIRCULATION," rev. 10 dtd 07/17/07.

References Provided to Applicant

None.

- 1-E-1 directs the operator to 'Operate ISRS pump(s) with SW aligned to respective hx(s) to maintain CTMT pressure less than 13 psia.' Added CTMT pressure to clarify event conditions.
- 2 right answers on original question
- The LBLOCA coincident with a loss of offsite power, due to the design of #3 EDG, both trains of ISRS will have power and in accordance with the procedure could be operated (statement: Operate ISRS as necessary)

Answer: C

5. 000000000022AA1.07 2

Initial conditions:
- Unit 1 is at 100% power.
- Letdown was removed from service to repair a leak.
- Excess letdown is in service.
- Pressurizer (PRZR) is at program level.

Current conditions:
- 1-AP-8.00, Loss of Normal Charging Flow, has been initiated.
- VCT level indicates off-scale low.
- Charging flow and discharge pressure is erratic.
- All three charging pumps are running.
- Charging pump amps are erratic on all pumps.
- System valve positions are normal.

Based on the current conditions, which one of the following describes...

(1) how excess letdown flow will be affected
and
(2) whether a reactor trip is required in accordance with 1-AP-8.00?

A. (1) Excess letdown will continue until manually isolated.
(2) Reactor trip is NOT required.

B. (1) Excess letdown will be automatically isolated.

(2) Reactor trip is NOT required.

C. (1) Excess letdown will be automatically isolated.
(2) Reactor trip is required.

D. (1) Excess letdown will continue until manually isolated.
(2) Reactor trip is required.

K/A

Loss of Reactor Coolant Makeup
Ability to operate and/or monitor *Excess Letdown isolation containment valve switches and indicators* as it applies to **Loss of Reactor Coolant Makeup**.
(CFR: 41.7/45.5/45.6) (RO – 2.8)

K/A Match Analysis

The RO applicant is required to determine how excess letdown flow path is affected by a loss of Reactor makeup.

Answer Choice Analysis

A. INCORRECT. *Plausible because the first half of the response is correct – the excess letdown system will continue to operate. Also, if the applicant believes the conditions indicate something other than gas binding (e.g., low suction pressure, low flow, etc.) then other actions are taken to re-establish charging flow.*

B. INCORRECT. *Plausible because normal letdown will automatically isolate and the AP does direct a reactor trip if gas binding exists on the charging system*.

C. INCORRECT. *Plausible because normal letdown will automatically isolate. Also, if the applicant believes the conditions indicate something other than gas binding (e.g., low suction pressure, low flow, etc.) then other actions are taken to re-establish charging flow.*

D. CORRECT. Excess letdown flow will continue until manually isolated and the AP does direct a reactor trip if gas binding exists on the charging system.

Supporting References

1. Surry lesson plan ND-88.3-LP-3, "Seal Injection," rev. 10, Obj E, p. 15.

2. 1-AP-8.00, “Loss of Normal Charging”, rev. 12, pg. 2

3. This question is modified from Sequoyah 2010-301 NRC exam (Q #5). Added "gas binding" conditions to the stem and modified the second half of the distractors to focus on whether a reactor trip is required versus identifying a condition that requires a reactor trip.

References Provided to Applicant

None.

The step in AP-8.00 reads that "Running CHG pump suspected of gas binding and ...." The question as written did not give a reason to suspect gas binding. The VCT low level provides this reason. All charging pumps running can be expected with all pumps gas bound. Remove stem statement about motors operating normally because amps fluctuating tends to oppose this statement.

Answer: D

6. 000000000025AA1.12 2

Unit 1 Initial Sequence of events at 180 hours after shutdown:
- Mid loop operation was entered after a shutdown from 100%
- The running RHR pump suction was vortexing
- 1-AP-27.00, LOSS OF DECAY HEAT REMOVAL CAPABILITY, was entered
- The crew reduced RHR flow and raised RCS level, which stopped the vortexing

Current Conditions:
- RCS temperature is 145^{o}F and increasing slowly
- The RHR controls are in their normal configuration

Which one of the following correctly describes: (1) actions that will stabilize temperature, and (2) the response of RHR cold leg return flow after the actions have been taken?

A. (1) Adjust the potentiometer to manually open 1-RH-HCV-1758 (RHR HXS FLOW).
(2) RHR cold leg return flow indication will rise.

B. (1) Adjust the potentiometer to manually open 1-RH-HCV-1758 (RHR HXS FLOW).
(2) RHR cold leg return flow indication will remain approximately stable.

C. (1) Raise the setpoint on the controller to allow 1-RH-HCV-1758 (RHR HXS FLOW) to automatically open.
(2) RHR cold leg return flow indication will rise.

D. (1) Raise the setpoint on the controller to allow 1-RH-HCV-1758 (RHR HXS FLOW) to automatically open.
(2) RHR cold leg return flow indication will remain approximately stable.

K/A

Loss of RHR

Ability to operate and/or monitor the following as they apply to the loss of RHR System: RCS temperature indicators.

K/A Match Analysis

Applicant must know how RHR flow will respond to the manipulations to control temperature.

Answer Choice Analysis

A. INCORRECT. Incorrect. Plausible because, opening valves typically results in flow going up; however, in this system, there is another valve downstream of two parallel flow paths the maintains flow constant.

B. CORRECT. According to the LP, HCV-1758 has no automatic controls. Therefore, this valve must be manually controlled. With controls in their normal configuration, FCV-1605 would be in auto controlling at the chosen setpt. Therefore, no change in RHR flow should occur.

C. INCORRECT. 1-HCV-1758 does not have any auto functions; however, FCV-1605 does, which makes this choice plausible. Second part plausible for same reason as listed in "A".

D. INCORRECT. See above.

Supporting References

ND-88.2-LP-1, Residual Heat Removal System Description, Rev 8.
1-AP-27.00, LOSS OF DECAY HEAT REMOVAL CAPABILITY, Rev. 20.

References Provided to Applicant

None.

Answer: B
7. 000000000026AA2.03 2

Initial conditions:
- Both units are at 100% power

Current unit one conditions:

- Annunciator 1C-A1 - RCP 1A CC RETURN LO FLOW is LIT
- Annunciator 1C-B1 - RCP 1B CC RETURN LO FLOW is LIT
- Annunciator 1C-C1 - RCP 1C CC RETURN LO FLOW is LIT
- Annunciator 1C-H2 - CTMT CC OUT HDR 1B LO FLOW is LIT
- Annunciator VSP-D7 - CC SURGE TK Hi-Lo LVL is LIT

- Annunciator 1K-E7 - CC PPs DISCH HDR LO PRESS is LIT
- Annunciator RM-N6 - CC/SW HX C ALERT/FAILURE is LIT

- The turbine building operator has aligned make-up to the CC surge tank
- CC surge tank level is decreasing at 7%/minute with make-up aligned
- The operating CC pump amps are oscillating erratically from 0 to 60 amps.
- CC surge tank level is currently 2%

Based on the current conditions, which one of the following components would isolate the leakage?

Close the CC inlet and outlet valves to the …

A. RCP supply and return headers

B. "C" component cooling water heat exchanger

C. aligned RHR heat exchanger

D. seal water heat exchanger

K/A

Loss of Component Cooling Water
Ability to determine and/or interpret *the valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition* as it applies to **Loss of Component Cooling Water**.
(CFR: 43.5/45.13) (RO – 2.6)

K/A Match Analysis

The RO applicant must evaluate current plant configuration, CC flow indications and CC surge tank level trends to determine which component has experienced a CC leak and identify the component that should be isolated.

Answer Choice Analysis

A. INCORRECT. *Plausible because with the RCP return low flow alarms LIT the candidate may believe that isolation of the RCP pumps would isolate the CC Leak.*

B. CORRECT. Combination of the CC RM alarm, low discharge pressure, and low flow allow the candidate to arrive at this answer.

C. INCORRECT. *Plausible, because with the containment low flow alarm LIT, the* *candidate may believe that isolation of the RHR HX may be required.*

D. INCORRECT. *Plausible because the combination of the containment low flow and RCP low flow alarms, the candidate may believe this combination would lend itself to a seal water heat exchanger leak. Additionally, if the seal water heat exchanger were to leak into the CC system a CC RM alarm would be expected (but not a CC/SW RM alarm).*

<u>Supporting References</u>

1. Surry lesson plan ND-88-5-LP-1, "Component Cooling Systems," rev. 21, Objective B, slides 19, 25, 26, 30, 43 and 45.
2. 1K-E7, CC PPS DISCH HDR LO PRESS, Rev. 1
3. Dwg # 11448-FM-072A, Sh 1 of 7, Rev. 28
4. Dwg # 11448-FM-072B, Sh 1 of 3, Rev. 29
5. Dwg # 11448-FM-072B, Sh 2 of 3, Rev. 29*
6. Dwg # 11448-FM-072C, Sh 1 of 5, Rev. 39
7. Dwg # 11448-FM-072C, Sh 2 of 5, Rev. 34*
8. Dwg # 11448-FM-072D, Sh 2 of 5, Rev. 33

* Need to include copies of these drawings in the documentation file.

<u>References Provided to Applicant</u>

None.

Answer: B

8. 000000000027AK2.03 2

Unit 1 Initial Conditions:
- Time = 1500.
- 100% Power.
- Pressurizer (PZR) Pressure = 2235 psig.
- All PZR Pressure control components are in their normal 100% power alignments.

Current Conditions:
- Time = 1502.
- 1-RC-PC-1444J (PZR Pressure Master Controller) **<u>output</u>** fails to a constant value of 70%.
- No operator actions have been taken.

Based upon the current conditions, what is the expected configuration of the PZR Pressure control system?

A. Proportional Heaters OFF, Both Spray Valves FULL OPEN, PZR PORV 1-RC-PCV-1455C OPEN.

B. Proportional Heaters OFF, Both Spray Valves FULL OPEN, PZR PORV 1-RC-PCV-1455C CLOSED.

C. Proportional Heaters 50% OUTPUT, Both Spray Valves CLOSED, PZR PORV 1-RC-PCV-1455C CLOSED.

D. Proportional Heaters 100% OUTPUT, Both Spray Valves CLOSED, PZR PORV 1-RC-PCV-1455C CLOSED.

K/A

Pressurizer Pressure Control System Malfunction
027AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners.
(CFR 41.7 / 45.7) (RO – 2.6)

K/A Match Analysis

This question matches the K/A statement by requiring the applicant to correctly recall the relationship between master PZR pressure controller output and the expected response of the PZR pressure control system. The question is testing fundamental knowledge in that an applicant could arrive at the correct answer by straightforward recall of controller setpoints (such as the table on handout page 5.4 of the included Surry lesson plan). If the applicant does not recall this relationship directly, he or she could also arrive at the correct answer by calculation of pressure setpoints vs. percent controller output. To do so, the controller span (i.e. 0% to 100% is equivalent to 200 psig, -60 psig to +140 psig) must be remembered:

$$\frac{x}{200} = \frac{70\%}{100\%}$$

$$\Rightarrow x = 140 \Rightarrow x \sim NOP + 80\ psig$$

$$\therefore x = (2235 + 80) = 2315\ psig\ .$$

Answer Choice Analysis

A. INCORRECT. This distractor is plausible if an applicant mis-remembers the controller output setpoints from the PZR Pressure master controller. This distractor would correspond to a controller output of 80% or greater.

B. CORRECT. At a constant output of 70% a short time after PZR Pressure is at NOP (2235 psig), it is expected that the proportional (modulating) heaters would be OFF, the sprays OPEN, and the PORV CLOSED.

C. INCORRECT. This distractor is plausible if an applicant mis-remembers the controller output setpoints from the PZR Pressure master controller. This distractor would correspond to a controller output of 30%--i.e, the applicant mentally failed the controller in the wrong Pressure direction.

D. CORRECT. This distractor is plausible if an applicant mis-remembers the controller output setpoints from the PZR Pressure master controller. This distractor would correspond to a controller output of 22.5% or lower--i.e. the applicant mentally failed the controller in the wrong Pressure direction.

Supporting References

1. Surry Lesson Plan ND-93.3-LP-5, “Pressurizer Pressure Control,” rev. 13 dtd 12/14/08. Specifically p. 7 and the summary of setpoints on handout 5.4.

References Provided to Applicant

None.

Answer: B

9. 000000000029EA2.08 2

Unit 1 initial conditions:
- Reactor power = 12%
- Main Turbine surveillance testing is in progress
- CERPI: Bank A = 228 Steps
Bank B = 228 Steps
Bank C = 228 Steps
Bank D = 140 Steps

Current plant conditions:
- Main Turbine trip occurs during testing
- Following the turbine trip CERPI indicates as follows:
Bank A = 228 Steps
Bank B = 228 Steps
Bank C = 228 Steps
Bank D = 140 Steps except control rod H-14 which = 0 Steps

Based on the above conditions, which one of the following states: (1) what the Control

Rod Group Step Counter for Group D should read and (2) the correct procedure the crew is required to initiate to address the event?

A. (1) 140 Steps
   (2) 1-E-0 REACTOR TRIP OR SAFETY INJECTION

B. (1) 140 Steps
   (2) 0-AP-1.00 CONTROL ROD SYSTEM MALFUNCTION

C. (1) 0 Steps
   (2) 1-E-0 REACTOR TRIP OR SAFETY INJECTION

D. (1) 0 Steps
   (2) 0-AP-1.00 CONTROL ROD SYSTEM MALFUNCTION

K/A

Ability to determine or interpret the following as they apply to an ATWS. Rod bank step counters and RPI.

K/A Match Analysis

Requires applicant to know how equipment is effected by a fire based on the location of the fire.

Answer Choice Analysis

A. Correct: Control Rod Group Counter will remain at its prior position. With reactor power = 12%, the reactor should have tripped (ATWS) so entry into E-0 is required.

B. Incorrect: 1st part is correct. The reactor should have tripped so E-0 is the correct procedure to enter. 2nd part is plausible because if power was < 10%, it would be correct.

C. Incorrect: Step counter does not track actual CR position. Plausible because it does show the demand signal for that CR. 2nd part is correct.

D. Incorrect: 1Step counter does not track actual CR position. Plausible because it does show the demand signal for that CR. The reactor should have tripped so E-0 is the correct procedure to enter. 2nd part is plausible because if power was < 10%, it would be correct.

Supporting References

ND-93.3-LP3 Rod Control System Obj: K
ND-93.3-LP4 CERPI System Obj: E

References Provided to Applicant

none

Request Licensee to provide valid CR number and operation of step counters.

Answer: A

10. 000000000038EK1.02 2

Unit 1 Initial Conditions:

- 100% Power.
- Reactor Coolant System (RCS) Pressure is 2235 psig and stable.
- Steam Generator (S/G) 'A' Pressure is 790 psig and stable.
- A seismic event causes a complete shear of one tube in S/G 'A' coincident with a loss of offsite power.
- Initial tube rupture flow rate from the RCS to S/G 'A' is 800 gpm.

Current conditions:

- Operators correctly perform the step to isolate ruptured S/G(s) in 1-E-3, STEAM GENERATOR TUBE RUPTURE, and have properly adjusted the 'A' S/G PORV controller setpoint.
- S/G 'A' pressure is 20 psig lower than the 'A' S/G PORV controller setpoint.
- S/G 'A' PORV is closed.
- RCS pressure is 1650 psig.

Based on the current conditions, which one of the following is: (1) the current tube rupture flow rate from the RCS to S/G 'A,' AND (2) if S/G 'A' PORV were to open at setpoint, the required action in accordance with 1-E-3 is to __________ ?

A. (1) 350 gpm
(2) place 'A' S/G PORV controller in MANUAL and close the PORV.

B. (1) 350 gpm
(2) verify 'A' S/G PORV closed when 'A' S/G pressure is less than setpoint.

C. (1) 530 gpm
(2) place 'A' S/G PORV controller in MANUAL and close the PORV.

D. (1) 530 gpm
(2) verify 'A' S/G PORV closed when 'A' S/G pressure is less than setpoint.

<u>K/A</u>

Steam Generator Tube Rupture (SGTR)

038EK1.02 Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop.

(CFR 41.8 / 41.10 / 45.3) (RO - 3.2)

K/A Match Analysis

This question matches the K/A statement by requiring the applicants to correctly calculate a current SGTR leak rate, given initial and current conditions that correspond to the design basis SGTR accident as described in the Surry FSAR. To arrive at the correct answer, the applicant must recall/understand the basic relationship between volumetric flow rates and differential pressure. The applicant must also recall an important S/G PORV pressure setpoint from E-3 in order to obtain the correct flow rate value. The second part of the question requires the applicant to correctly recall the required actions for an open ruptured S/G PORV, in accordance with 1-E-3.

Answer Choice Analysis

A. INCORRECT. This distractor is based upon an incorrect belief that volumetric flowrate is directly proportional to pressure drop (instead of the square root of differential pressure) multiplied by a density correction factor. This mistake is plausible because it is approximately the method used to determine mass flow rates. 1-E-3, STEAM GENERATOR TUBE RUPTURE, requires that the ruptured S/G PORV controller setpoint be adjusted to 1035 psig as part of the step to isolate the ruptured S/G (step 3). Because the question stem states that the ruptured S/G pressure is 20 psig less than the setpoint, the applicant will use 1015 psig in the calculation, which proceeds as follows:

$$\dot{V} \propto \rho_{RCS}\Delta P \ (incorrect\ \ formula)$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{\left(47.03482\,{}^{lbm}\!/\!_{ft^3}\right)\left(2235\,psig - 790\,psig\right)}{\left(46.63061\,{}^{lbm}\!/\!_{ft^3}\right)\left(1650\,psig - 1015\,psig\right)}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{67965.315}{29610.43735}$$

$$\Rightarrow \dot{V}_{current} = 349\ gpm\ (incorrect)$$

The second part of this distractor is also incorrect, because step 3.b. RNO states "WHEN ruptured SG pressure less than 1035 psig, THEN verify SG PORV closed." The second part distractor is plausible, because it is the required action if the SG PORV does not close when pressure goes less than 1035 psig. It is further plausible because closing the PORV will terminate a radioactive release, and also the closed PORV will create a higher backpressure, which will lead to a lower SGTR flow rate from the RCS to the ruptured SG (K/A match).

B. INCORRECT. This distractor is based upon an incorrect belief that volumetric flowrate is directly proportional to pressure drop (instead of the square root of differential pressure) multiplied by a density correction factor. This mistake is plausible because it is approximately the method used to determine mass flow rates. 1-E-3, STEAM GENERATOR TUBE RUPTURE, requires that the ruptured S/G PORV controller setpoint be adjusted to 1035 psig as part of the step to isolate the ruptured S/G (step 3). Because the question stem states that the ruptured S/G pressure is 20 psig less than the setpoint, the applicant will use 1015 psig in the calculation, which proceeds as follows:

$$\dot{V} \propto \rho_{RCS}\Delta P \ (incorrect \ \ formula)$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{\left(47.03482\,{}^{lbm}\!/\!{}_{ft^3}\right)(2235\,psig - 790\,psig)}{\left(46.63061\,{}^{lbm}\!/\!{}_{ft^3}\right)(1650\,psig - 1015\,psig)}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{67965.315}{29610.43735}$$

$$\Rightarrow \dot{V}_{current} = 349 \ gpm \ (incorrect)$$

The second part of this distractor is correct, because step 3.b. RNO states "WHEN ruptured SG pressure less than 1035 psig, THEN verify SG PORV closed."

C. INCORRECT. 1-E-3, STEAM GENERATOR TUBE RUPTURE, requires that the ruptured S/G PORV controller setpoint be adjusted to 1035 psig as part of the step to isolate the ruptured S/G (step 3). Because the question stem states that the ruptured S/G pressure is 20 psig less than the setpoint, the applicant will use 1015 psig in the calculation, which proceeds as follows:

$$\dot{V} \propto \sqrt{\Delta P}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{\sqrt{(2235\,psig - 790\,psig)}}{\sqrt{(1650\,psig - 1015\,psig)}}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{38.0131556}{25.1992063}$$

$$\Rightarrow \dot{V}_{current} = 530 \ gpm \ (correct \ answer) \ q.e.d.$$

The second part of this distractor is incorrect, because step 3.b. RNO states "WHEN ruptured SG pressure less than 1035 psig, THEN verify SG PORV closed." The second part distractor is plausible, because it is the required action if the SG PORV does not close when pressure goes less than 1035 psig. It is further plausible because closing the PORV will terminate a radioactive release, and also the closed PORV will create a higher backpressure, which will lead to a lower SGTR flow rate from the RCS to the ruptured SG (K/A match).

D. CORRECT. 1-E-3, STEAM GENERATOR TUBE RUPTURE, requires that the ruptured S/G PORV controller setpoint be adjusted to 1035 psig as part of the step to isolate the ruptured S/G (step 3). Because the question stem states that the ruptured S/G pressure is 20 psig less than the setpoint, the applicant will use 1015 psig in the calculation, which proceeds as follows:

$$\dot{V} \propto \sqrt{\Delta P}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{\sqrt{(2235\,psig - 790\,psig)}}{\sqrt{(1650\,psig - 1015\,psig)}}$$

$$\frac{800\,gpm}{\dot{V}_{current}} = \frac{38.0131556}{25.1992063}$$

$$\Rightarrow \dot{V}_{current} = 530\ gpm\ (correct\ answer)\ q.e.d.$$

The second part of this distractor is also correct, because step 3.b. RNO states “WHEN ruptured SG pressure less than 1035 psig, THEN verify SG PORV closed.”

<u>Supporting References</u>

1. Surry Power Station UFSAR rev. 41 (as of 09/20/09) section 14.3.1, “Steam Generator Tube Rupture.” On p. 14.3-4, under figure 14.3-6, “Break Flow,” it is stated that initial break flow is approximately 800 gpm, the value used in the question stem.

2. Surry Procedure 1-E-3, “STEAM GENERATOR TUBE RUPTURE,” rev. 38. Step 3 of 1-E-3, “ISOLATE RUPTURED SG(s):” requires the ruptured S/G PORV controller setpoint be adjusted to 1035 psig.

3. Surry Lesson Plan ND-89.1-LP-2 rev. 24, “Main Steam System.” Page 19 states that main steam pressure is approximately 800 psig at 100% power (close to the 790 psig used in the question stem).

4. Steam Tables. http://www.steamtablesonline.com. Printout provided (density values for initial and current conditions).

<u>References Provided to Applicant</u>

Steam Tables.

Answer: D

11. 000000000054AK1.02 2

Unit 1 Initial Conditions:

- A small break LOCA occured following a reactor trip due to a loss of main feedwater.
- Subsequently, the AFW system degraded and 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, was entered.

Unit 1 Current Conditions:

- RCS Bleed and Feed has been established
- Hot leg RCS temperatures are 475^{o}F and stable
- CETC temperatures are 475^{o}F and stable
- SG WR Levels are: A = 4%, B = 6%, C = 5%
- Containment pressure is 25 psia and lowering
- AFW flow capability has been restored

Based on the current conditions, which one of the following states the 1-FR-H.1 requirements for establishing AFW?

A. Feed SGs at a minimum total flow of 350 gpm.

B. Feed SGs at a minimum total flow of 450 gpm.

C. Feed only one SG at no more than 60 gpm.

D. Feed only one SG at no more than 100 gpm.

<u>K/A</u>

Loss of Main Feedwater
Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): Effects of feedwater introduction on dry S/G.

<u>K/A Match Analysis</u>

The question requires the applicant to know the operational implications (I.E. AFW feed requirements) of feeding a dry steam generator.

<u>Answer Choice Analysis</u>

A. INCORRECT. Plausible because the only difference between this and the correct answer is that this choice uses the non-adverse value for AFW flow requirements.

B. CORRECT. Adverse containment numbers are in effect because ctmt pressure is above 20 psia. Step 23 of H.1 directs that attempts to establish AFW flow should still be made iaw Step 2. Step 2 states that total flow should be at least 450 gpm.

C. INCORRECT. See D below. The same justification exists for C, except that the non-adverse value is used.

D. INCORRECT. The Continuous Action Page criteria are only to be used for a hot/dry SG, which is indicated by WR SG levels < 7%[22%] AND RCS hot leg temperatures greater than 550 F. The stem of this question has temperatures at less than 550 F. If the applicant applied the criteria for a hot dry SG, then this would be a correct answer.

<u>Supporting References</u>

1. 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Rev. 28.

<u>References Provided to Applicant</u>

None.

Answer: B

12. 000000000057AG2.2.37 3

Unit 1 Initial Conditions:

- Power = 100%.
- Time = 0255. A large electrical fire at the site caused the loss of Vital Buses 1-III, 1-IIIA, 1-IV and 1-IVA.

Current conditions:

- Time = 0300. Operators initiate a manual Reactor Trip.
- Time = 0301. One Reactor Coolant Pump (RCP) is secured.
- Time = 0303. Operators note the following:
  - Loop ‘A’ Tave = 541 °F
  - Loop ‘B’ Tave = 551 °F
  - Loop ‘C’ Tave = Failed Low
- Time = 0304. Safety Injection is NOT actuated.

Based upon the current conditions, which one of the following describes (1) whether an automatic safety injection should have occurred by Time = 0304, AND (2) how many

RCPs have lost Component Cooling (CC) to the lube oil coolers?

A. (1) An automatic Safety Injection should have actuated.
   (2) One RCP has lost CC to the lube oil coolers.

B. (1) An automatic Safety Injection should NOT have actuated.
   (2) More than one RCP has lost CC to the lube oil coolers.

C. (1) An automatic Safety Injection should have actuated.
   (2) More than one RCP has lost CC to the lube oil coolers.

D. (1) An automatic Safety Injection should NOT have actuated.
   (2) One RCP has lost CC to the lube oil coolers.

<u>K/A</u>

057 Loss of Vital AC Instrument Bus
057AG2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR 41.7 / 43.5 / 45.12) (RO – 3.6)

<u>K/A Match Analysis</u>

This question matches the K/A statement by requiring the RO applicant to determine whether a safety injection should have occurred, and the status of vital CC cooling to RCP motors, given an operationally valid scenario where multiple Vital AC Instrument Buses have lost power.

<u>Answer Choice Analysis</u>

A. INCORRECT. Part (1) of this distractor is correct. The Surry lesson plan for Vital and Semi-Vital Bus Distribution states the following for a loss of Vital Bus III: “It should be noted that Safety Injection is imminent if A or B Loop Tave drops below 543 °F. This is due to the High Steam Flow/Low Tave Channel III bistables being in a tripped condition due to loss of power.” A NOTE in 1-AP-10.03, “LOSS OF VITAL BUS III,” before step 7 states essentially the same thing as the lesson plan. Therefore, based on the conditions given in the stem, SI should have automatically actuated. Part (2) of this distractor is incorrect. The Surry lesson plan again states: “A loss of Vital Bus II, III, or IV requires the operator to manually trip the reactor and secure the appropriate RCP due to a loss of CC to the lube oil coolers for the RCP.” Because both Vital Bus III and IV have been lost, CC to the lube oil coolers has been lost to the ‘A’ and ‘C’ RCPs.

B. INCORRECT. Part (1) is incorrect, part (2) correct. See analysis of ‘A’ above.

C. CORRECT. Parts (1) and (2) correct. See analysis of ‘A’ above.

D. INCORRECT. Part (1) incorrect, part (2) incorrect. Plausible if an applicant wrongly remembers the power supply/Vital Bus arrangement to the High Steam Flow/Low Tave SI bistables; also plausible because if Vital Bus I is lost, CC to the lube oil coolers to a RCP is not lost (if an applicant mis-remembers the correct Vital Bus lineup).

Supporting References

1. Surry Procedure 1-AP-10.03, “LOSS OF VITAL BUS III,” rev. 16. Especially NOTE before step 7.

2. Surry Lesson Plan ND-90.3-LP-5, “VITAL AND SEMI-VITAL BUS DISTRIBUTION,” rev. 17.

References Provided to Applicant

None.

Answer: C
13. 000000000058G2.2.36 2

Initial Conditions:
- Unit One is at hot shutdown preparing for a reactor start-up.

Current conditions:
- While performing the monthly battery test on the 1A station battery, a tool is dropped on the battery lead terminals and the short results in the battery leads separating from the battery. Due to the fault current experienced, the battery chargers on both UPS trip off-line, resulting in a loss of the "A" DC bus.

Which one of the following describes (1) the impact of this failure on #1 EDG, and (2) the status of LCO 3.16, Emergency Power System?

A. (1) #1 EDG starts and automatically assumes 1H bus.
(2) Actions of LCO 3.16 are required.

B. (1) #1 EDG starts and automatically assumes 1H bus.
(2) Actions of LCO 3.16 are NOT required.

C. (1) #1 EDG starts but cannot be loaded from the control room.
(2) Actions of LCO 3.16 are NOT required.

D. (1) #1 EDG starts but cannot be loaded from the control room.
(2) Actions of LCO 3.16 are required.

K/A

Loss of DC Power
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (as related to **Loss of DC Power**)
(CFR: 41.10/43.2/45.13) (RO – 3.1)

K/A Match Analysis

The RO applicant must determine how an loss of the 1A 125 VDC system due to maintenance effects the LCO status.

Answer Choice Analysis

A. INCORRECT. Part one is INCORRECT but plausible if the candidate believes that the EDG battery bank supplies power to the EDG output breaker. Part two is correct iaw T.S. 3.16

B. INCORRECT. Part one is INCORRECT but plausible if the candidate believes that the EDG battery bank supplies power to the EDG output breaker. Part two is plausible as 3.16 has two statements for applicability (one that is redundant with the other) and if the candidate believes that 3.16 has a ***critical*** applicability, them this distracter would seem correct.

C. INCORRECT. Part one is CORRECT Part two is plausible as 3.16 has two statements for applicability (one that is redundant with the other) and if the candidate believes that 3.16 has a ***critical*** applicability, them this distracter would seem correct.

D. CORRECT.

Supporting References

1. Surry lesson plan ND-90.3-LP-6, "125VDC DISTRIBUTION,"

2. 1-AP-10.06

3. Technical Specifications, LCO 3.16

References Provided to Applicant

None.

Answer: D

14. 000000000062AK3.02 1

Which ONE of the following is correct regarding automatic alignments of the station service water (SW) system in response to a Hi-Hi CLS and the reason for that feature?

A. If the Hi-Hi CLS occurred with a station blackout, SW to the CCHXs will automatically isolate to conserve canal inventory. These valves can then be re-opened immediately to allow heat sink restoration to the CCHXs.

B. If the Hi-HI CLS occurred with a loss of intake canal level, SW to two RSHXs will automatically throttle close to 25% to conserve intake canal level. These valves can then be re-opened immediately to allow for the additional heat removal capability of the system.

C. If the Hi-Hi CLS occurred with a station blackout, SW to the CCHXs will automatically isolate to conserve canal inventory. These valves can then be re-opened after 5 minutes to allow heat sink restoration to the CCHXs.

D. If the Hi-HI CLS occurred with a loss of intake canal level, SW to two RSHXs will automatically throttle close to 25% to conserve intake canal level. These valves can then be re-opened after 5 minutes to allow for the additional heat removal capability of the system.

K/A

Loss of Nuclear Service Water

Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS

<u>K/A Match Analysis</u>

This question requires the candidate to know the automatic actions that occur in the service water system in response to a ESF actuation

<u>Answer Choice Analysis</u>

a- Incorrect but plausible if the, forgets about the 5 minute lockout associated with the SW MOVs b- Incorrect but plausible, as the CW MOVs due throttle to 25% closed under these conditions to conserve intake canal level. Additionally, the design of the plant allows for 2 RSHXs to be isolated.

c- Correct.

d- Incorrect but plausible, as the CW MOVs due throttle to 25% closed under these conditions to conserve intake canal level. Additionally, the design of the plant allows for 2 RSHXs to be isolated.

Supporting References

LP-ND-89.5-2, Service Water System, Rev. 16.

References Provided to Applicant

None.

Answer: C
15. 000000000077AK3.02 2

Current conditions:

- The station has been notified that the real-time contingency analysis program is unavailable.
- Unit 1 and Unit 2 crews have entered 0-AP-10.18, Response to Grid Instability.
- While performing step #7 (Check system conditions), the system operator reports the following voltage readings:
    - 230 KV bus = 236 KV
    - 500 KV bus = 494 KV
- Which one of the following describes the concern with these readings in accordance with 0-AP-10.18
  and
- The maximum (minimum) MVAR limit allowed to compensate for this condition and the basis for this limit?

A. (1) The maximum voltage limit has been exceeded on the 230 KV bus.
   (2) 400 MVARS in to limit stator winding temperature.

B. (1) The maximum voltage limit has been exceeded on the 230 KV bus.
   (2) 500 MVARS in to limit stator winding temperature.

C. (1) The minimum voltage limit has been reached on the 500 KV bus.
   (2) 500 MVARS out to limit rotor winding temperature.

D. (1) The minimum voltage limit has been reached on the 500 KV bus.
   (2) 400 MVARS out to limit rotor winding temperature.

K/A

Generator Voltage and Electric Grid Disturbances
Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: Actions contained in abnormal procedure for voltage and grid disturbances.
(CFR: 41.4, 41.5, 41.7, 41.10/45.8) (RO – 3.6)

K/A Match Analysis

The RO applicant is required to recognize off-normal voltage conditions on the offsite sources addressed in the abnormal procedure for grid disturbances and understand the reason for the voltage limits contained in the abnormal procedure.

Answer Choice Analysis

A. INCORRECT. *Plausible because the voltage is significantly higher than the schedule deviation (±6 KV) allowed by 1-OP-26.5, 230 KV Switchyard Voltage, and at 236kv it is the first step that the OP requires actions. The second part is correct for MVARS out iaw the generator capability curve (DRP-003).*

B. INCORRECT. *Plausible because the voltage is significantly higher than the schedule deviation (±6 KV) allowed by 1-OP-26.5, 230 KV Switchyard Voltage, and at 236kv it is the first step that the OP requires actions. The second part is incorrect for MVARS out iaw the generator capability curve (DRP-003), but plausible if the candidate recalls the max limit for MVARS (for lower MWe output).*

C. INCORRECT. *Plausible because the minimum voltage limit has been reached on the 500 KV bus. The second part is incorrect for MVARS out iaw the generator capability curve (DRP-003), but plausible if the candidate recalls the max limit for MVARS (for lower MWe output).*

D. CORRECT. The minimum voltage limit on the 500 KV bus is 505 KV. The second part is also correct IAW the generator capability curve.

Supporting References

1. Surry lesson plan ND-95.1-LP-8, "Loss of Offsite Power," rev. 12, Objective C, pg. 18.

2. 0-AP-10.18, Response to Grid Instability, Rev. 19, pgs. 5 & 7

3. Generator capability curve - DRP-003.

References Provided to Applicant

None.

Answer: D
16. 000000000WE04EA1.3 1

Unit 1 Initial Conditions:

- An automatic Reactor Trip and Safety Injection occurred from 100% power.

Current Conditions:

- Operators have just transitioned to 1-ECA-1.2, “LOCA OUTSIDE CONTAINMENT.”

Which ONE of the following parameters is used to determine if the break has been isolated, in accordance with 1-ECA-1.2?

A. RVLIS level

B. RCS subcooling

C. RCS pressure

D. PRZR level

K/A

LOCA Outside Containment
W/E04EA1.3 Ability to operate and/or monitor the following as they apply to the (LOCA Outside Containment): Desired operating results during abnormal and emergency situations.
(CFR 41.7 / 45.5 / 45.6) (RO – 3.8)

K/A Match Analysis

This question matches the K/A statement by requiring the applicant to correctly determine the proper system parameter used in ECA-1.2 to determine if the LOCA has been isolated by the actions of ECA-1.2. The question is a straightforward memory-level examination of exactly what the K/A requires us to test: “desired operating results.” Specifically, in ECA-1.2 the ultimate desired operating result (or outcome) is RCS pressure increasing due to isolating the break.

Answer Choice Analysis

A. INCORRECT. This distractor is plausible because RVLIS level should begin to INCREASE once the LOCA is isolated and SI flow continues. However, RVLIS level is not the parameter required to be monitored by ECA-1.2, and A is therefore an incorrect distractor. Previous versions of this question used Pressurizer level for this distractor; however, the exam author changed this to RVLIS level due to his belief that it was a more plausible choice.

B. INCORRECT. This distractor is plausible because once the LOCA is isolated, SI flow and an increase in RCS pressure would also cause Subcooling to increase; however, this is not the parameter required by ECA-1.2 and is therefore incorrect.

C. CORRECT. The high-level action statement of step 2 of 1-ECA-1.2 is to “TRY TO IDENTIFY AND ISOLATE BREAK,” and step b) states “Check RCS pressure – INCREASING.” If the answer is “yes,” the operator is eventually directed to transition to E-1 based on the success of the isolation efforts.

D. INCORRECT. This distractor is plausible because once the LOCA is isolated, Pressurizer level would be expected to increase due to SI injection flow.

Supporting References

1. Surry Procedure 1-ECA-1.2, “LOCA OUTSIDE CONTAINMENT,” rev. 6.

2. Surry Lesson Plan ND-95.3-LP-21, “ECA-1.2, LOCA OUTSIDE CONTAINMENT,” rev. 7 dtd 11/13/96.

3. This question is modified from Turkey Point W/E04EA2.2 from 2002-301 and Farley WE04EA2.2 from 2006-301 to apply to the Surry Power Station and to enhance plausibility.

References Provided to Applicant

None.

Answer: C
17. 000000000WE05EK2.2 3

Unit 1 current conditions:

- Unit trip occured due to a loss of main feedwater pumps.
- 1-FR-H.1, Response to Loss of Secondary Heat Sink, has been initiated.
- All Main Feedwater (MFW) and Auxiliary Feedwater (AFW) pumps are unavailable.
- Unit 2 AFW cross-tie valve will not open.
- Charging and letdown are in service.
- Pressurizer pressure is 2330 psig.
- Core Exit Thermocouple Temperature is 270°F.
- SG Wide range level indications are as follows:
    - SG 1A = 10% SG 1B = 9% SG 1C = 10%
- The crew has reached step 5 of 1-FR-H.1 (Depressurize RCS to1950 psig).

Based on the current conditions, which one of the following is consistent with the mitigation strategy :

A. Initiate RCS Feed and Bleed immediately due to inadequate steam generator inventory.

B. Initiate RCS Feed and Bleed immediately due to excessive RCS pressure.

C. Continue with RCS depressurization to 1950 psig using one pressurizer PORV.

D. Continue with RCS depressurization to 1950 psig using auxiliary spray.

<u>K/A</u>

Loss of Secondary Heat Sink

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and the relations between proper operation of these systems to operation of the facility.

(CFR: 41.7/45.7) (RO – 3.9)

<u>K/A Match Analysis</u>

The RO applicant is required to distinguish whether conditions require initiating RCS feed and bleed or continuing performance of the normal sequence of actions contained in FR-H.1, Loss of Secondary Heat Sink.

.

<u>Answer Choice Analysis</u>

A. *INCORRECT. Plausible because all the SGs are less than 12% level which is used* *as a level limit for other steps in the procedure. However, the 12% limit is based on the narrow range level gauges not the wide range gauges, which establish a level limit of 7% wide range as the point where RCS Feed and Bleed are to be initiated (See caution statement before step 2 and Foldout page of FR-H.1).*

B. INCORRECT. *Plausible because RCS pressure is abnormally high and FR-H.1 directs RCS Feed and Bleed due to high RCS pressure. However, the pressure limit is 2335 psig(See caution statement before step 2 and Foldout page of FR-H.1).*

C. INCORRECT. *Plausible because continuing with RCS depressurization to 1950 is the correct action. However, the procedure step directs the use of the pressurizer PORV only if letdown is not in service.*

D. CORRECT. Continuing RCS depressurization to 1950 psig is the correct action using the auxiliary spray to minimize thermal shock on the pressurizer.

Supporting References
1. Surry lesson plan ND-95.3-LP-41, "FR-H.1," rev. 12, Objective C, pgs. 17-18, 21.

2. 1-FR-H.1, Response to Loss of Secondary Heat Sink, rev. 8, step 5 and foldout page

References Provided to Applicant
none

Answer: D
18. 000000000WE12EA2.2 2

Unit 1 initial plant conditions:
- Reactor power = 100%
- Main Steam Line break occurs in unit one safeguards
- Main Steam Trip Valves failed to close
- Transition to 1-ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS has occurred

Current plant conditions:
- SG A, B and C NR Level = 6%
- RCS cooldown rate is 124°F/hr.

Based on the above conditions, which ONE of the following states: (1) The Minimum AFW flow requirements per steam generator and (2) The maximum cooldown rate allowed IAW 1-ECA-2.1?

A. (1) 60 gpm
(2) 100 ^{o}F/hr

B. (1) 60 gpm
(2) 50 ^{o}F/hr

C. (1) 120 gpm
(2) 100 ^{o}F/hr

D. (1) 120 gpm
(2) 50 ^{o}F/hr

K/A

Steam Line Rupture – Excessive Heat Transfer. Ability to determine and interpret the following as they apply to the (Uncontrolled Depressurization of all Steam Generators): Adherence to appropriate procedures and operation within limitations in the facility's license and amendments.

K/A Match Analysis

Question requires knowledge of facility procedures including cooldown rate limitations during a faulted SG.

Answer Choice Analysis

A. CORRECT. Per 1-ECA-2.1, if SG NR level < 12% [18%], maintain a minimum of 60 gpm [100gpm]. Cooldown rate is limited to 100°F/hr.

B. INCORRECT. 1st part is correct. 2nd part is plausible because normal admin limit of 50°F/hr cooldown rate.

C. INCORRECT. 1st part is incorrect but plausible if candidate applies normal AFW flow requirements by EOP network (and entry into FR-H.1). Second part is correct.

D. INCORRECT. 1st part is incorrect but plausible if candidate applies normal AFW flow requirements by EOP network (and entry into FR-H.1). 2nd part is plausible because normal admin limit of 50°F/hr cooldown rate.

Supporting References

1-ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS
ND-95.3-LP-22 Obj: A

References Provided to Applicant

None.

Answer: A
19.   000000003AA1.02 2

Unit 1 Initial Conditions:

- 70% Power.
- 0-AP-1.01, “CONTROL ROD MISALIGNMENT,” is being implemented.
- A dropped Control Bank ‘C’ rod has just been re-aligned with its group, but the lift coil disconnect switches have NOT been returned to normal.

Current Conditions:

- While attempting to operate the INTERNAL ALARM RESET pushbutton, the operator inadvertently operates the STARTUP RESET pushbutton.

Based on the current conditions, which one of the following describes the effect of operating the incorrect reset?

A. All of the Control Bank 'C' rods drop into the core.   No other rods drop into the core.

B. All rods, including all Control Bank and Shutdown Bank rods, drop into the core.

C. All rods remain in their current position.   There is no effect on the control rod group step counters.

D. All rods remain in their current position.   The control rod group step counters indicates all rods are fully inserted.

<u>K/A</u>

Dropped Control Rod
003AA1.02   Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: Controls and components necessary to recover rod.
(CFR 41.7 / 45.5 / 45.6) (RO – 3.6)

<u>K/A Match Analysis</u>

This question matches the K/A statement by testing the applicant’s ability to monitor the expected plant response during a dropped rod recovery, when an important component is mis-operated.   The applicant needs to apply understanding of the rod control system, specifically the function of the rod control startup pushbutton, to arrive at the correct answer.

Answer Choice Analysis

A. INCORRECT. This distractor is plausible if the applicant misunderstands the rod control system alignment during rod recovery, and believes that the lift coil disconnect switch alignment will cause only the Control Bank ‘C’ rods to drop into the core. Plausibility for this distractor is enhanced by emphasizing in the stem that the lift coil disconnect switches are still in the ‘abnormal’ alignment required to recover the dropped rod.

B. INCORRECT. This distractor is plausible if the applicant does not understand the operation of the rod control startup pushbutton, and confuses the rod control startup pushbutton with the Rx trip breaker reset pushbutton, and believes operation of this button cycles the reactor trip breakers (which would cause all rods to drop).

C. INCORRECT. The first part of this distractor is correct; no rods will drop when the rod control startup pushbutton is pressed. However, all rod indications will reset to zero, and therefore there will be an effect on Rod Control System circuitry.

D. CORRECT. As per Surry lesson plan ND-93.3-LP-3, “Rod Control System,” the STARTUP RESET pushbutton resets the following to zero: P/A converter counters, Internal memory and alarm circuit, group step counters, slave cycler counters, master cycler counters, and bank overlap counters. Therefore, no rods would move, and all indications would reset to zero.

Supporting References

1. Surry Procedure 0-AP-1.01, “CONTROL ROD MISALIGNMENT,” rev. 20. especially step 27.

2. Surry Lesson Plan ND-95.3-LP-3 “Rod Control System,” rev. 19 dtd 09/01/09.

3. This question is modified from Harris 003AA1.02 to apply to the Surry Power Station and to enhance plausibility.

References Provided to Applicant

None.

Answer: D
20. 000000005AG2.4.46 3

With reactor power initially stable at 50%, the following alarms were received:

- 1C-B8 - PRZR LO PRESS
- 1G-B5 - COMPUTER PRINTOUT ROD CONT SYS
- 1G-E2 - RPI SYS TROUBLE
- 1G-H1 - NIS DROPPED ROD FLUX DECREASE >5% PER 2 SEC
- 1G-H2 - RPI ROD BOTTOM <20 STEPS

Which ONE of following states 1) whether the problem is a dropped rod or CERPI issue, and 2) an alarm that supports this conclusion (and eliminates the other)?

A. 1) The problem is a CERPI indication issue
   2) 1G-E2 - RPI SYS TROUBLE

B. 1) The problem is a CERPI indication issue
   2) 1G-H2 - RPI ROD BOTTOM <20 STEPS

C. 1) The problem is an actual dropped rod
   2) 1G-H1 - NIS DROPPED ROD FLUX DECREASE >5% PER 2 SEC

D. 1) The problem is an actual dropped rod
   2) 1G-H2 - RPI ROD BOTTOM <20 STEPS

<u>K/A</u>

Inoperable / Stuck Control Rod
Ability to verify that the alarms are consistent with the plant conditions.
(CFR: 41.10/43.5/45.3/45.12) (RO – 4.2)

<u>K/A Match Analysis</u>

The RO applicant is required to evaluate rod position information and determine that conditions would result first in 1G-B5.

<u>Answer Choice Analysis</u>

A. INCORRECT. Plausible because the candidate may believe that the alarm indicates a rod withdrawal error on the "D" control bank

B. INCORRECT. Plausible because candidate may believe that the alarm indicates a mis-step in the control bank during movement.

C. CORRECT.

D. INCORRECT. Plausible because this alarm is typically received during power operations for a dropped or mis-aligned control rod.

Supporting References

1. Surry lesson plan ND-93.3-LP-3, "Rod Control System," rev. 19, Obj F, pg. 60 - 61.

2. Annunciator Response Procedure, 1G-B5, COMPUTER PRINTOUT ROD CONT SYS, Rev. 14.

3. Annunciator Response Procedure, 1G-A6, ROD CONT SYS URGENT FAILURE, Rev. 1.

4. This question is modified from Harris 2008-301 Exam (Q #5). Modified conditions to be consistent with an urgent rod system failure rather than a dropped/mispositioned rod.

References Provided to Applicant

None.

Answer: C
21. 000000032AK3.02 2

A large break LOCA has occured

Plant conditions 10 minutes into the event:
- Containment pressure is 32 psia
- Pressurizer pressure - 3 psig
- Pressurizer level - 0%
- RVLIS full range - 46%
- Containment High Range radiation - $3.45x10^6$

Current plant conditions:
- Containment pressure is 16 psia
- Pressurizer pressure - 2 psig
- Pressurizer level - 0%
- RVLIS full range - 52%
- Containment High Range radiation - $9.45x10^4$

Which ONE of the following states the instrument to be used to monitor core reactivity and the basis for this decision?

A. Source range NIs, because containment conditions are no longer adverse.

B. Source range NIs, because RVLIS level has increased such that voiding is no longer occuring and SRNIs will indicate properly.

C. Gamma-Metrics because adverse containment conditions have been exceeded during the event.

D. Gamma-Metrics because RVLIS level was low enough during the event to indicate vessel voiding and cause the SRNIs to indicate erratically.

K/A

Loss of Source Range Nuclear Instrumentation
Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Guidance contained in EOP for loss of source-range nuclear instrumentation.

K/A Match Analysis

The applicant is required to know which instruments to monitor when SR instruments have been lost and the reason why that instrument is required to be used.

Answer Choice Analysis

A. INCORRECT - In accordance with FR-S.1, once adverse containment numbers have been exceeded SRNIs can no longer be utilized. This choice is plausible as all other adverse numbers can be reverted to normal numbers once adverse conditions have cleared.

B. INCORRECT - Plausible, if the candidate recalls FR-C.2 entry conditions indicate voiding has occured and that level above this entry condition would allow SRNIs to indicate properly, regardless of previous containment conditions.

C. CORRECT

D. INCORRECT - Gamma-Meterics would be correct, however, the reason is based on containment conditions not RVLIS level. However, plausibility is added as the initial RVLIS level is indication of voiding, which would cause the SRNI to indicate erratically.

Supporting References

1-E-1, Loss of Secondary or Reactor Coolant, Rev. 34.
1-ECA-0.0, Loss of All AC Power, Rev. 32.

References Provided to Applicant

None.

DISCUSS WITH THE LICENSEE IF ADV CTMT CAN BE EXITED IF EITHER PRESSURE OR RADIATION DROPS BACK BELOW THE SETPOINT FOR ADV NUMBERS.

Answer: C

22. 000000051AG2.4.35 2

Unit 1 Initial Conditions:

- 100% Power.
- The operations crew notes degrading condenser vacuum and enters 1-AP-14.00, "LOSS OF MAIN CONDENSER VACUUM."
- The crew is performing steps in Attachment 3, Low Air Ejector Flow Rate.

Current Conditions:

- The Turbine Building operator checks both condenser air ejector loop seal drain lines to condenser, and reports that one loop seal drain line is **very hot** to the touch, and the other is normal.

Based on the current conditions, which one of the following are the required actions, in accordance with 1-AP-14.00?

A. Isolate ONLY the **hot** loop seal drain line, verify the condenser hoggers are in service, and secure ONLY the set of air ejectors associated with the **hot** loop seal drain line.

B. Isolate ONLY the **hot** loop seal drain line, if Air Ejector flows are NOT normal leave the drain valve closed for approximately 5 minutes, then reopen the **hot** loop seal drain isolation valve.

C. Isolate **BOTH** loop seal drain lines, verify the condenser hoggers are in service, and secure ONLY the set of air ejectors associated with the **hot** loop seal drain line.

D. Isolate **BOTH** loop seal drain lines, if Air Ejector flows are NOT normal leave both drain valves closed for approximately 5 minutes, then reopen **both** loop seal drain isolation valves.

K/A

Loss of Condenser Vacuum
051AG2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
(CFR 41.10 / 43.5 / 45.13) (RO – 3.8)

K/A Match Analysis

This question matches the K/A statement by requiring the applicant to correctly apply knowledge of the AP for loss of condenser vacuum when directing local auxiliary operator tasks, given an operationally plausible situation. An auxiliary operator may not have the procedure in hand when performing local tasks during an emergency, and may rely on guidance provided from the control room.

Answer Choice Analysis

A. INCORRECT. First part of answer is correct regarding isolation of the HOT loop seal, but the direction to align the Hoggers and remove the HOT air ejector from service is INCORRECT.

B. CORRECT.

C. INCORRECT. First part of answer is incorrect but plausible as a recent procedure revision changed this from being correct to incorrect. Second part is incorrect (See 'A')

D. INCORRECT. First part is incorrect (see "C"), second part is correct in theory but not applied to both air ejectors.

Supporting References

1. Modified from question 051AG2.4.35 from the North Anna 2008-301 exam to support Surry references.

2. Surry Procedure 1-AP-14.00, "LOSS OF MAIN CONDENSER VACUUM," rev. 7. Especially Attachment 3 step 4.

References Provided to Applicant

None.

Answer: B
23. 000000059AA1.01 2

Current conditions on Unit 1:

- 1000 - A Large Break Loss of Coolant Accident has occurred.
- 1030 - 1B train of Recirc Spray is the only available train and started automatically.
- 1038 - Increasing trend is identified on 1-SW-RM-115 (RS/SW HX B RM)
- 1042 - Alarm 1-RM-A8, RS/SW HX B ALERT / FAILURE is received.
- 1048 - Alarm 1-RM-B8, 1-SW-RI-115 HIGH, alarm is received.
- 1050 - An increasing trend has been identified on 1-SW-RI-120 (DISCH TUNNEL)
- 1052 1-SW-RI-115 indicates all EEEEEs and the following lights are lit on the radiation monitor:
    - Warning
    - High
    - Range

Based on the current conditions, which one of the following describes the response of the radiation monitors and an acceptable action in accordance with alarm response procedures?

A. 1-SW-RM-115 has failed high. Request HP sampling per the Offsite Dose Calculation Manual.

B. Current radiation levels are above the range of 1-SW-RM-115. Request HP sampling per the Offsite Dose Calculation Manual.

C. 1-SW-RM-115 has failed high. Secure 1B Recirc Spray train.

D. Current radiation levels are above the range of 1-SW-RM-115. Secure 1B Recirc Spray train.

K/A

Accidental Liquid Radwaste Release
Ability to operate and/or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive – liquid monitor.
(CFR: 41.7/45.5/45.6) (RO – 3.5)

K/A Match Analysis

The RO applicant is required to monitor indications associated with the radiation monitors on the Recirculation Spray heat exchangers, recognize that elevated readings exist, determine that an accidental liquid release is occurring and apply the correct actions to address the release.

Answer Choice Analysis

NOTE: *The question focuses on whether the applicant can determine from the indications of the radiation monitor whether the instrument has failed high or is detecting radiation levels outside the meters range.*

A. INCORRECT. *Plausible because per the note prior to step 1 of 1-RM-A8 the digital ratemeter with all EEEEEs displayed indicates a failure of the digital ratemeter. However, all EEEEs with the warning, high and range alarm lights lit indicates that radiation levels are above the range of the meter. An upscale failure of the monitor would also cause the fail alarm light to light up. In addition, requesting HP sampling is the correct answer. Per the caution statement prior to step five of the ARP, the RS train should only be secured if a redundant train is available.*

B. CORRECT. All EEEEs with the warning, high, and range alarm lights lit indicates that radiation levels are above the range of the meter. In addition, requesting HP sampling is the correct answer. The RNO actions for a RS heat exchanger that cannot be removed from service (See step 6 of the ARP) is either monitor the Discharge Tunnel radiation monitor or have HP conduct sampling per the ODCM.

C. INCORRECT. *Plausible because per the note prior to step 1 of 1-RM-A8 the digital ratemeter with all EEEEEs displayed indicates a failure of the digital ratemeter. However, all EEEEs with the warning, high and range alarm lights lit indicates that radiation levels are above the range of the meter. The second half of the response is plausible because on indications of a leak on an RS heat exchanger (high radiation reading), the ARP directs securing the RS train. However, per the caution statement prior to step five of the ARP, the RS train should only be secured if a redundant train is available.*

D. INCORRECT. *Plausible because the first half of the response is correct. The second half of the response is plausible because on indications of a leak on an RS heat exchanger (high radiation reading), the ARP directs securing the RS train. However, per the caution statement prior to step five of the ARP, the RS train should only be secured if a redundant train is available.*

Supporting References

1. ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 10, Pgs. 5-7 and11-13

2. ND-91-LP-6, Recirculation Spray System, Rev. 13, pg. 12

3. 1-RM-A8, RS/SW HX B ALERT/FAILURE, Rev. 4

References Provided to Applicant

None.

Answer: B

24. 000000067AA2.17 2

Unit 1 initial conditions:

Fire is reported in the "B" 4160v station service bus
The SRO directs the bus to be de-energized

Based on the above conditions, which one of the following states (1) the load lost when de-energizing the “B” 4160v station service bus and (2) the type of fire suppression system in the 4160V Station Service Switchgear room?

A. (1) 1-BC-P-1A "A" BC Pump
(2) Low Pressure CO2

B. (1) 1-SD-P-1B "B" HP HTR Drain Pump (2) Low Pressure CO2

C. (1) 1-BC-P-1A "A" BC Pump (2) High Pressure CO2

D. (1) 1-SD-P-1B "B" HP HTR Drain Pump (2) High Pressure CO2

K/A

Plant Fire On-Site. **Ability to determine and interpret the following as they apply to the plant fire on Site:** Systems that may be affected by the fire.

K/A Match Analysis

Requires applicant to know how a fire on a specific switchgear will affect plant equipment.

Answer Choice Analysis

A Correct: The 1-BC-P-1A "A" BC is powered from the "B" 4160v station service bus. The fire suppression system used for the 4160V Station Service Switchgear rooms is Low pressure CO2.

B Incorrect: 1st part is incorrect because the 1-SD-P-1B "B" HP is powered from the “C” 4160v station service bus. 1st part is plausible because it is pump 1B “B” which by convention would be powered from the B switchgear. 2nd part is correct.

C Incorrect: 1st part is correct. 2nd part is incorrect because these switchgear rooms have LP CO2 systems. Plausible because they are CO2 systems.

D Incorrect: : 1st part is incorrect because the 1-SD-P-1B "B" HP is powered from the "C" 4160v station service bus. 1st part is plausible because it is pump 1B "B" which by convention would be powered from the B switchgear. 2nd part is incorrect because these switchgear rooms have LP CO2 systems. Plausible because they are CO2 systems.

Ref

ND-90.2-LP-2 Obj: A

References Provided to Applicant

None.

Answer: A
25. 000000068AK2.07 2

Unit 1 Initial Conditions:

- A large electrical fire occurred in the Main Control Room.
- Operators implemented 0-FCA-1.00, "LIMITING MCR FIRE."
- Both units' Reactor and Turbine are tripped.
- All Reactor Coolant Pumps (RCPs) are placed in Pull-to-Lock (PTL).
- During the confusion, the Unit 1 operator inadvertently placed ALL Unit 1 Charging Pumps in PTL.

Current Conditions:

- You were dispatched to the #1 EDG with 0-FCA-12.00, "EMERGENCY DIESEL GENERATOR OPERATION."
- Upon arrival at #1 EDG, you report that the #1 EDG is NOT running.
- The Senior Reactor Operator (SRO) informs you that the station has NOT lost normal offsite power.

Based on the current conditions, which one of the following is (1) the required action for loss of RCP seal injection, in accordance with 0-FCA-1.00, AND (2) the required action(s) for operation of the #1 EDG, in accordance with 0-FCA-12.00?

A. (1) Wait at least 30 minutes and then isolate RCP seals.
(2) Verify switch lineup at #1 EDG for normal automatic operation, but do NOT transfer #1 EDG control to the local panel.

B. (1) Wait at least 30 minutes and then isolate RCP seals.
(2) Transfer #1 EDG control to the local panel.

C. (1) RCP seals must be isolated and remain isolated until the RCS is cooled down to less than 200 °F.
(2) Transfer #1 EDG control to the local panel.

D. (1) RCP seals must be isolated and remain isolated until the RCS is cooled down to less than 200 °F.
(2) Verify switch lineup at #1 EDG for normal automatic operation, but do NOT transfer #1 EDG control to the local panel.

K/A

068 Control Room Evacuation
068AK2.07 Knowledge of the interrelations between the Control Room Evacuation and the following: ED/G.
(CFR 41.7 / 45.7) (RO – 3.3)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to demonstrate knowledge of the overall mitigating strategy for #1 EDG operation during a Control Room Evacuation casualty when offsite power is NOT lost to the station. In the other part of the question, the RO applicant will demonstrate knowledge of RCP seal isolation during a Control Room Evacuation casualty, given a plausible operationally valid situation (loss of seal injection flow coincident with Control Room Evacuation).

Answer Choice Analysis

A. INCORRECT. Part (1) is incorrect. A CAUTION statement before step 20 of 0-FCA-1.00, “LIMITING MCR FIRE,” (CHECK CHARGING PUMPS – AT LEAST ONE RUNNING ON EACH UNIT) clearly states: “If RCP Seal Injection has been lost, RCP seals must be isolated and remain isolated until the RCS is cooled down to less than 200 °F.” Part (1) is plausible because 30 minutes is a realistic time frame to RCP seal failure. Part (2) is also incorrect. Given the operational situation, the operator will use 0-FCA-12.00 to verify the EDG switch lineup, then transfer EDG control to the local panel, then verify offsite power available and breaker 15H8 closed, and then transition to step 34 to determine if #3 EDG needs to be ‘fast started.’ Part (2) is plausible because the question stem does not mention any effects on the #1 EDG automatic functions—so it is plausible that the EDG will auto-start and load as needed if offsite power was lost. However, the procedure 0-FCA-12.00 directs transferring control of the #1 EDG to local in every circumstance.

B. INCORRECT. Part (1) is incorrect, part (2) is correct. See analysis of ‘A.’ above.

C. CORRECT. Part (1) is correct, part (2) is correct. See analysis of ‘A.’ above.

D. INCORRECT. Part (1) is correct, part (2) is incorrect. See analysis of ‘A.’ above.

<u>Supporting References</u>

1. Surry Procedure 0-FCA-1.00, “LIMITING MCR FIRE,” rev. 42. Especially step 20 and its CAUTION statement.

2. Surry Procedure 0-FCA-12.00, “EMERGENCY DIESEL GENERATOR OPERATION,” rev. 14. Procedural flow path for operation of #1 EDG for a Control Room Evacuation <u>without</u> a loss of offsite power.

<u>References Provided to Applicant</u>

None.

Answer: C
26. 000000WE08EK1.1 1

Which one of the following correctly states the SI reduction criteria for 1-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION (1) for subcooling margin and (2) the RCS inventory with all RCPs running?

A. (1) > 30 F [85 F]
(2) > 82% Dynamic Range RVLIS

B. (1) > 30 F [85 F]
(2) > 22% [50%] PZRZ level

C. (1) > 80 F [135 F]
(2) > 82% Dynamic Range RVLIS

D. (1) > 80 F [135 F]
(2) > 22% [50%] PZRZ level

<u>K/A</u>

RCS Overcooling
Knowledge of operational implications of the following concepts as they apply to the (Pressurizer Thermal Shock): Components, capacity, and function of emergency systems.

K/A Match Analysis

To arrive at the correct answer, the applicant must know the SI reduction criteria from P.1. This is testing knowledge of components, systems, and their function as it relates to PTS.

Answer Choice Analysis

A. INCORRECT. The second part is correct. The first part is plausible because this is the criteria used in E-2.

B. INCORRECT. Both parts are plausible because this is the criteria used in E-2.

C. CORRECT. These criteria are spelled out in step 6 of P.1.

D. INCORRECT. The first part is correct. The second part is plausible because this is the criteria used in E-2.

Supporting References

1. 1-E-2, FAULTED STEAM GENERATOR ISOLATION, Rev. 17.
2. 1-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, Rev. 17.

References Provided to Applicant

None.

Answer: C

27. 000000WE10EK1.2 3

Current conditions:
- Unit 1 experienced a reactor trip and a loss of offsite power.
- 1-ES-0.4, Natural Circulation Cooldown with Steam Void in Rx Vessel (W/O RVLIS), is in progress.

During RCS depressurization with Pressurizer level above 90% per 1-ES-0.4, which ONE of the following states 1) the action required in response to this pressurizer level and 2) the basis for this action?

A. 1) Energize pressurizer heaters
   2) To prevent water relief out of the safety valves

B. 1) Commence RCS Drain

2) To prevent water relief out of the safety valves

C. 1) Energize pressurizer heaters
2) To promote reactor head cooling and collapse any void that may exist

D. 1) Commence RCS Drain
2) To promote reactor head cooling and collapse any void that may exist

K/A

Natural Circulation
Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Normal, abnormal, emergency operating procedures associated with (Natural Circulation with Steam Void in Vessel with/without RVLIS)
(CFR: 41.8/41.10/45.3) (RO – 3.5)

K/A Match Analysis

The RO applicant is required to know the operational reason for the action requiring energizing the pressurizer heaters when pressurizer level is greater than 90%.

Answer Choice Analysis

A. INCORRECT. The first part is correct. The second part is plausible as energizing heaters will increase pressurizer pressure and push water out of the pressurizer and will minimize the likelihood of water relief.

B. INCORRECT. The first part is incorrect, but plausible as the candidate may believe this pressurizer level is too high (i.e., greater than reactor trip setpoint) and that level reduction is required to maintain the steam bubble. The second part is plausible as reducing pressurizer level will minimize the likelihood of water relief.

C. CORRECT. Correct.

D. INCORRECT. The first part is incorrect, but plausible as the candidate may believe this pressurizer level is too high (i.e., greater than reactor trip setpoint) and that level reduction is required to maintain the steam bubble. The second part is correct.

Supporting References

1. Surry lesson plan ND-95.3-LP-55, "Emergency Response Guidelines ES-0.4, NC Cooldown with Steam Void in Rx Vessel (without RVLIS)," rev. 10, Obj B, pg. 19.

2. 1- ES-0.4, “NC Cooldown with Steam Void in Rx Vessel (without RVLIS), rev. 12, pg. 5.

3. This question is modified from Sequoyah 2009-301 (Q #26). Modified to a two-part question stem requiring the reason PZR heater energizing during RCS cooldown and depressurization and the parameters used to determine when use of PZR heaters is complete.

References Provided to Applicant

None.

Answer: C

28. 000003K4.07 2

Unit 1 Initial Conditions:

- 100% Power.

Current Conditions:

- No. 1 Seal Leakoff on 1-RC-P-1B, the ‘B’ Reactor Coolant Pump (RCP), unexpectedly rises to a value of 6.5 gpm and stabilizes.
- Operators enter 1-AP-9.00, “RCP ABNORMAL CONDITIONS.”
- All temperatures, and all other monitored parameters for 1-RC-P-1B are within normal operating limits and are stable.

Based on the current conditions, which one of the following:

- is the required action, in accordance with 1-AP-9.00, AND
- describes the No. 2 seal design for 1-RC-P-1B during normal operation?

A. (1) Manually trip the Reactor, and secure the ‘B’ RCP within 5 minutes.
(2) The No. 2 seal is a film-riding seal consisting of a runner which rotates with the shaft and a non-rotating seal ring attached to the lower seal housing.

B. (1) Manually trip the Reactor, and secure the ‘B’ RCP within 5 minutes.
(2) The No. 2 seal is a rubbing-face seal comprised of a chrome carbide seal ring and a chrome-carbide runner that is fixed on the pump shaft.

C. (1) Shut down the plant, and secure the ‘B’ RCP within 8 hours.
(2) The No. 2 seal is a rubbing-face seal comprised of a chrome carbide seal ring and a chrome-carbide runner that is fixed on the pump shaft.

D. (1) Shut down the plant, and secure the ‘B’ RCP within 8 hours.
(2) The No. 2 seal is a film-riding seal consisting of a runner which rotates with the

shaft and a non-rotating seal ring attached to the lower seal housing.

K/A

Reactor Coolant Pump
003K4.07 Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals).
(CFR 41.7) (RO – 3.2)

K/A Match Analysis

This question matches the K/A statement by requiring the applicant to correctly remember the basic design of the No. 2 RCP seal (vs. the design of the No. 1 seal). The question further recalls the applicant to determine the correct required action, given an operationally valid scenario involving the failure of a No. 1 RCP seal. This part of the question indirectly tests knowledge of the No. 2 RCP seal; whether it is capable of minimizing leakage during a controlled plant shutdown, or whether an immediate trip and RCP shutdown is required.

Answer Choice Analysis

A. INCORRECT. The first part of the distractor is incorrect. This part is plausible because it is the action to take when seal leakoff is greater than 8 gpm and temperatures are increasing. However, the stem of the question states that all parameters are stable. The second part of the distractor is also incorrect, but plausible, because it describes the No. 1 RCP seal design exactly as the lesson plan describes it.

B. INCORRECT. The first part of the distractor is incorrect as detailed in A. above. The second part of the distractor is correct; the description of the No. 2 RCP seal design is taken directly from the RCP lesson plan.

C. CORRECT. As detailed in the CAUTION statement before step 39 of 1-AP-9.00, with a high No. 1 seal leakoff (>8gpm) condition and no other parameters rising, the required action is to shut down the plant and secure the affected RCP within 8 hours. The second part distractor is also correct, as detailed above.

D. INCORRECT. See above descriptions.

Supporting References

1. Surry Procedure 1-AP-9.00, “RCP ABNORMAL CONDITIONS,” rev. 28. Especially CAUTION statement before step 39.

2. Surry Lesson Plan ND-88.1-LP-6, “REACTOR COOLANT PUMPS,” rev. 20 dtd

12/16/09.

References Provided to Applicant

None.

Answer: C

29. 000004A2.25 2

Current conditions:

- The latest shutdown margin calculation indicates the shutdown margin has decreased to 1.5% delta-K/K.
- RCP ‘A’ is in operation and preparations are underway to shutdown the pump.
- RHR system is in operation.
- RCS pressure has been reduced to 300 psig.
- RCS temperature is 210°F.

Based on the current conditions, which one of the following completes the statements below?

A tube leak exists on the _______(1)_________. Shutdown Margin as defined by Technical Specifications is ___________(2)_________.

	(1)	(2)
A.	Seal water return heat exchanger.	met.
B.	Seal water return heat exchanger.	NOT met.
C.	In-service RHR heat exchanger.	met.
D.	In-service RHR heat exchanger.	NOT met.

K/A

Chemical and Volume Control

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: uncontrolled boration or dilution.

(CFR: 41.5/43.5/45.3/45.5) (RO – 3.8)

Question History: Modified Bank (Sequoyah 2009-302, Q#30). Changed plant conditions from operation to Cold Shutdown and its effects on shutdown margin versus control rod movement.

LOK: C/A LOD: 3

K/A Match Analysis

The RO applicant is required to identify a possible cause of an uncontrolled dilution via the CVCS and determine the actions required when shutdown margin has been reduced below 1.77% delta-K/K.

Answer Choice Analysis

NOTE: RCS pressure is maintained at approximately 300 psig when establishing RHR in shutdown cooling. An RCS temperature of 210°F is consistent with the current plant conditions. GOP-2.5, Unit Cooldown - 351°F to less than 205°F, doesn't provide a specific temperature band, but directs placing RHR in service once RCS pressure is approximately 300 psig.

A. INCORRECT. *Plausible because the first part of the answer is correct. Plausible because at Cold Shutdown (i.e. RCS temperature less than 200°F) the shutdown margin limit is 1.0% delta-K/K. However, in the Intermediate Shutdown mode (i.e. RCS temperature between 200°F and 547°F), the minimum shutdown margin is 1.77% delta-K/K.*

B. CORRECT. A leak on the Seal Water Return heat exchanger would allow Component Cooling (CC) water to enter the CVCS and cause an uncontrolled dilution of the RCS. The current shutdown margin of 1.7% delta-K/K does not meet the shutdown margin of 1.77% delta-K/K required for Intermediate Shutdown.

C. INCORRECT. *Plausible because a CC water cools the RHR heat exchanger. However, at the current RCS pressure (300 psig), a tube leak would result in flow from the RCS rather than into the RCS. In the Intermediate Shutdown mode, the minimum shutdown margin is 1.77% delta-K/K. Plausible because at Cold Shutdown (i.e. RCS temperature less than 200°F) the shutdown margin limit is 1.0% delta-K/K.*

D. INCORRECT. *Plausible because a CC water cools the RHR heat exchanger. However, at the current RCS pressure (300 psig), a tube leak would result in flow from the RCS rather than into the RCS. The second half of the choice is correct.*

Supporting References

1. Surry lesson plan ND-95.1-LP-10, "Dilution Accident," rev. 8, Obj B., pg. 6.
2. Surry lesson plan ND-88.2-LP-2, “Operations of RHR System”, Obj B., pp. 7-13.

3. T.S. 1.0 Definitions, Reactor Operations, pg. T.S 1.0-1, Amdnt 203
4. T.S. 3.12.G, Shutdown Margin, Amdnts 265 and 264.

References Provided to Applicant
None

Answer: B
30. 000004K6.36 2

Initial conditions:
- Unit 1 at 100% and stable with all systems in normal configuration

Current Conditions:
- 1D-E5 - (CHG PP TO REGEN HX Hi-Lo FLOW) is LIT
- VCT level is 49% and increasing
- Pressurizer level is 52% and decreasing
- Letdown flow is oscillating
- Letdown pressure is oscillating

Which ONE of the following identifies the failure that has resulted in the above indications?

A. 1-CH-FT-1122 (CHG FLOW) transmitter has failed high

B. 1-CH-PCV-1145 has failed closed

C. 1-CH-LCV-1460A (Letdown Isol Vlv) has failed closed

D. 1-CH-FCV-1122 (Chg Flow Cntrl Vlv) instrument air has been lost

K/A

Chemical and Volume Control
Knowledge of the effect of a loss or malfunction on the following CVCS components: Letdown pressure control to prevent RCS coolant from flashing to steam in letdown piping.
(CFR: 41.7/45.7) (RO – 2.9)

Question History: New.

K/A Match Analysis

The RO applicant is required to evaluate the plant conditions and determine that the backpressure valve (1-CH-PCV-1145) has failed open (PCV-1145 fails open on loss of air).

<u>Answer Choice Analysis</u>

A. CORRECT. A loss of instrument air to 1-CH-PCV-1145 will cause the valve to open. The resulting transient will allow the fluid to flash to steam and create a downstream pressure and temperature transient that will cause flow to be diverted around the ion exchangers and lift RV-1209 which will route high temperature water to the VCT.

B. INCORRECT. *Plausible because a failure of this pressure transmitter could cause 1-CH-PCV-1145 to open. However, since the transmitter senses pressure upstream of the valve a downscale failure would cause the valve to close rather than open*.

C. INCORRECT. *Plausible because closure of the 1-CC-TCV-103 would result in the same indications provided in the stem of the question. However, 1-CC-TCV-103 fails open on a loss of instrument air*.

D. INCORRECT. *Plausible because a failure of the temperature transmitter could cause 1-CC-TCV-103 to close. However, an upscale failure of this transmitter would cause 1-CC-TCV-103 to open.*

<u>Supporting References</u>

1. Surry lesson plan ND-88.3-LP-2, "Charging and Letdown," rev. 15, Obj C, p. 15.
2. 1D-F1, VCT HI TEMP, Rev.0
3. 1D-G3, DEMIN INL DIVERT HI TEMP, Rev. 1
4. Dwg # 11448-FM-072C, Sh. 4 of 5, Rev. 28

<u>References Provided to Applicant</u>
None

Answer: A
31. 000005A4.01 2

Unit 1 Initial Conditions:

- The plant is solid.
- RCS temperature is being controlled with the ‘A’ RHR pump running on the "A" RHR heat exchanger. The ‘B’ RHR pump is not running.
- RCS pressure is being controlled with the 1-CH-PCV-1145 (Letdown Pressure Control Valve) controller in AUTO and the 1-CH-FCV-1122 (Charging flow control valve) controller in MANUAL.

Current conditions:

- The ‘A’ RHR pump just tripped.

Based on the current conditions, which one of the following (1) describes the response of RCS pressure if NO operator actions are taken, AND (2) the action the operator is required to take to restore RCS pressure to its value before the ‘A’ RHR pump tripped?

A. (1) RCS pressure will increase
(2) Lower the setpoint on 1-CH-PCV-1145 controller to lower RCS pressure

B. (1) RCS pressure will decrease
(2) Raise the setpoint on 1-CH-PCV-1145 controller to raise RCS pressure

C. (1) RCS pressure will decrease
(2) Lower the setpoint on 1-CH-PCV-1145 controller to raise RCS pressure

D. (1) RCS pressure will increase
(2) Raise the setpoint on 1-CH-PCV-1145 controller to lower RCS pressure

K/A

005 Residual Heat Removal
005A4.01 Ability to manually operate and/or monitor in the control room: Controls and indication for RHR pumps.
(CFR 41.7 / 45.5 to 45.8) (RO – 3.6)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly apply knowledge of the integrated plant response to an RHR pump trip with the plant in an operationally valid situation (solid condition). The RO applicant is required to understand how monitoring the RHR pump status relates to expected plant response, as well as how to operate controls in manual to mitigate the effects of the RHR pump trip on RCS pressure.

Answer Choice Analysis

A. CORRECT. RCS pressure will increase. Once the dynamic pressure of the RHR pump goes away, 1-CH-PCV-1145 will close to maintain letdown pressure at setpoint, this will reduce the amount of water being removed from the RCS and since charging flow is constant, cause RCS pressure to increase (since it is solid). Lowering the setpoint on 1-CH-PCV-1145 will reduce RCS pressure to normal.

B. INCORRECT. Part 1 is incorrect, but plausible if the candidate confuses letdown pressure with RCS pressure or fails to identify that 1-CH-PCV-1145 is in automatic and controlling letdown pressure. Part 2 is incorrect, but plausibility is gained as it supports the first part of the distractor and would be the appropriate action is RCS pressure were to decrease.

C. INCORRECT. Part 1 is incorrect, but plausible if the candidate confuses letdown pressure with RCS pressure or fails to identify that 1-CH-PCV-1145 is in automatic and controlling letdown pressure. Part 2 is incorrect, but plausible as this part supports part 1 of the distractor and the candidate may think that 1-CH-PCV-1145 is reverse acting (refer to 1-CH-LCV-1122).

D. CORRECT. RCS pressure will increase. Once the dynamic pressure of the RHR pump goes away, 1-CH-PCV-1145 will close to maintain letdown pressure at setpoint, this will reduce the amount of water being removed from the RCS and since charging flow is constant, cause RCS pressure to increase (since it is solid). Part 2 is incorrect, but plausible as this part supports part 1 of the distractor and the candidate may think that 1-CH-PCV-1145 is reverse acting (refer to 1-CH-LCV-1122).

Supporting References

1. Modified from VC Summer 2004-301 Question 005A4.01.

2. Surry Lesson Plan ND-88.2-LP-2, “Operation of Residual Heat Removal System,” rev. 16 dtd. 11/21/06. Especially section 3.1, Solid Plant Ops.

References Provided to Applicant

None.

Answer: A
32. 000006K1.07 2

Which one of the following correctly states a condition that would result in automatic isolation (closure) of all the main feedwater regulating valves **AND** all the main feedwater bypass flow control valves?

Evaluate each condition listed below as a separate and individual event with the unit initally at 100% power with all systems in normal configuration.

A. Narrow range level in one steam generator 80% on 2/3 channels.

B. Reactor trip breakers open and median Tave 547° F.

C. Pressurizer pressure is 1760 psig.

D. Non-recoverable loss of vital bus 1-II.

K/A

Emergency Core Cooling
Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: MFW System.
(CFR: 41.2 to 41.9 /45.7 to 45.8) (RO – 2.9)

K/A Match Analysis

The RO applicant is required to recognize the parameters and protection logic trends that will lead to a Safety Injection actuation and cause a Main Feedwater Isolation.

Answer Choice Analysis

A. INCORRECT. This signal will not cause a complete FW isolation.

B. INCORRECT. *Plausible because with the reactor tripped and Median Tave below 554°F will cause a closure of the Main Feed Reg Valves but not a complete MFW isolation.*

C. CORRECT. When pressurizer pressure reaches 1780 psig an SI actuation will occur which will in turn cause a MFW isolation.

D. INCORRECT. *Plausible if the candidate confuses feedwater isolation with AFW starting (i.e., believes that AFW will not flow at the same time MFW flows).*

Supporting References

1. Surry lesson plan ND-91-LP-3, "SI System Operations," rev. 20, Objective A and D, pp 3-7, 13-14.

2. Surry lesson plan ND-89.3-LP-3, “Main Feedwater System”, Rev. 21, Objective B, pgs. 10-11.

References Provided to Applicant

None.

Answer: C

33. 000007A2.05 2

The following Unit 1 conditions exist:

- Pressurizer PORV leakage exists
- PRT pressure is 12 psig
- PRT gas samples indicate Xe-133 activity at $8x10^{-2}$ micro-curies/ml
- Operators have been directed to vent the PRT

Which one of the following describes where the PRT is required to be vented in accordance with 1-OP-RC-011, Pressurizer Relief Tank Operations?

A. Vent Vent System

B. Overhead Gas System

C. Through the Sample System to the Process Vent System

D. Process Vent System directly (I.E., not through the Sample System)

K/A

Pressurizer Relief / Quench Tank A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Exceeding PRT high-pressure limits.

K/A Match Analysis

The PRT normal pressure band is 2-4 psig and the high pressure alarm annunciates at 10 psig. Pressure can be vented to clear the alarm and get back within the normal operating band. The K/A requires the applicant to use knowledge of procedures to address exceeding a PRT high pressure limit. This question does require the applicant to have procedural knowledge of how to vent the PRT under a high pressure condition with activity in the PRT.

Answer Choice Analysis

A. Incorrect: due to the CAUTION statements in the OP. Plausible because the Vent Vent System is one procedurally directed option for venting the PRT under different conditions.

B. Correct: With Xe-133 activity above $5x10^{-2}$ micro-curie/ml, the CAUTION statements at the top of the PRT venting sections of 1-OP-RC-011 require the PRT to be vented to the Overhead Gas System.

C. Incorrect: due to the CAUTION statements in the OP. Plausible because the Sample System via the Process Vent System is one procedurally directed option for venting the PRT under different conditions.

D. Incorrect: due to the CAUTION statements in the OP. Plausible because the Process Vent System is one procedurally directed option for venting the PRT under different conditions.

<u>Supporting References</u>

1-OP-RC-011, Pressurizer Relief Tank Operations, Rev. 24.
Memory Level, RO, LOD=2, NEW

<u>References Provided to Applicant</u>
none

Answer: B
34. 000008A3.05 2

Unit 1 Initial Conditions:
- Power = 100%.

Current Conditions:
- Reactor Coolant System (RCS) Pressure is 900 psig and slowly lowering.
- Pressurizer level is offscale low.
- Containment pressure reached a maximum at 19 psia and is now 18 psia and slowly lowering.
- Steam Generator pressures and levels are equal and stable.

Based on the current conditions, which one of the following is the Component Cooling (CC) valve lineup? (consider that ONLY automatic actuations occurred, no manual re-positioning of valves)

1-CC-TV-105A, B, C   Reactor Coolant Pump (RCP) Cooler Return Valves
1-CC-TV-110A, B, C   Containment Air Recirc CC Return Valves

	<u>1-CC-TV-105A, B, C</u>	<u>1-CC-TV-110A, B, C</u>
A.	CLOSED	CLOSED
B.	OPEN	CLOSED

C.	CLOSED	OPEN
D.	OPEN	OPEN

K/A

008 Component Cooling Water
008A3.05 Ability to monitor automatic operation of the CCWS, including: Control of the electrically operated, automatic isolation valves in the CCWS.
(CFR 41.7 / 45.5) (RO – 3.0)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to apply knowledge of the automatic isolation valve logic for important CCW valves, given an operationally valid scenario of an SI signal, Hi-CLS signal, but NO HI-HI-CLS signal.

Answer Choice Analysis

A. INCORRECT. The question stem states that RCS Pressure is 900 psig and lowering; therefore, an SI signal was actuated. Also, with a peak containment pressure of 19 psia, a HI-CLS signal would be actuated (HI-CLS setpoint is 17.7 psia), but a HI-HI-CLS signal would NOT have actuated (HI-HI-CLS setpoint of 23 psia). The Surry lesson plan for CCW states that the -109 valves auto close on SI, the -110 valves auto close on HI-HI CLS, and the -105 valves also auto close only on HI-HI CLS. Therefore, the correct alignment is the -109 valves CLOSED, the -110 valves OPEN, and the -105 valves OPEN. All other combinations are plausible misconceptions of the correct alignment.

B. INCORRECT. See analysis of ‘A’ above.

C. INCORRECT. See analysis of ‘A’ above.

D. CORRECT. See analysis of ‘A’ above.

Supporting References

1. Surry Lesson Plan ND-88.5-LP-1, “COMPONENT COOLING WATER,” rev. 23.

2. Surry Lesson Plan ND-88.4-LP-2, “CONTAINMENT VESEEL,” rev. 12. Page 15 gives the logic setpoints for the CLS signals.

3. Extensively modified from a DC Cook 2001 008A3.05 ILO exam question to apply to Surry.

References Provided to Applicant

None.

Answer: D

35. 000008K4.09 2

Unit 1 Initial Conditions:

- Power = 100%.
- Component Cooling (CC) Pump 1-CC-P-1B is RUNNING.
- An electrical fault caused a loss of normal offsite power to the '1H' Emergency Bus.
- Operators took action in accordance with 1-AP-10.07, "LOSS OF UNIT 1 POWER," and have re-energized the 'stub bus' from the #1 EDG.
- Operators have completed 1-AP-10.07 and are at the 'when/then' step to initiate 0-AP-10.08, STATION POWER RESTORATION, once the cause of power loss is corrected.

Current Conditions:

- Time = 0800. A Large Break LOCA occurs.
- Time = 0802. Containment pressure peaks at 28 psia.
- Time = 0810. 1-CC-P-1B trips.

Which one of the following completes the below statements?

(1) The standby CC pump low discharge header pressure auto start setpoint is ____________________ .

(2) Based on the current conditions, when CC discharge pressure reaches the low discharge header pressure auto start setpoint, CC Pump 1-CC-P-1A will ____________________ .

A. (1) 55 psig
(2) automatically start on low discharge header pressure

B. (1) 55 psig
(2) NOT automatically start on low discharge header pressure

C. (1) 75 psig
(2) automatically start on low discharge header pressure

D. (1) 75 psig
   (2) NOT automatically start on low discharge header pressure

K/A

008 Component Cooling Water
008K4.09 Knowledge of the CCWS design feature(s) and/or interlock(s) which provide for the following: The “standby” feature for the CCW pumps.
(CFR 41.7) (RO – 2.7)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to demonstrate knowledge of the correct auto-start setpoint for low CCW discharge pressure, as well as demonstrate that there is no interlock preventing the standby CCW pump from auto-starting in the presence of a HI-HI-CLS signal, although the procedures are full of CAUTIONS stating that this is not a desired condition.

Answer Choice Analysis

A. INCORRECT. The Surry lesson plan for Component Cooling Water (CC) states: “The standby pump will auto start on a low discharge header pressure of 55 psig.” Working through the 1-AP-10.07 procedure for a loss of normal offsite power to the ‘1H’ bus, the standby CC pump (1-CC-P-1A) will be placed in PTL, the stub bus energized from the diesel, and then the CC pump switch will be returned to AUTO-AFTER-STOP. There is an electrical interlock preventing the standby CC pump from auto starting. Auto Start Inhibit will be active due to the HI HI CLS signal.

B. CORRECT.

C. INCORRECT. Parts (1) and (2) both incorrect. See analysis of ‘A’ above.

D. INCORRECT. Part (1) is incorrect, part (2) is correct. See analysis of ‘A’ above.

Supporting References

1. Surry Lesson Plan ND-88.5-LP-1, “COMPONENT COOLING SYSTEM,” rev. 23.

2. Surry Procedure 1-AP-10.07, “LOSS OF UNIT 1 POWER,” rev. 54. Especially NOTE before step 47.

2. Surry Lesson Plan ND-90.3-LP-7, “STATION SERVICE AND EMERGENCY DISTRIBUTION PROTECTION AND CONTROL,” rev. 23.

References Provided to Applicant

None.

Answer: B
36. 000010K5.02 2

Unit 1 Initial Conditions:

- 100% Power.
- Pressurizer Relief Tank (PRT) pressure is 2.2 psig.
- A transient causes a Pressurizer (PZR) PORV to open.
- The PZR PORV will not reseat, and operators are unable to close the associated PORV block valve.
- Maximum PZR pressure was 2340 psig and has continually DECREASED from that value.

Current conditions:

- Containment pressure is 10.6 psia.
- PZR Pressure is 1550 psig and DECREASING at 5 psig/min.
- PRT Pressure is 90 psig and INCREASING at 10 psig/min.

Based on the current conditions, and assuming a completely ideal thermodynamic process, which one of the following is

(1) the expected temperature trend observed on PZR PORV downstream temperature instrument (1-RC-TE-1463) from the maximum observed PZR pressure to the current conditions

AND

(2) the expected temperature value read on 1-RC-TE-1463 two minutes after the current conditions, considering no operator actions and no significant change in the Pressurizer PORV leakrate for the two minute interval?

	Expected PORV discharge temp trend (1-RC-TE-1463)	Expected PORV discharge temp trend (1-RC-TE-1463) two minutes later
A.	INCREASING	GREATER THAN current

temperature on 1-RC-TE-1463

B. INCREASING LESS THAN current

temperature on 1-RC-TE-1463

C. DECREASING GREATER THAN current

temperature on 1-RC-TE-1463

D. DECREASING LESS THAN current

temperature on 1-RC-TE-1463

K/A

Pressurizer Pressure Control
010K5.02 Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Constant enthalpy expansion through a valve.
(CFR 41.5 / 45.7) (RO - 2.6)

K/A Match Analysis

The question author wanted to examine the applicants' fundamental understanding of the thermodynamic process, instead of a GFES-style temperature-lookup-with-the-steam tables question. The question matches the K/A statement by requiring the applicants to correctly understand how the Mollier diagram describes the constant enthalpy (throttling) process for an operationally valid situation. To arrive at the correct answer, the applicant must either use the Mollier diagram to determine that downstream tailpipe temperatures would be increasing, until the point at which the PRT rupture disc fails (at 100 psig), at which time the PZR PORV will be exhausting to a containment volume that is at an initially subatmospheric pressure.

Answer Choice Analysis

A. INCORRECT. Part (1) of this distractor is correct; as the PORV discharges to the saturated PRT system and PRT pressure increases, the downstream/tailpipe temperature is expected to increase (essentially following the saturation temperature for the given PRT pressure). Part (2) of this distractor is incorrect; two minutes after the current conditions, the PRT rupture disc will fail, and the PORV will be exhausting to the containment volume which is initially at a vacuum. Because the rupture disc fails, the expected TE-463 temperature will be less than the reading at "current" conditions. It is plausible for an applicant to believe that the pressure would continue to increase if the applicant does not recognize that the PRT will rupture at 100 psig.

B. CORRECT. See discussion for A. above. Plotting a few points will also get the correct answer:
-at PZR P=2340 psig, PRT P=2.2 psig, TE-463 ~ 220 °F
-at PZR P=2000 psig, PRT P=4.2 psig, TE-463 ~ 226 °F
-at PZR P=1700 psig, PRT P=20 psig, TE-463 ~ 258 °F
-at PZR P=1550 psig, PRT P=90 psig, TE-463 ~ 331 °F ["current" conditions]
-at PZR P=1450 psig, PRT P=Containment P ~11 psia, TE-463 ~ 198 °F [conditions 2 min after "current" conditions].

C. INCORRECT. Part (1) of this distractor is incorrect. This part of the distractor is plausible if the applicant believes that the downstream temperature will follow the PZR steam space temperatures, which will be decreasing as the pressurizer depressurizes. The second part of the distractor is plausible if the applicant mis-applies the Mollier diagram and believes that the peak saturation line pressure is 1500 psia (instead of the correct 500 psia). If the PZR was at 500 psia, as it continued to depressurize it could enter the superheated steam region.

D. INCORRECT. Part (1) of this distractor is incorrect, as discussed above in the analysis of distractor C. Part (2) of this distractor is correct; in this case, plausibility is actually enhanced by the part (1) misconception, if the applicant simply believes that the current trends will continue for a two minute interval.

<u>Supporting References</u>

1. Surry Lesson Plan ND-88.1-LP-3-DRR, "Pressurizer and Power Relief," rev. 16 dtd 07/07/08. Designates TE-463 as downstream PORV temperature instrument, PRT rupture disc set for 100 psig.

2. Steam Tables.

<u>References Provided to Applicant</u>

Steam Tables.

Answer: B
37. 000012K2.01 1

Unit 1 Initial Conditions:

- 5% Power.

- The **'A'** reactor trip breaker and the **'B'** reactor trip **bypass** breaker are CLOSED.

Current conditions:

- The **'A'** DC bus loses power/de-energizes.

Based on the current conditions, which one of the following is the automatic response of the Reactor Protection System?

	'A' Reactor Trip Breaker	'B' Reactor Trip Bypass Breaker
A.	OPEN	CLOSED
B.	CLOSED	OPEN
C.	CLOSED	CLOSED
D.	OPEN	OPEN

K/A

012 Reactor Protection
012K2.01 Knowledge of bus power supplies to the following: RPS channels, components, and interconnections.
(CFR 41.7) (RO – 3.3)

K/A Match Analysis

The question matches the K/A by straightforwardly examining the RO applicant's knowledge of the bus power supplies to RPS channels, with a slightly abnormal situation (a Rx trip bypass breaker closed) in order to raise the question's LOD.

Answer Choice Analysis

A. INCORRECT. Plausible if applicant believes "B" train reactor trip bypass breaker UV coils are powered from "B" 125 VDC.

B. INCORRECT. Plausible if applicant believes "A" train reactor trip breaker UV coils are powered from "B" 125 VDC.

C. INCORRECT. Plausible if applicant does not recognize that 125 VDC powers reactor trip and bypass breakers.

D. CORRECT. The "A" 125V DC train supplies power to both the "A" reactor trip UV trip coils and the "B" reactor trip bypass breaker UV trip coils.

Supporting References

1. Surry Lesson Plan ND-93.3-LP-10, “Reactor Protection – General,” rev. 7, dtd 12/04/2008.

2. Surry Lesson Plan ND-90.3-LP-6, “125 VDC Distribution,” rev. 16, dtd 01/05/2010.

3. Slightly modified (not enough to not consider it ‘BANK’) from SURRY ILO exam 2006-301 012K2.01. This question was also based on Surry examination bank question number DC00012.

References Provided to Applicant

None.

Answer: D
38. 000012K3.04 2

Unit 1 Initial Conditions:

- Time = 0300.
- Power = 100%
- No equipment out of service.
- Pressurizer (PZR) Pressure channel 1-RC-PT-1455 (PRZR PRESS PROTECT CH1) unexpectedly fails high.

Current conditions:

- Time = 0850 (same day). All required bistables for 1-RC-PT-1455 have been placed in the TRIP condition.

Based on the current conditions, which one of the following identifies the Reactor Protection System (RPS) and Engineered Safety Function (ESF) actuation logic coincidence required, from the remaining in-service PZR Pressure protection channels, to initiate an automatic low PZR Pressure Reactor Trip and automatic Low PZR Pressure Safety Injection at time 0851?

	Low PZR Pressure Rx Trip Actuation Logic Coincidence	Low PZR Pressure Safety Injection Actuation Logic Coincidence
A.	1/3	1/3

B.	1/2	1/2
C.	2/3	2/3
D.	1/3	1/2

K/A

012 Reactor Protection
012K3.04 Knowledge of the effect that a loss or malfunction of the RPS will have on the following: ESFAS.
(CFR 41.7 / 45.6) (RO – 3.8)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly apply knowledge of the actuation logic coincidence of the low pressurizer pressure reactor trip signal and the low pressurizer SI signal, given an operationally valid situation involving a malfunction of an RPS component, along with taking the Tech Spec required actions for that malfunction (placing the channel bistables to TRIP condition).

Answer Choice Analysis

A. INCORRECT. Auto Rx Trip and auto SI is normally 2/3 for low pressurizer pressure. Channel 455 feeds both circuits. When a protection channel is removed from service, all bistables are tripped in all cases (except for the AUTO hi-hi CLS/Containment Spray actuation). Therefore, an auto SI will occur is either of the two remaining bistables trip, and an auto Rx trip will occur is either of the two remaining bistables trip. 1/3 and 2/3 are credible distractors because the applicant must know what state bistables will be in, and that some RPS functions (such as high PRNI flux) are 2 out of 4 logic under normal conditions.

B. CORRECT. See analysis of 'A' above.

C. INCORRECT. See analysis of 'A' above.

D. INCORRECT. See analysis of 'A' above.

Supporting References

1. Modified from Harris 2006 Question 012K3.04.

2. Surry Procedure 0-DRP-004, "PRECAUTIONS, LIMITATIONS AND SETPOINTS," rev. 63. Especially attachment 1, "REACTOR PROTECTION SYSTEM SETPOINTS,"

1.A.1. and 2.E.

3. Surry Technical Specifications 3.7, "INSTRUMENTATION SYSTEMS," especially Table 3.7-1, Table 3.7-2, and associated ACTIONS.

4. Surry Lesson Plan ND-93.3-LP-10, "Reactor Protection – General," rev. 7 dtd. 12/04/08.

References Provided to Applicant

None.

Answer: B
39. 000013K5.02 2

On Unit 1, 1-MS-PT-1474 (SG 'A' PRESS PROTECT CH 2) is out of service and Technical Specifications actions have been taken for that channel to place it in trip.

Which one of the following is now the minimum logic necessary for a valid, reliable Safety Injection (SI) signal to occur?

A. High steam flow on **1** of 2 channels on **one** of the three steam lines in coincidence with low pressure on **either** 'B' or 'C' steam line.

B. High steam flow on **1** of 2 channels on **two** of the three steam lines in coincidence with low pressure on **either** 'B' or 'C' steam line.

C. High steam flow on **2** of 2 channels on **one** of the three steam lines in coincidence with low pressure on **both** 'B' and 'C' steam lines.

D. High steam flow on **2** of 2 channels on **two** of the three steam lines in coincidence with low pressure on **both** 'B' and 'C' steam lines.

K/A

Engineered Safety Features Actuation
0013 K5.02 Knowledge of the operational implications of the following concepts as they apply to ESFAS: Safety system logic and reliability.
(CFR 41.5 /45.7) (RO – 2.9)

K/A Match Analysis

This question matches the K/A statement by requiring the applicants to correctly identify the logic of the SI actuation system (ie. system logic) with one steam line pressure detector out of service (i.e. reliability).

<u>Answer Choice Analysis</u>

A. INCORRECT. *Plausible because it only requires a high steam flow condition to be sensed on 1 of 2 channels, but it requires a high steam flow condition on 2 of 3 steam generators rather than only one steam generator. The logic for low steam pressure is correct. With one detector out of service its bistable would be placed in the tripped condition, so that a low steam line pressure on either 'B' or 'C' steam line will satisfy the 2 of 3 logic for low steam line pressure.*

B. CORRECT. A high steam flow condition sensed on 1 of 2 channels on 2 of 3 steam generators coincident with low steam pressure on either 'B' or 'C' steam line will result in an SI actuation.

C. INCORRECT. *Plausible because requiring 2 of 2 channels would ensure a channel faillure would not result in an input to the SI actuation logic. In addition, it requires a high steam flow condition on 2 of 3 steam generators rather than only one steam generator which ensures that a single channel failure would not result in an input to the SI actuation logic. The low steam line pressure on both 'B' and 'C' steam lines is plausible if the applicant doesn't recognize that the bistable for the failed detector is placed in the tripped condition as part of the Technical Specification actions.*

D. INCORRECT. *lausible because requiring 2 of 2 channels would ensure a channel faillure would not result in an input to the SI actuation logic. The response contains the correct logic for high steam flow (i.e. 2 of 3 steam generators). The low steam line pressure on both 'B' and 'C' steam lines is plausible if the applicant doesn't recognize that the bistable for the failed detector is placed in the tripped condition as part of the Technical Specification actions.*

<u>Supporting References</u>

1. Surry lesson plan ND-91-LP-3, Safety Injection System Operations, Rev. 22, Obj. A, pg. 7

<u>References Provided to Applicant</u>

None.

Answer: B
40. 000022G2.1.7 2

Current conditions:

- Unit 1 is operating at 100% power in mid August
- 1-CD-REF-1A ('A' Chilled Water Refrigeration Unit) is operating.
- The following alarms on annunciator panel 1B are lit:
    - A6 – CTMT PART PRESS -0.1 CH 1
    - B6 – CTMT PART PRESS -0.1 CH 2
- Bulk Containment temperature on the Plant Computer shows a slowly rising trend.
- Three control rod drive mechanism (CRDM) fans are running.
- Three Containment Air Recirculation Fans (CARF) are running.

Based on the above conditions, which ONE of the following actions will clear the containment partial pressure alarms?

A. Place the in-service Containment Vacuum Pump control switch to OFF.

B. Increase Component Cooling (CC) water flow in Header ‘A’.

C. Increase cooling through the operating Chilled CC heat exchanger.

D. Start an additional CRDM cooling fan.

K/A

Containment Cooling
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrumentation interpretation.
(CFR: 41.5/43.5/45.12/45.13) (RO – 4.4)

K/A Match Analysis

The RO applicant is required to know the relation between containment partial pressure, containment temperature and the operating characteristics of the containment ventilation cooling and containment vacuum systems to determine the correct action.

Answer Choice Analysis

A. INCORRECT. *Plausible because the operating containment vacuum pump can cause partial pressure to decrease and it is an action directed by ARPs 1B-A6, -B6. However, it would not explain the increasing trend on Containment temperature and the vacuum pump automatically stops when partial pressure is -0.1 from setpoint.*

B. INCORRECT. *Plausible because CC provides cooling to the CRDM coolers and can provide cooling to the CARFs. However, under these conditions (i.e., full power)* *the CC Chilled Water system would be aligned to provide cooling to the CARFs and CC header 'B' provides cooling to Containment Ventilation, not CC header 'A'.*

C. CORRECT. Partial pressure can decrease as a result of either decreasing containment pressure or as a result of increasing containment temperature (i.e. increasing Psat). So a decreasing partial pressure coupled with increasing containment temperature would mean more containment cooling was needed. Increasing coolant flow through the operating CC Chilled Water chiller will cool the containment air passing through the CARFs and in turn cause the containment temperature to decrease and bring the partial pressure back to its setpoint. As containment temperature increases vapor pressure increases non-linearly. Partial pressure is determined by subtracting the actual containment pressure from the calculated vapor pressure to establish the partial pressure for the water vapor. The partial air pressure is determined by subtracting the partial pressure due to water vapor from actual containment pressure.

D. INCORRECT. *Plausible because there are six CRDM coolers. However, there are three control switches with two CRDM fans controlled by each switch. Only one fan can be operated at a time using these switches. With three CRDM coolers operating it would not be possible to start an additional CRDM cooler*.

<u>Supporting References</u>

1. Surry lesson plan ND-88.4-LP-6, "Containment Ventilation," rev. 9, Obj A & B, pp. 3-5.

2. Surry lesson plan ND-88.4-LP-5, "Containment Vacuum," rev. 12, Obj A, pp. 3-7.

3. 1B-A6, CTMT PART PRESS -0.1 CH 1, Rev. 7

<u>References Provided to Applicant</u>

None.

Answer: C
41. 000026K2.01 1

Unit 1 conditions:
- Time = 0930: Large Break LOCA
- Time = 1002: Loss of Offsite Power

Based on the above conditions, as the Emergency Buses energize, which one of the following states (1) which recirc spray pumps start first and (2) the reason for this action?

A. (1) Inside Recirc Pumps
(2) Their spray header is full of water which results in spraying the containment sooner

B. (1) Inside Recirc Pumps
(2) Their starting current is larger so starting them earlier in the sequence will minimize the chance of overloading the DGs

C. (1) Outside Recirc Pumps
(2) Their spray header is full of water which results in spraying the containment sooner

D. (1) Outside Recirc Pumps
(2) Their starting current is larger so starting them earlier in the sequence will minimize the chance of overloading the DGs

<u>K/A</u>

Containment Spray: Knowledge of bus power supplies to the following: Containment spray pumps.

<u>K/A Match Analysis</u>

Requires knowledge of how power is sequenced to the Containment Spray Pumps after power was lost.

<u>Answer Choice Analysis</u>

A. INCORRECT. The Outside Containment Recirc Pumps are started first. 1st part is plausible because the Inside Containment Recirc Pumps have shorter distance from the pumps to the spray header. 2nd part is correct.

B. INCORRECT. The Outside Containment Recirc Pumps are started first. 1st part is plausible because the Inside Containment Recirc Pumps have shorter distance from the pumps to the spray header. 2nd part is incorrect but plausible because assuming that this Pump will pump more water when starting, it will have a higher starting current.

C. CORRECT. The Outside Containment Recirc Pumps are started first because the headers are kept full of water by a check valve. This results in water spraying the containment as soon as the pump starts which limits containment pressure.

D. INCORRECT. 1st part is correct. 2nd part is incorrect but plausible because assuming that this Pump will pump more water when starting, it will have a higher starting current.

Supporting References

ND-90.3-LP-7 Obj: D

References Provided to Applicant

None.

Answer: C
42. 000039A1.03 4

Unit 1 Initial Conditions (1000 hrs):

- Reactor power is at 2%
- RCS temperature is 550°F
- Main steam lines are being heated using the Main Steam Trip Valve bypass valves

Current Conditions (1015 hours):

- RCS temperature is stable at 524°F

Based on the current conditions, which one of the following states (1) if the technical specification minimum temperature for criticality has been violated, and (2) if Technical Specification cooldown rate limits have been exceeded?

A. (1) Minimum temperature for criticality has been violated.
(2) Cooldown rate limits exceeded.

B. (1) Minimum temperature for criticality has NOT been violated.
(2) Cooldown rate limits NOT exceeded.

C. (1) Minimum temperature for criticality has NOT been violated.
(2) Cooldown rate limits exceeded.

D. (1) Minimum temperature for criticality has been violated.
(2) Cooldown rate limits NOT exceeded.

K/A
Main and Reheat Steam
A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Primary system temperature indications, and required values, during main steam system warm-up.

K/A Match Analysis

The question requires knowledge of procedure actions (in this case the procedure is Tech Specs) as a result of exceeding the minimum temperature for criticality during the warming of the main steam lines. Therefore, the K/A is met because the applicant must monitor the change in RCS temp and know what actions to take based on that observed temperature change.

Answer Choice Analysis

A. INCORRECT. The first part is correct. The second part is incorrect but plausible because the cooldown rate, if applied over 15 minutes, would be exceeded

B. CORRECT. The minimum temp for criticality is 522 F. The cooldown causes RCS temp to be below this, which requires a reactor trip. The tech spec cooldown limit calculated over 15 minutes is in excess of the 100 F/hr Tech Spec limit; however, this would be a misapplication of the Tech Spec cooldown limits which are to be applied over a one hour period and not a 15 minute period. The second part is incorrect, but plausible because the applicant could misapply the cooldown rate limits over 15 minutes.

C. INCORRECT. The first part is not correct as discussed above, but it is plausible because not tripping, but simply closing MSIV bypass valves would be a method to lower the cooldown rate.

D. INCORRECT. The first part is not correct but plausible because the applicant may recognize that cooldown rates are not exceeded. The second part is correct.

Supporting References

- Tech Spec 3.1.E, Reactor Coolant System – Minimum Temperature for Criticality.
- Tech Spec 3.1.B, Reactor Coolant System - Heatup and Cooldown.

References Provided to Applicant

None.
The licensee will need to help supply supporting material to justify whether a reactor trip is required. Verbally, the licensee stated that a reactor trip was required, but the supporting references tend to state that being below the limit is not allowed, but then does not state what actions are required because of being below the limit. Either B or D is correct, pending research by the licensee.

Answer: B
43. 000059A2.11 2

Unit 1 Initial Conditions:

- Power = 15% and stable for a chemistry hold.
- 1-MS-PT-1446 (TURB 1st STAGE PRESS CH 3) is the selected channel.
- All Feed Regulating Valves (FRV) are controlling steam generator (S/G) levels in AUTO.

Current conditions:

- Time = 1440. 1-MS-PT-1446 fails high over a two minute period.
- Time = 1442. The operator realizes that 1-MS-PT-1446 has failed and begins to take the appropriate actions.

Based on the current conditions, which one of the following (1) is the response of the steam generator level control system to the 1-MS-PT-1446 failure **IF** no operator actions are taken, AND (2) the required operator actions to restore the steam generator level control system, in accordance with 0-AP-53.00, "LOSS OF VITAL INSTRUMENTATION/CONTROLS?"

A. (1) S/G levels will stabilize at 44%.
(2) VERIFY 1-MS-PT-1447 (TURB 1st STAGE PRESS CH 4) indications are NORMAL and then select 1-MS-PT-1447. Once S/G levels are stable then place all FRVs to MANUAL and then back to AUTO to remove windup from the controllers.

B. (1) S/G levels will stabilize at 44%.
(2) Place all FRVs to MANUAL, select 1-MS-PT-1447(TURB 1st STAGE PRESS CH 4), reduce SG levels to program, and then place the FRVs back to AUTO.

C. (1) S/G levels will increase until the High S/G level trip setpoint is reached.
(2) Place all FRVs to MANUAL, then Select 1-MS-PT-1447(TURB 1st STAGE PRESS CH 4), then place the FRVs back to AUTO.

D. (1) S/G levels will increase until the High S/G level trip setpoint is reached.
(2) VERIFY 1-MS-PT-1447 (TURB 1st STAGE PRESS CH 4) indications are NORMAL, then Select 1-MS-PT-1447. Once S/G levels are stable, place all FRVs to MANUAL and then back to AUTO to remove windup from the controllers.

K/A

059 Main Feedwater
059A2.11 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of feedwater control system.

(CFR 41.5 / 43.5 / 45.3 / 45.13) (RO – 3.0)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly predict the impact of a failure of the feedwater control system (a failure of the selected impulse pressure channel, which is an input into SGWLC control), and to the demonstrate knowledge of the correct actions to take in accordance with plant procedures to mitigate the effects of the feedwater control system failure.

Answer Choice Analysis

A. INCORRECT. Part (1) of this distractor is correct, S/G levels will stabilize at 44%, which is the SGWLC reference level setpoint from 20-100% power. The Surry lesson plan on SGWLC states the following for the selected Pimp channel failing high: "If turbine power was <20% and the FRV was in automatic, the FRV will throttle open to increase S/G level to 44%. Part (2) of this distractor is incorrect. The same lesson plan states: "The required operator action for a Pimp failure either high or low will be to place the FRVs in manual and then select the unaffected Pimp channel for SGWLC. The operator can then place the FRVs back in automatic." This statement is a summary of actions contained in Surry procedure 0-AP-53.00, specifically in Attachment 2 of this procedure. The part (2) distractor is plausible, because taking these actions will restore S/G levels, and will also remove windup from the controllers; however, the sequence is incorrect as laid out in the AP procedure.

B. CORRECT. See analysis of 'A' above.

C. INCORRECT. See analysis of 'A' above. Part (1) is incorrect, but plausible, because a failure of the pressure channel high does cause S/G levels to go up—it just 'clips' the signal at the high level setpoint (44%), instead of increasing without bound.

D. INCORRECT. See analysis of 'A' and 'C' above. Both parts (1) and (2) incorrect.

Supporting References

1. Surry Lesson Plan ND-93.3-LP-8, "SG WATER LEVEL CONTROL SYSTEM," rev. 7 dtd 09/27/04.

2. Surry Procedure 0-AP-53.00, " LOSS OF VITAL INSTRUMENTATION/CONTROLS," rev. 15.

References Provided to Applicant

None.

Answer: B
44. 000059A4.03 2

Unit 1 Initial Conditions:

- Power = 96%.
- The unit has operated at full power for 356 days of continuous operation, and has been in a "power coastdown" for the past four days.
- RCS Boron concentration is 20 ppm.

Current conditions:

- Time = 0004. The operations team has just commenced a power reduction in accordance with 1-GOP-2.1, "UNIT SHUTDOWN, POWER DECREASE FROM ALLOWABLE POWER TO LESS THAN 30% REACTOR POWER," in order to begin a refueling outage.
- An operator inadvertently opens 1-FW-FCV-150A, 1A Main Feed Pump Recirc Valve.

Based on the current conditions, which one of the following completes the below statement, in accordance with 1-GOP-2.1?

When the Main Feed Pump Recirc Valve opens, Feedwater temperature will ______(1)______ which will add ______(2)______ to the core as compared to the exact same situation at beginning of life (BOL).

A. (1) decrease
   (2) a larger amount of positive reactivity

B. (1) decrease
   (2) a smaller amount of positive reactivity

C. (1) increase
   (2) a larger amount of negative reactivity

D. (1) increase
   (2) a smaller amount of negative reactivity

K/A

059 Main Feedwater
059A4.03 Ability to manually operate and monitor in the control room: Feedwater control during a power increase and decrease.
(CFR 41.7 / 45.5 to 45.8) (RO – 2.9)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly apply knowledge of the integrated plant response to an inadvertent operation of the main feed recirc valve during a downpower situation.

Answer Choice Analysis

A. CORRECT. A NOTE before step 5.3.8 of 1-GOP-2.1 states the following: “When a Main Feed Pump Recirc Valve opens, a decrease in FW temperature will occur that can add positive reactivity to the core. The magnitude of the reactivity change is dependent on the time in core life and the value of the Moderator Temperature Coefficient.” Therefore, FW temperature will decrease and part (1) is correct. Because the given situation is at the extreme EOL, the magnitude of the MTC is at its greatest point (most negative) as compared to BOL, when its magnitude is smaller. Therefore, the reactivity effect is greater than the effect of the same transient at BOL, and part (2) is also correct.

B. INCORRECT. See analysis of ‘A’ above. Part (1) is correct, part (2) incorrect. A smaller magnitude of reactivity change is plausible if the applicant wrongly believes that the magnitude of the MTC is less at EOL than at BOL.

C. INCORRECT. See analysis of ‘A’ above. Part (1) is incorrect, part (2) would be correct if part (1) was correct.

D. INCORRECT. See analysis of ‘A’ and ‘B’ above. Both parts (1) and (2) incorrect.

Supporting References

1. Surry Procedure 1-GOP-2.1, “UNIT SHUTDOWN, POWER DECREASE FROM ALLOWABLE POWER TO LESS THAN 30% REACTOR POWER,” rev. 34. Especially NOTE before step 5.3.8.

2. Surry Procedure 1-DRP-003, “CURVE BOOK,” rev. 90. Attachments 26 and 46.

References Provided to Applicant

None.

Answer: A
45. 000061A3.03 5

With the unit initially at 100% power, the control room team closes the AFW MOVs to the "A" steam generator for a special test. No other switches are manipulated for this test.

Subsequently, an inadvertant safety injection occurs.

Which ONE of the following correctly states the expected response of the AFW system 35 seconds after the safety injection?

A. All AFW water pumps are running with AFW aligned to all steam generators.

B. Only the steam driven AFW pump in service to only "B" and "C" steam generators.

C. All AFW pumps are running with AFW aligned to only "B" and "C" steam generators.

D. Only the steam driven AFW pump in service with AFW aligned to all steam generators.

K/A

Auxiliary Feedwater System
Ability to monitor automatic operation of the AFW, including: AFW Steam Generator level control on automatic start.
(CFR: 41.7/45.5) (RO – 3.9)

K/A Match Analysis

The RO applicant is required to determine whether the steam generator level control valves (1-FW-MOV-151 valves) will automatically open when an AFW auto-start signal (SI initiation) is received and a reactor trip causes another signal.

Answer Choice Analysis

A. INCORRECT. Plausible if the candidate believes that the AFW pumps will start on low level and the SI signal did cause the reactor trip (i.e., open signal already present).

B. INCORRECT. Plausible as the steam driven AFW pump will start, but incorrect as 'A' AFW MOVs will open.

C. INCORRECT. Plausible if the candidates believes that the AFW pumps will all start on low level and does not recognize the auto open signal for 'A' AFW MOVs.

D. CORRECT. Steam Driven AFW pump will start, motor drivens will be delayed for 50 seconds (due to SI).

Supporting References

1. Surry lesson plan ND-89.3-LP-4, "Auxiliary Feed System," rev. 26, Obj F, pgs. 15-17.

References Provided to Applicant

None.

Answer: D
46. 000061K5.05 3

Current conditions on Unit 1:

- 100% power
- Locally, in Unit One Safeguards, the ‘J’ AFW header temperature indicates 168°F and increasing.

Which ONE of the following describes:
(1) the actions that should be taken in accordance with 1-AP-21.01, Response to AFW Check Valve Backleakage
AND
(2) the reason the action is taken?

A. (1) Close MOV-FW-151A, -151C, and -151E.
(2) To prevent S/G blowdown in the event of a piping rupture upstream of the valve.

B. (1) Close MOV-FW-151B, -151D, and -151F.
(2) To prevent S/G blowdown in the event of a piping rupture upstream of the valve.

C. (1) Close MOV-FW-151A, -151C, and -151E
(2) To prevent vapor binding of the AFW headers and pump casings.

D. (1) Close MOV-FW-151B, -151D, and -151F
(2) To prevent vapor binding of the AFW headers and pump casings.

Auxiliary Feedwater System
Knowledge of the operational implications of the following concepts as they apply to the AFW: Feed line voiding and water hammer.
(CFR: 41.5/45.7) (RO – 2.7)

K/A Match Analysis

The RO applicant is required to recognize that the elevated AFW header temperature indicates leakage around the stop check valves and the operational implications would be the potential for voiding of the AFW headers and pump casings.

Answer Choice Analysis

A. INCORRECT. *Plausible because the listed valves are the 151 MOVs for each of the steam generators on the 'A' AFW header. Also, prevention of S/G blowdown is the purpose FW check valves on the feedwater headers.*

B. INCORRECT. *Plausible because these are the correct 151 MOVs that are closed to isolate the 'B' AFW header from the steam generators. Also, prevention of S/G blowdown is the purpose FW check valves on the feedwater headers.*

C. INCORRECT. *Plausible because the listed valves are the 151 MOVs for each of the steam generators on the 'A' AFW header. These valves are closed because the elevated temperature indicates leakage around the stop check valves. This can lead to voids developing in the AFW headers and pump casings and potentially result in vapor binding of the AFW pumps*.

D. CORRECT. When AFW Header temperature exceeds 165°F, Step 4 of 1-AP-21.01 directs the user to close either the individual 151 MOVs or to close all the 151 MOVs on the affected AFW header. The listed valves are the 151 MOVs for each of the steam generators on the 'B' AFW header.

Supporting References

1. Surry lesson plan ND-89.3-LP-3, "Main Feedwater System," rev. 21, p. 5

2. Surry lesson plan ND-89.3-LP-4, "Auxiliary Feed System," rev. 26, Obj C, p. 17.

3. 1-AP-21.01, Response to AFW Check Valve Backleakage, Rev. 5, pg. 2

References Provided to Applicant

None.

Answer: D
47. 000062A1.01 2

Unit 1 conditions:
- Rx Trip with loss of offsite power occurred
- EDG 1 is powering Bus 1H
- EDG 3 is powering Bus 1J
- 1-CH-P-1A is in pull-to-lock

- 1-CH-P-1B is secured in auto
- 1-CH-P-1C is in service
- 1-AP-10.07, Loss of Unit 1 Power, Attachment 4, Emergency Bus Load Alignment, is in progress

Based on the above conditions, which one of the following states (1) the action required to be taken with charging pump 1-CH-P-1B in accordance with 1-AP-10.07, Attachment 4, and (2) the consequence if this action is not taken?

A. (1) Start 1-CH-P-1B
(2) Subsequent HI HI CLS actuation could overload EDG 3

B. (1) Start 1-CH-P-1B
(2) Subsequent HI HI CLS actuation could overload EDG 1

C. (1) Place 1-CH-P-1B in PTL
(2) Subsequent HI HI CLS actuation could overload EDG 3

D. (1) Place 1-CH-P-1B in PTL
(2) Subsequent HI HI CLS actuation could overload EDG 1

<u>K/A</u>

Electrical Distribution: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits.

<u>K/A Match Analysis</u>

Requires knowledge of Emergency Bus load limit when being powered from the EDGs.

<u>Answer Choice Analysis</u>

A. CORRECT. Per 1-AP-10.07, If EDG 3 is the sole source of power to Emergency Bus 1J, start 1-CH-P-1B. The purpose of Attachement 4 is stated as: To limit EDG loading so that a subsequent HI HI CLS actuation willnot overload the EDG.

B. INCORRECT. 1st part is correct. 2nd part is incorrect because 1-CH-P-1B is not powered from EDG 1. Plausible because EDG 1 does does provide power 1-CH-P-1A and 1C.

C. INCORRECT. Per 1-AP-10.07, If EDG 3 is the sole source of power to Emergency Bus 1J, start 1-CH-P-1B. 1st part is plausible because if -CH-P-1C was lined up to the same bus and was operating, it would be correct. 2nd part is correct.

D. INCORRECT. Per 1-AP-10.07, If EDG 3 is the sole source of power to Emergency Bus 1J, start 1-CH-P-1B. 1st part is plausible because if -CH-P-1C was lined up to the same bus and was operating, it would be correct.2nd part is incorrect because 1-CH-P-1B is not powered from EDG 1. Plausible because EDG 1 does does provide power 1-CH-P-1A and 1C.

Supporting References
1AP-10.07 Attachment 4, Emergency Bus Load Alignment

References Provided to Applicant
None.

Answer: A
48. 000062K3.01 3

Initial conditions:
0130 - Unit 1 a non-recoverable loss of the semi-vital bus occurs.

Current plant conditions:
0230 - Unit 1 tripped during the down power required due to the de-energized semi-vital bus

Based on current plant conditions, which one of the following actions are required to maintain Tave at 547^{o}F?

A. Dump steam using the steam dumps.

B. Dump steam using local operation of the SG PORVs.

C. Dump steam using auto controller operation of the SG PORVs.

D. Dump steam using manual controller operation of the SG PORVs.

K/A
062K3.01
AC Electrical Distribution
Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: Major system loads.

K/A Match Analysis
The question requires knowledge of how to maintain Tave in the presence of a loss of AC. The local dumping of steam is an effect of the AC malfunction.

Answer Choice Analysis
A. Incorrect. Plausible and incorrect because IAW Page 3 of 1-AP-10.05, dumps will be available in Tave mode for only 30 minutes after the loss of power.
B. Correct. Local operation is directed by 1-AP-10.05. Att 4 provides direction.
C. Incorrect. Local operation is the only option after 30 minutes. Plausible because this is a viable option until the UPS runs out of power at about 30 minutes.
D. Incorrect. Local operation is directed by 1-AP-10.05. Att 4 provides direction.
C. Incorrect. Local operation is the only option after 30 minutes. Plausible because this is a viable option until the UPS runs out of power at about 30 minutes.

Supporting References
1-AP-10.05, Loss of Semi-Vital Bus, Rev. 25.

References Provided to Applicant
None.

Question History
Surry 2002 Exam

Answer: B
49. 000063G2.2.44 1

Unit 1 Initial Conditions:

- Power = 100%.

Current conditions:

- Reactor trips with no operator action.
- Turbine trips.
- Generator output breakers open, but the generator exciter field breaker does NOT open automatically when the turbine initially trips.
- In the absence of operator action, the exciter field eventually automatically secures on Volts/Hertz protection.
- #1 EDG automatically starts but the #1 EDG output breaker does NOT operate either automatically or remotely from the control panels.

Based upon the current conditions, which one of the following describes (1) what event caused the above indications, AND (2) the impact on breaker overcurrent protection?

A. (1) Loss of ‘A’ DC Bus.
(2) ‘A’ DC Bus 480V breakers AND ‘A’ DC Bus 4160V breakers will trip on overcurrent.

B. (1) Loss of ‘A’ DC Bus.
(2) ‘A’ DC Bus 480V breakers will trip on overcurrent. ‘A’ DC Bus 4160V breakers will NOT trip on overcurrent.

C. (1) Loss of ‘B’ DC Bus.
(2) ‘B’ DC Bus 480V breakers AND ‘B’ DC Bus 4160V breakers will trip on overcurrent.

D. (1) Loss of ‘B’ DC Bus.
(2) ‘B’ DC Bus 480V breakers will trip on overcurrent. ‘B’ DC Bus 4160V breakers will NOT trip on overcurrent.

K/A

063 DC Electrical Distribution
063G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
(CFR 41.5 / 43.5 / 45.12) (RO – 4.2)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly interpret the control room indications to determine whether an operationally valid scenario is a loss of ‘A’ DC Bus or a loss of ‘B’ DC Bus, and then demonstrate knowledge of how the casualty impacts overall emeregency breaker operations.

Answer Choice Analysis

A. INCORRECT. Part (1) of this distractor is correct. The indications that are given are exactly as written in Surry lesson plan ND-90.3-LP-6, “125 VDC DISTRIBUTION,” for “Expected plant response of a loss of ‘A’ DC bus while at power” (pages ND-90.3-AIA-6.2 page 1 and 2). Part (2) of this distractor is incorrect. The NOTE on page 2 of 9 in the lesson plan states: “the 4160V breakers do not have overcurrent protection. The 480V breakers overcurrent protection is mechanical (excessive current through an overcurrent coil pulls up on a tripper bar inside the breaker) so they will trip.”

B. CORRECT. See analysis of ‘A’ above.

C. INCORRECT. See analysis of ‘A’ above.

D. INCORRECT. See analysis of ‘A’ above.

Supporting References

1. Surry lesson plan ND-90.3-LP-6, “125VDC DISTRIBUTION,” rev. 16. Especially section for for “Expected plant response of a loss of ‘A’ DC bus while at power” (pages ND-90.3-AIA-6.2 page 1 and 2).

References Provided to Applicant

None.

Answer: B

50. 000063K1.02 2

Unit 1 Initial Conditions:

- Refueling outage is in progress.
- Reactor is completely de-fueled.
- Electrical systems from the 480V Emergency Buses to the Vital Buses are in a normal at-power line-up.

Current conditions:

- Time = 0215. Battery ‘1B’ is disconnected from DC Bus ‘1B’ for maintenance.
- Time = 0220. An electrical fault causes all sections of UPS ‘1B1’ to de-energize.

Based on the current conditions, with no operator actions taken, which one of the following is the status of Vital Buses as of Time = 0221 ?

	Vital Buses 1-II and 1-IIA	Vital Buses 1-IV and 1-IVA
A.	De-energized	De-energized
B.	Energized	De-energized
C.	Energized	Energized
D.	De-energized	Energized

K/A

063 DC Electrical Distribution

063K1.02 Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system.

(CFR 41.2 to 41.9 / 45.7 to 45.8) (RO – 2.7)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to recall in a straight-forward fashion the one-line diagram from the "VITAL AND EMERGENCY DISTRIBUTION," and to correctly determine which vital AC buses would lose power based on operationally valid problems with the DC electrical system.

Answer Choice Analysis

A. INCORRECT. Based on the provided diagram ND-90.3-H/T-5.2, "VITAL AND EMERGENCY DISTRIBUTION," when Battery 1B, DC Bus 1B, and UPS 1B1 lose power, Vital Buses 1-II and 1-IIA will lose power. Vital Bus 1-III and 1-IIIA will remain energized from UPS 1A2, and Vital Bus 1-IV and 1-IVA will remain energized by UPS 1B2 via MCC 1J1-1 or MCC 1J1-2. All other choices are plausible combinations of these buses.

B. INCORRECT. See analysis of 'A' above.

C. INCORRECT. See analysis of 'A' above.

D. CORRECT. See analysis of 'A' above.

Supporting References

1. Surry Lesson Plan ND-90.3-LP-5, "VITAL AND SEMI-VITAL BUS DISTRIBUTION," rev. 15. Especially the diagram that is provided as handout 5.2, "VITAL AND EMERGENCY DISTRIBUTION."

References Provided to Applicant

None.

Answer: D
51. 000064K6.08 2

Current conditions:

- Unit 1 has experienced a loss of offsite power.
- #1 EDG has started but failed to load.
- A field operator reports the exciter field circuit breaker on #1 EDG is tripped.

Based on the current conditions, which one of the following describes:

- the impact this will have on the ability to transfer fuel oil to the #1 EDG
  and
- the minimum volume necessary to meet the Technical Specification Day Tank requirements?

The fuel oil system will no longer be able to transfer fuel to the______(1)_______.
The Day Tank volume must be greater than or equal to _____(2)___.

A. (1) Base Tank, but will continue to transfer fuel to the Wall Tank.
   (2) 290 gallons.

B. (1) Base Tank, but will continue to transfer fuel to the Wall Tank.
   (2) 375 gallons.

C. (1) Base Tank and it will be unable to transfer fuel to the Wall Tank.
   (2) 375 gallons.

D. (1) Base Tank and it will be unable to transfer fuel to the Wall Tank.
   (2) 290 gallons.

<u>K/A</u>

Emergency Diesel Generator
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks
(CFR: 41.7/45.7) (RO – 3.2)
<u>K/A Match Analysis</u>

The RO applicant is required to recognize that a major portion of the Fuel Storage volume will be lost and that only the volume of the Base Tank will be available to the EDG. In addition, the RO applicant is required to know that the Technical Specification minimum Day Tank volume is based on the combined volumes of the Wall and Base Tanks.

<u>Answer Choice Analysis</u>

A. INCORRECT. *Plausible because the pumps that transfer fuel from the Wall Tank to the Base Tank are powered by the EDG field. So with exciter field breaker open power would not be available to these pumps. However, without an exciter field the EDG will not load. The fuel transfer pumps from the underground storage tank are fed from the emergency bus (H-1) supplied by the EDG. The system would be unable to replenish either the Wall Tank or the Base Tank. Also, the T.S. Day Tank volume requirement is 290 gallons.*

B. INCORRECT. *Plausible because the pumps that transfer fuel from the Wall Tank* *to the Base Tank are powered by the EDG field. So with exciter field breaker open power would not be available to these pumps. However, without an exciter field the EDG will not load. The fuel transfer pumps from the underground storage tank are fed from the emergency bus (H-1) supplied by the EDG. The system would be unable to replenish either the Wall Tank or the Base Tank. The second half of the question is plausible because that is the low level auto-start signal for the fuel transfer pumps to the Base Tank.*

C. INCORRECT. *The first half of the answer is correct. The second half of the question is plausible because that is the low level auto-start signal for the fuel transfer pumps to the Base Tank.*

D. CORRECT. See the discussion for choice A above.

<u>Supporting References</u>

1. Surry lesson plan ND-90.3-LP-1, "Emergency Diesel Generator," rev. 20, Objectives B & M , pgs. 9, 47.

2. Technical Specifications, 3.16

<u>References Provided to Applicant</u>

None.

Answer: D
52. 000073G2.2.22 2

Unit 1 plant conditions:
Refueling is in progress
The containment gas radiation monitor (1-RI-RM-160) loses power

Based on the above conditions, which one of the following states the actions required by Technical Specifications / Technical Requirements Manual?

A. Fuel movement must be suspended until the automatic actions of the radiation monitor are verified, then fuel movement may resume.

B. Fuel movement may continue provided that the containment particulate and manipulator crane radiation monitors (1-RI-RM-159/162) are operable.

C. Fuel movement is allowed to continue for up to one hour while repair attempts are made on the detector.

D. Fuel movement must be suspended immediately and no operations which increase the reactivity of the core shall be made.

<u>K/A</u>
Process Radiation monitoring: G2.2.22 Process Radiation Monitoring System: Knowledge of limiting conditions for operations and safety limits.

<u>K/A Match Analysis</u>
Requires applicant to know how various Rad Monitors will indicate when exceeding plant limits.

<u>Answer Choice Analysis</u>

A. Incorrect: Once the automatic actions are completed, the affected components are verified in their fail safe condition. The candidate may come to the conclusion that with the RMs in the fail safe condition TS do not require the RM to be operable.
B. Incorrect: Plausible because there are other means to measure containment radiation (refer to TS 3.1, must have at least two means to detect containment radiation) and the candidate may confuse TS 3.1 requirements with TS 3.10.
C. Incorrect: Plausible because two detectors are still operable and it allows the operator time to place fuel in a safe location.
D. Correct: Per TR 3.3.3: A.1 As specified in TS 3.10.C, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease, work shall be initiated to correct the conditions so that TS 3.10.A.3 or TS 3.10.B.1 are met, and no operations which increase the reactivity of the core shall be made.

<u>Supporting References</u>
TS 3.10.B.1 Irradiated Fuel Movement
TRM 3.3.3 Radiation Monitor Requirements

<u>References Provided to Applicant</u>
none

Answer: D
53. 000076A3.02 1

Current conditions:

- Unit 1 has experienced a large break LOCA.
- Containment pressure peaked at 24.5 psia.

Based on the above current conditions, which one of the following components will have their supporting Service Water equipment repositioned from the equipment's original position before the LOCA?

A. Recirculation Spray heat exchangers.

B. Component Cooling Water heat exchangers.

C. Bearing Cooling Water heat exchangers.

D. Charging Pump oil coolers.

<u>K/A</u>

Service Water
Ability to monitor automatic operation of the SWS, including: Emergency heat loads.
(CFR: 41.7/45.5) (RO – 3.7)

<u>K/A Match Analysis</u>

The RO applicant is required to recognize the systems that will automatically realign as a result of a Hi-Hi Containment condition.

<u>Answer Choice Analysis</u>

A. CORRECT. A Hi-Hi CLS signal will cause 1-MOV-103, -104 and -105 valves on each Recirculation Spray heat exchangers to open to allow Service Water flow to the heat exchangers.

B. INCORRECT. *Plausible because the inlet and outlet SW valves on the CC heat exchangers close on Hi-Hi CLS signal, but only in coincidence with a blackout (LOOP) signal.*

C. INCORRECT. *Plausible because the supply SW valves on the Bearing Cooling heat exchangers close on Hi-Hi CLS signal, but only in coincidence with a blackout (LOOP) signal.*

D. INCORRECT. *Plausible because the Charging Pump Service Water System does receive an automatic signal, but it results from a low discharge pressure signal, which will autostart the standby pump.*

<u>Supporting References</u>

1. Surry lesson plan ND-89.5-LP-2, "Service Water System," rev. 27, Obj F, pp. 15, 18-19.

2References Provided to Applicant

None.

Answer: A
54. 000078K4.02 2

Unit 1 initial plant conditions:
- Reactor is at Hot Shutdown conditions
- The Containment Instrument Air Compressors are non-functional.

Based on the above conditions, which one of the following states (1)operator action that will be required to allow air to be aligned to containment (2) if the operator stationed at the containment isolation valve(s) is required to assume adminstrative control per TS 3.8 Containment?

A. (1) Open 1-CP-FIC-101 (2) No

B. (1) Open 1-CP-FIC-101 (2) Yes

C. (1) Open 1-IA-446 and 1-IA-447 (2) No

D. (1) Open 1-IA-446 and 1-IA-447 (2) Yes

K/A
Instrument Air. Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to the other air systems.

K/A Match Analysis
Requires applicant to know how the Instrument Air system connects to the Containment Air System.

Answer Choice Analysis

A. Incorrect: Instrument Air is the backup to the Containment Instrument Air system. Plausible because the Service Air system backs up the Instrument Air system. 2nd part is plausible because TS 3.8 also requires the operator to be at the containment isolation valve which the applicant may think is sufficient.

B. Incorrect: Instrument Air is the backup to the Containment Instrument Air system. Plausible because the Service Air system backs up the Instrument Air system. 2nd part is correct per 3.8 Containment.

C. Incorrect: 1st part is correct. Instrument Air is the backup to the Containment Instrument Air system. 2nd part is plausible because TS 3.8 also requires the operator to be at the containment isolation valve which the applicant may think is sufficient.

D. Correct: Instrument Air is the backup to the Containment Instrument Air system. 2nd part is correct per 3.8 Containment: The opening of manual or deactivated automatic containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls.

Supporting References
ND-92.1-LP-1 Obj E
TS 3.8

References Provided to Applicant
none

Answer: D
55. 000103K3.01 2

Unit one is at hot shutdown in preparation for a refueling shutdown.

A team is heading to containment to perform the hot shutdown walkdown.

After opening the outer door the team leader identifies air flow noise coming from the inner door. The team leader verified that the inner door was closed fully. The entry was cancelled and the outer door was closed.

Which ONE of the following states the required actions in response to this event in accordance with Technical Specification section 3.8 (Containment)?

A. The personnel hatch outer door may be opened for a period not to exceed 15 minutes for repair and retest of the inner door seal.

B. The personnel hatch outer door must remain closed. Any repair attempts must be made using the equipment hatch escape hatch.

C. The personnel hatch outer door may be opened and a 1 hour clock initiated due to loss of containment integrity.

D. The unit is in a condition that containment integrity is no longer required. Repairs may commence immediately.

K/A
Containment System: Knowledge of the effect that a loss of malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions.

K/A Match Analysis
Requires applicant to know how a containment isolation valve affects containment integrity and what actions are required.

Answer Choice Analysis

A. Correct.
B. Incorrect: TS 3.8. Plausible if the candidate does not recall that TS allow for a 15 minute time period to repair the inner door. Canidate may believe that the outer hatch must be closed to keep integrity intact.
C. Incorrect: Plausible if the candidate confuses the containment integrity spec with the containment door spec.
D. Incorrect: Plausible since the unit is in hot shutdown and the candidate may not know that the containment integrity spec is a CSD spec vice a HSD spec.

Supporting References
TS 1.0-1 Definitions
TS 3.8 Containment

References Provided to Applicant
none

Licensee to verify that a mode change is not allowed while in an LCO.

Answer: A
56. 002K1.01 1

Unit 1 Current Conditions:

- Reactor Vessel Disassembly is in progress, and the Refueling team is ready to 'flood up' the Reactor Cavity.
- Testing of Safety Injection check valves is NOT required.

Based on the current conditions, which one of the following completes the below statements in accordance with 1-OP-FH-001, "CONTROLLING PROCEDURE FOR REFUELING?"

(1) The preferred method for cavity fill is from the Refueling Water Storage Tank

(RWST) through the Reactor Coolant System ___________________ .

(2) When the cavity is filled to between 26 and 27 feet, the final **minimum** required RWST level is ___________________ .

A. (1) cold legs
(2) 22%

B. (1) cold legs
(2) 12%

C. (1) hot legs
(2) 22%

D. (1) hot legs
(2) 12%

K/A

002 Reactor Coolant
002K1.01 Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems: RWST.
(CFR 41.2 to 41.9) (RO – 3.7)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to remember in a straight-forward fashion that it is preferred to fill the cavity via the hot legs to prevent introduction of crud into the cavity; also to remember that the RWST is checked at 85% level to ensure a minimum 22% level when the cavity is full.

Answer Choice Analysis

A. INCORRECT. Part (1) of the distractor is incorrect. The Surry lesson plan for “Unit Refueling Overview” states the following: “Cavity fill is normally done through the safety injection hot legs (high head or low head). Fill using the cold legs is permitted only to test the check valves. If the cavity is filled using the cold legs crud is flushed from the reactor vessel reducing water clarity and raising radiation levels.” Part (1), using cold legs, is plausible if the applicant mis-remembers the reasons for the preferred flow path, or if the applicant wrongly missed the bullet in the question stem that stated check valve testing was NOT to be performed. Part (2) of this distractor is correct. The RWST level is filled to at least 85% before cavity fill, in order to ensure at least 22% RWST level when the cavity fill operation is secured.

B. INCORRECT. Parts (1) and (2) incorrect. See analysis of ‘A’ above. The answer for part (2) is 12% RWST level, which is a plausible wrong misconception given that the correct part (2) answer is 22%.

C. CORRECT. See analysis of ‘A’ above.

D. INCORRECT. Part (1) correct, part (2) incorrect. See analysis of ‘A’ and 'B' above.

Supporting References

1. Surry Lesson Plan ND-92.5-LP-1, “UNIT REFUELING OVERVIEW,” rev. 18.

2. Surry Procedure 1-OP-FH-001, “CONTROLLING PROCEDURE FOR REFUELING,” rev. 23.

References Provided to Applicant

None.

Answer: C

57. 011A4.04 2

Initial conditions:

- Unit 1 is at 100% power.
- The PRZR LVL – CH SEL Switch is in the “CH 3 & 2” position.
- Pressurizer backup heater bank ‘B’ and bank 'C' proportional heaters are energized.

Current conditions:

- Pressurizer level transmitter 1-RC-LT-1461 (PRZR LEVEL PROTECT CH 3) fails high.

Based on current conditions, which one of the following statements describes the effect on pressurizer level control and on the pressurizer heaters?

The pressurizer level control system …

A. will continue to maintain pressurizer level at program level and the remaining backup heater banks will energize.

B. is required in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls, to be placed in manual to adjust pressurizer level back to the program level setting and the remaining backup heater banks will energize.

C. will continue to maintain pressurizer level at program level, however pressurizer backup heater bank ‘B’ will de-energize.

D. is required in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls, to be placed in manual to adjust pressurizer level back to the program level setting, however pressurizer backup heater bank ‘B’ will de-energize.

K/A

Pressurizer Level Control
Ability to manually operate and/or monitor in the control room: Transfer of PZR LCS from automatic to manual.
(CFR: 41.7/45.5 to 45.8) (RO – 3.2)

K/A Match Analysis

The RO applicant is required to apply the knowledge that the selected level transmitter (CH III) to the upper control channel will provide the input to the pressurizer level control system and to the pressurizer heaters to energize on 5% level deviation signal. An upscale failure of Channel III pressurizer level transmitter, which corresponds to 1-RC-LT-1461, will require operator action to place the pressurizer level control system in manual to control pressurizer level and will result in energizing the pressurizer heaters.

Answer Choice Analysis

A. INCORRECT. *Plausible because the pressurizer level control system would continue to automatically control pressurizer level if the failed transmitter were assigned to the lower control channel. In addition, the lower control channel signal also provides a signal to the pressurizer heaters, however, it will cause the heaters to de-energize on low pressurizer level rather than energize on a deviation of pressurizer level above program level.*

B. CORRECT. Since Channel 3 (1-RC-LT-1461) is selected to the upper control channel then an upscale failure of this transmitter will cause the pressurizer level controller to reduce the demand signal to the charging system. The pressurizer level controller will need to be placed in manual to control pressurizer level. Also, the level transmitter selected to the upper control channel provides the signal to the pressurizer heaters to energize when pressurizer level increases 5% above program level.

C. INCORRECT. *Plausible because the pressurizer level control system would continue to automatically control pressurizer level if the failed transmitter were assigned to the lower control channel. In addition, the lower control channel does not provide a* *signal to the pressurizer heaters on a 5% level deviation above program level.*

D. INCORRECT. *Plausible because the first half of the response is correct. The second half is plausible if a misconception exists as to which control channel provides the signal to energize the pressurizer heaters.*

Supporting References

1. Surry lesson plan ND-93.3-LP-7, "Pressurizer Level Control System," rev. 10, Objectives B and C, pp. 5, 9.

2. 0-AP-53, Loss of Vital Instrumentation/Controls, rev. 15, pg. 10.

References Provided to Applicant

None.

Answer: B
58. 014K5.02 2

Unit 2 Initial Conditions (0800 hours):
- Startup and power escalation are in progress.
- Reactor power is 25%.
- All control rods bank demand positions have remained aligned within 6 steps of their CERPI indication.

Unit 2 Current Conditions (0830 hours):
- Reactor power is 35% when the CERPI indication for control rod F8 drifts 14 steps lower than its bank demand position

Based on current conditions, which one of the following describes (1) whether the conditions of Tech Spec LCO 3.12.E, ROD POSITION INDICATION SYSTEM AND BANK DEMAND POSITION INDICATION SYSTEM, are currently met, and (2) required actions, if any, in accordance with 0-AP-1.01, CONTROL ROD MISALIGNMENT?

A. (1) CERPI F8 is currently operable.
(2) Shutdown margin requirements ARE required to be verified within one hour.

B. (1) CERPI F8 is currently operable.
(2) Shutdown margin requirements ARE NOT required to be verified within one hour.

C. (1) CERPI F8 is currently inoperable.
(2) Shutdown margin requirements ARE required to be verified within one hour.

D. (1) CERPI F8 is currently inoperable.
(2) Shutdown margin requirements ARE NOT required to be verified within one hour.

K/A: 014K5.02
Rod Position Indication
Knowledge of the operational implications of the following concepts as they apply to RPIS: RPIS independent of demand position.

KA MATCH ANALYSIS:
Knowledge of the operational implications of the misaligned rod is required to arrive at the correct answer. The applicant must know that the rod can be misaligned for up to one hour before the LCO is not met.

ANSWER CHOICE ANALYSIS:
A. Incorrect. The first part is correct. The second part is plausible because SDM verification is step 6 in the AP, but step 6 is skipped over when a rod has not dropped.
B. Correct. IAW TS 3.12.E, IRPI and DRPI may deviate by 24 steps for up to one hour each day. Since the deviation has existed for a max of 30 minutes, the conditions of the LCO are still met and will be for at least another half hour. SDM verification is not required IAW the procedure because the rod has not dropped.
C. Incorrect. See above.
D. Incorrect. See above.

REFERENCES TO BE PROVIDED TO THE APPLICANT: None

REFERENCES:
1. Tech Spec 3.12.E, ROD POSITION INDICATION SYSTEM AND BANK DEMAND POSITION INDICATION SYSTEM.
2. 0-AP-1.01, CONTROL ROD MISALIGNMENT, Rev. 20.

Answer: B
59. 015K2.01 2

Which one of the following completes the below statements?

(1) The normal power supply to NFI-NM-190A/B, the remote (local) Unit 1 Excore Neutron Flux Monitor (Excore Fission Chamber), is ____________________ .

(2) In the event of a loss of power on Unit 1, the power supply for NFI-NM-190A/B may be aligned to Unit 2 using a transfer switch located in
________________________ .

A. (1) Vital Bus 1-I
(2) the Unit 1 Cable Tray Room

B. (1) Vital Bus 1-I
(2) the Unit 2 Emergency Switchgear Room (ESGR)

C. (1) Vital Bus 1-II
(2) the Unit 1 Cable Tray Room

D. (1) Vital Bus 1-II
(2) the Unit 2 Emergency Switchgear Room (ESGR)

<u>K/A</u>

015 Nuclear Instrumentation System
015K2.01 Knowledge of bus power supplies to the following: NIS channels, components, and interconnections.
(CFR 41.7) (RO – 3.3)

<u>K/A Match Analysis</u>

This question matches the K/A statement by requiring the RO applicant in a straightforward fashion recall the Appendix R electrical distribution system that powers the remote indication Nuclear Instrument system.

<u>Answer Choice Analysis</u>

A. INCORRECT. Part (1) of the distractor is correct; the normal power supply to the remote (Channel I) excore neutron flux monitor NFI-NM-190A/B is Vital Bus 1-I. Part (2) of the distractor is incorrect; the transfer switch is located in the Unit 2 ESGR. Part (2) is plausible because there is another transfer switch located in the Unit 1 Cable Tray room, which is used to transfer power to a LFFG portable generator (Large Fire or Flooding Generator).

B. CORRECT. See analysis of ‘A’ above.

C. INCORRECT. Part (1) of this distractor is incorrect. However, Vital Bus 1-II is plausible because it is the normal power supply to the channel 2 excore neutron flux monitor, which is only read in the MCR (not remotely for Appendix R purposes). Part (2) is also incorrect, see analysis of ‘A’ above.

D. INCORRECT. Part (1) is incorrect, part (2) correct.

<u>Supporting References</u>

1. Surry Lesson Plan ND-93.2-LP-5, “EXCORE FISSION CHAMBER SYSTEM,” rev.
5. Especially page ND-93.2-H/T-5.6, “APPENDIX R POWER SUPPLY.”

References Provided to Applicant

None.

Answer: B

60. 029K3.01 2

Unit 1 is shutdown for refueling with conditions as follows:

- Core off load to the fuel building is in progress
- 1-FH-1 (transfer canal gate valve) is open
- Containment Purge is in service
- One containment air recirc fan is in service
- All other containment ventilation fans are secured
- The equipment hatch is closed
- Operators are maintaining one door of the personnel hatch closed.
- Outside air is aligned to containment and the containment instrument air compressors are secured.

The containment purge **supply** MOVs inadvertantly CLOSE while the containment purge exhaust MOVs remain OPEN. No other changes to the purge alignment have occured.

Which ONE of the following states 1) the expected response to containment parameters to this event, and 2) the impact of this event on Spent Fuel Pit Level?

A. 1) Unit 1 containment pressure increases.
   2) Spent Fuel Pit Level increases

B. 1) Unit 1 containment pressure decreases.
   2) Spent Fuel Pit Level decreases

C. 1) Unit 1 containment pressure decreases.
   2) Spent Fuel Pit Level increases

D. 1) Unit 1 containment pressure increases.
   2) Spent Fuel Pit Level decreases

K/A

Containment Purge: Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: Containment parameters.

K/A Match Analysis

Requires knowledge of how system failure will affect containment pressure.

Answer Choice Analysis

A. INCORRECT. Part 1 is incorrect, but plausible if the candidate believes that purge isolation simply "bottles-up" containment. Part 2 is plausbile if the candidate believes that without purge, containment and subsequently the refueling cavity will heat up and cause level to increase.
B. CORRECT.
C. INCORRECT. Part 1 is correct. Part 2 is plausbile if the candidate believes that without purge, containment and subsequently the refueling cavity will heat up and cause level to increase.
D. INCORRECT. Part 1 is incorrect, but plausible if the candidate believes that purge isolation simply "bottles-up" containment. Part 2 is correct, but plausible if the candidate thinks that with containment "bottled up" and the fuel building under suction that level will still decrease.

Supporting References

ND-88.4-LP-6 Obj D, Revision 9

References Provided to Applicant

None.
Licensee to verify this question based on what containment pressure is maintained at when containment purge is operating.

Answer: B
61. 034A2.01 1

Initial conditions:
- Unit 1 is in REFUELING
- Fuel off-loading is in progress.

Current conditions:
- A report is received that a spent fuel assembly being removed from the core has slipped from the manipulator crane and is lying on the lower core plate.
- Bubbles were seen escaping the dropped fuel assembly.
- No radiation alarms have been received from the radiation monitors associated with the manipulator crane or the refueling area.
- HP is monitoring the area, but report normal readings around the reactor cavity area.

Based on current conditions, which one of the following describes the actions required by 0-AP-22.00, Fuel Handling Abnormal Conditions?

A. Stop fuel handling operations and evacuate Containment. Fuel Building evacuation is not required.

B. Stop fuel handling operations, Notify Shift Supervision, OMOC and the Shift Technical Advisor. Area evacuations are not required at this time.

C. Stop fuel handling operations and evacuate Containment and the Fuel Building.

D. Stop fuel handling operations, Notify Shift Supervision, Fuel Performance Analysis and Health Physics. Area evacuations are not required at this time.

K/A

Fuel Handling Equipment
Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element.
(CFR: 41.5/43.5/45.3/45.13) (RO – 3.6)

K/A Match Analysis

The RO applicant is required recognize the impact from a dropped fuel assembly with normal radiation levels results in termination of fuel handling operations and evacuation of the containment based on the actions contained in 0-AP-22.00, Fuel Handling Abnormal Conditions.

Answer Choice Analysis

AA. CORRECT. These are the actions that would be performed as a result of the dropped assembly. Fuel building would not require evacuation since the dropped assembly occurred inside containment.

B. INCORRECT. *Plausible because with no detectable radiation the applicant may think that evacuation is not required. Following a dropped fuel assembly, step 2 of 0-AP-22.00 states "Check local radiation levels – Normal." With normal radiation levels the user continues on to step 3 which directs the user to step 20 to make the listed notifications and then exit the procedure. However, the user would never get to step 2 because the RNO actions for step 1 (Check for fuel repairs) sends the user to step 4.0 which directs stopping fuel handling operations.*

C. INCORRECT. *Plausible because evacuation of both areas (step 5) is addressed by 0-AP-22.00. However, it is an OR statement rather than an AND statement.*

D. INCORRECT. *Plausible because with no detectable radiation the applicant may think that evacuation is not required. The second sentence is plausible because the precautions and limitations (4.3) section of 1-OP-FH-001, Controlling Procedure for Refueling, states that the Refueling SRO (Shift Supervision) and Fuel Performance Analysis will be notified immediately anytime fuel assembly damage is observed or expected. However, these positions are not listed in 0-AP-22.00 as requiring notification. In addition, the Health Physics department is notified only if abnormal radiation levels are observed.*

<u>Supporting References</u>

1. Surry lesson plan ND-92.5-LP-7, "Refueling APs," rev. 13, Objective C, pp. 9-10.

2. 0-AP-22.00, Fueling Handling Abnormal Conditions, Rev. 23, pgs. 2 and 6.

<u>References Provided to Applicant</u>

None.

Answer: A
62. 041K6.03 4

Unit 1 Initial Conditions:

- Power = 100%.
- 1G-A6 ROD CONTROL SYSTEM URGENT FAILURE is in alarm.
- I&C has determined that the fault is in the logic cabinets
- Rod control is in MANUAL.

Current Conditions:

- Main Turbine control valves unexpectedly throttle closed, resulting in a 12% load rejection over a 90 second period.

Based on the current conditions, which one of the following describes the effect on Reactor Coolant System (RCS) temperature with **<u>no</u>** operator action?

A. RCS temperature will change based on the higher Xenon concentration, with no manual rod motion possible and no steam dump operation.

B. RCS temperature will be controlled entirely by manual use of control rods. No steam dumps will operate.

C. RCS temperature will be controlled entirely by steam dump operation. No manual control rod motion is possible.

D. RCS temperature will be controlled with a combination of manual control rod motion and steam dump operation.

K/A

041 Steam Dump System (SDS)/Turbine Bypass Control
041K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the Steam Dump System: Controller and Positioners, including ICS, S/G, CRDS.
(CFR 41.7 / 45.7) (RO – 2.7)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly assess the effect of a rod control malfunction (CRDS) on the Steam Dump System and the control of RCS temperature, given an operationally valid/plausible situation.

Answer Choice Analysis

A. INCORRECT. Steam dumps will operate as stated in correct answer. Plausible because rods will not operate and applicant may not make the connection that steam dumps are armed with a demand signal.

B. INCORRECT. Control Rods will not move due to the urgent failure. Plausible if applicant does not know that the alarm is indication that rods will not move because rods are capable of handling 10% load rejects without the help of the steam dumps.

C. CORRECT. Rods will not move due to the urgent failure. Steam dumps will arm with a 10% rejection in less than 2 minutes. Therefore, with an arming signal and a Tave-Tref deviation (hence a demand signal), steam dumps will open.

D. INCORRECT. Rods will not move due to the urgent failure. Plausible if applicant does not know that the alarm is indication that rods will not move.

Supporting References

1. Slightly modified from Vogtle 2005-301 ILO exam question 041K6.10, still counting as ‘BANK’ question.

2. Surry Lesson Plan ND-93.3-LP-9, “STEAM DUMPS,” rev. 13.

3. Surry Lesson Plan ND-93.3-LP-3, “ROD CONTROL SYSTEM,” rev. 19.

References Provided to Applicant

None.

Answer: C

63. 045A1.05 2

Unit 1 initial conditions:

- Reactor shutdown in progress
- Reactor power = 8%
- Turbine Trip occurs
- Main Steam Dump Valves fail closed

Based on the above conditions, which one of the following states (1) if the Pressurizer PORV(s) will open during the initial transient (1st minute) and (2) the Tcold at which the RCS will stabilize?

A. (1) Yes
(2) 550 ºF

B. (1) Yes
(2) 556 ºF

C. (1) No
(2) 550 ºF

D. (1) No
(2) 556 ºF

K/A

Main Turbine Generator: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of primary plant parameters (temperature and pressure) following T/G trip.

K/A Match Analysis

Requires applicant to know how primary system pressure and temperature will respond to a turbine trip.

Answer Choice Analysis

A. Incorrect: 1^{st} part is plausible in that the normal secondary heat sink has been lost with the reactor still at power. 2^{nd} part is correct.

B. Incorrect: 1^{st} part is plausible in that the normal secondary heat sink has been lost with the reactor still at power. With the main steam dumps failed closed, RCS Tcold will stabilize at the saturation temperature of the SGs. The SG pressure will be maintained by the SG PORVs at1035 psig --- 1050 psia --- 550 ^{0}F Tsat.

C. Correct: The Pzr spary valves are designed to prevent lifting the Pzr Porv during a 10% step decrease in power. The 3 SG PORVs have a capacity of ~ 1.1 E6 lbm/hr. With the main steam dumps failed closed, RCS Tcold will stabilize at the saturation temperature of the SGs. The SG pressure will be maintained by the SG PORVs at1035 psig --- 1050 psia --- 550 ^{0}F Tsat.

D. Incorrect: 1^{st} part is correct. 2^{nd} part is plausible because if the steam demand were to exceed the capacity of the SG PORVs, it could be correct based on the lowest SG Safety setpt of 1085 psig --- 1100 psia --- 556 ^{0}F Tsat.

<u>Supporting References</u>
ND-89.1-LP-2
ND-88.1-LP-3

<u>References Provided to Applicant</u>
none

Licensee to verify RCS temperature and Pressurizer abbreviation

Answer: C
64. 071K4.01 1

Which one of the following corresponds to the pressure setting of the relief valve on the inner tank of the Waste Gas Decay Tanks?

A. 75 psig.

B. 100 psig.

C. 115 psig.

D. 150 psig.

<u>K/A</u>

Waste Gas Disposal

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Pressure capability of the waste gas decay tank.
(CFR: 41.7) (RO – 2.6)

K/A Match Analysis

The RO applicant is required to know the pressure the inner tank is capable of withstanding before its contents are released to the surrounding environment via the relief valve.

Answer Choice Analysis

A. INCORRECT. *Plausible because this is the alarm point for annunciator 0-WD-B9, WASTE GAS DECAY TKS RELIEF VVS HI PRESS. The alarm is sensing a high pressure condition between the relief valve and the downstream rupture disk.*

B. INCORRECT. *Plausible because this is the pressure setting for the relief valve on the outer tank of the Waste Gas Decay Tanks*.

C. INCORRECT. *Plausible because this is the maximum allowable pressure in accordance with logs.*

D. CORRECT. The relief valve on the inner tank of the Waste Gas Decay Tanks is set to lift at 150 psig.

Supporting References

1. Surry lesson plan ND-92.4-LP-1, "Gaseous and Liquid Waste Processing Systems," rev. 13, Obj. B, pp. 13-14.

References Provided to Applicant

None.

Answer: D
65. 072G2.2.40 1

Unit 2 is about to begin refueling operations (movement of irradiated fuel and water level is >23 feet). Per T.S. 3.10, Refueling, which ONE of the following will prevent refueling operations from commencing?

A. The 1A RHR pump was declared inoperable 10 hours ago, both RHR HXs are available.

B. The Manipulator Crane Area Rad. Monitors have failed and containment purge is isolated.

C. The inner door of the personnel airlock cannot be closed.

D. RCS boron concentration is 2360 ppm as verified by a sample taken 24 hours ago.

K/A

Area Radiation Monitoring / Area Radiation Monitoring: Ability to apply Technical Specifications for a system.

K/A Match Analysis

Requires knowledge of TS and how to apply statements of applicability.

Answer Choice Analysis

A. INCORRECT. T.S. 3.10 only requires one RHR pump and heat exchanger to be OPERABLE as long as water level is >23 feet.

B. CORRECT. T.S. 3.10 states the Manipulator Crane Area Rad. Monitors must be OPERABLE and continuously monitored.

C. INCORRECT. T. S. 3.10 states the equipment access hatch and at least one door in the personnel airlock shall be capable of being closed.

D. INCORRECT. T.S. 3.10 requires boron to be greater than 2300 ppm and sampled every 72 hours.

Supporting References

T.S. 3.10.A.4

ND-92.5-LP-1, Obj. D

References Provided to Applicant

None.

Answer: B
66. G2.1.25 2

Unit 1 Current conditions:

- Reactor has been shutdown for 5 days.
- RCS water level is being maintained at 7.0 inches as indicated 1-RC-LR-105 (Cold Shutdown RCS Level Narrow Range).
- The ‘B’ and ‘C’ loops are isolated.
- Steam generator (SG) primary manways have been removed in preparation for SG tube plugging efforts.
- The reactor vessel head is tensioned.
- ‘A’ RHR pump is in operation with oscillating amperage indications
- Flow indication on 1-RH-FI-1605 is oscillating between 3,300 and 3,700 gpm.

Based on the current conditions, which one of the following describes the proper adjustments of RCS level and RHR flow in accordance with 1-AP-27.00, Loss of Decay Heat Removal Capability?

(REFERENCE PROVIDED)

A. Raise RCS level to 9.0 inches as indicated on 1-RC-LR-105 and reduce flow to 3,100 gpm.

B. Maintain current RCS level as indicated on 1-RC-LR-105 and reduce flow to 1,500 gpm.

C. Maintain current RCS level as indicated on 1-RC-LR-105 and reduce flow to 3,100 gpm.

D. Raise RCS level to 9.0 inches as indicated on 1-RC-LR-105 and stabilize flow at 3,500 gpm.

<u>K/A</u>

Plant Wide Generics
Ability to obtain and interpret station reference materials such as graphs, monographs, and tables, which contain performance data.
(CFR: 41.10/43.5/45.12) (RO – 3.9)

<u>K/A Match Analysis</u>

The RO applicant must recognize that the RHR pump is vortexing and use the graphs on Attachment 1, RHR Flow Requirements Versus Time after Shutdown and Attachment 3, Minimum RCS Level Versus RHR Flow (1-RC-LR-105) to determine the RHR flow and RCS level that will stop pump vortexing.

Answer Choice Analysis

A. CORRECT. Per Attachment 1, a flow rate of 3,100 gpm satisfies the minimum RHR flow for the five days (120 hrs) the reactor has been shutdown. Per Attachment 3, a flow rate of 3,100 gpm requires a minimum RCS level of ~8.6 feet. With RCS level at 9.0 feet and RHR flow at 3,100 gpm ends up in the acceptable region of the graph in Attachment 3.

B. INCORRECT. *Plausible because the conditions of RCS level at 7.0 feet and RHR flow at 1,500 gpm ends up in the acceptable region of the Attachment 3 graph. However, the RHR flow is not adequate to satisfy the minimum flow requirements for Attachment 1 for the given shutdown period of five days.* .

C. INCORRECT. *Plausible because a flow 3,100 gpm will satisfy the flow requirements for Attachment 1 for the given shutdown period of five days. However, the RCS level of 7.0 feet does not provide adequate NPSH at 3,100 gpm flow to prevent vortexing per Attachment 3.*

D. INCORRECT. *Plausible because a flow 3,500 gpm will satisfy the flow requirements for Attachment 1 for the given shutdown period of five days. However, the RCS level of 9.0 feet does not provide adequate NPSH at 3,500 gpm flow to prevent vortexing per Attachment 3.*

Supporting References

1. Surry lesson plan ND-88.2-LP-3, "Draindown and Mid-Loop Operations," rev. 16, Objective C, pp. 20-23.

2. 1-AP-27.00, "Loss of Decay Heat Removal Capability", Rev. 20

3. This question was used on the Surry 2004-301 NRC exam. Modified conditions slightly and re-organized distractors, but question is essentially the same.

References Provided to Applicant

Attachments 1 and 3 of 1-AP-27.00

Answer: A
67. G2.1.37 2

Which one of the following statements lists the required items to be in a Reactivity Plan for a ramp up in power in accordance with GOP-1.5, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER?

A. Delta flux control, expected xenon transient, and recommendations for rod height and/or RCS boron adjustments.

B. Limitations on startup rate, expected xenon transient, and recommendations for rod height and/or RCS boron adjustments.

C. Delta flux control, expected xenon transient, and RCS temperature control.

D. Delta flux control, expected source range counts at the doubling points, and recommendations for rod height and/or RCS boron adjustments.

K/A G2.1.37
Knowledge of procedures, guidelines, or limitations associated with reactivity management?

KA MATCH ANALYSIS:
The above question requires knowledge of the components of a Reactivity Plan, which is testing knowledge of reactivity management guidelines.

ANSWER CHOICE ANALYSIS:
A. Correct. See list from GOP-1.4
B. Incorrect. The only incorrect item is the limitations on SUR. Plausible because there are limits on SUR that likely would be included in a reactivity brief.
C. Incorrect. The only incorrect item is RCS temperature. Plausible because there are limitations on RCS temperature that likely would be included in a reactivity brief.
D. Incorrect. The only incorrect item is SR counts at doubling points. This too is plausible because it is a likely component for a reactivity brief for performing this procedure.

REEFERENCES TO BE SUPPLIED TO APPLICANT: None

Memory Level / Lower Cog / Fundamental

LOD = 2

New Question

REFERENCS:
GOP-1.4, UNIT STARTUP, HSD TO 2% REACTOR POWER, Rev 34.

REFERENCES TO BE SUPPLIED TO APPLICANT: None

Answer: A
68. G2.2.13 1

Which one of the following completes the statement listed below, in accordance with

OP-AA-200, “Equipment Clearance?”

<u>IF</u> maintenance activities are to be performed on a (1) ____________________ that would normally be tagged OPEN, **<u>THEN</u> ENTER** the component on the Tagging Record to show the initial and final positions to maintain status control as a (2) ______________________ .

A. (1) breaker
(2) Operating Permit (blue lock), and only one Tagout Holder for each tag-out is allowed.

B. (1) breaker
(2) No Tag item, and only one Tagout Holder for each tag-out is allowed.

C. (1) vent or drain valve
(2) Operating Permit (blue lock), and more than one Tagout Holder for each tag-out is allowed.

D. (1) vent or drain valve
(2) No Tag item, and more than one Tagout Holder for each tag-out is allowed.

<u>K/A</u>

Generic: Equipment Control
G2.2.13 Knowledge of tagging and clearance procedures.
(CFR 41.10 / 45.13) (RO – 4.1)

<u>K/A Match Analysis</u>

This question matches the K/A statement by requiring the RO applicant to correctly apply knowledge of the Dominion tagging and clearance procedure in order to properly tag (or, in this case, “no tag”) a vent valve that needs to have maintenance performed on the valve.

<u>Answer Choice Analysis</u>

A. INCORRECT. Both part (1) and part (2) are incorrect. OP-AA-200 step 3.4.1.c. states “IF maintenance activities are to be performed on a vent or a drain that would normally be tagged open, THEN ENTER the component on the Tagging Record to show the initial and final positions to maintain status control as a “No Tag” item. It is plausible that maintenance may need to be performed on a breaker that is already part of a tagging order; in this case, OP-AA-200 section 3.3.7.e states that the Danger Tag is moved to the breaker cubicle face or to the breaker door, then returned to the racking device or appropriate electrical isolation location. OP-AA-200 section 3.5 describes the use of Operating Permits (blue locks), which allows the Permit Holder (Tagout Holder) the authorization to OPERATE (but not do maintenance on) a designated component. Therefore, Operations Permit is also a plausible, but incorrect, distractor.

B. INCORRECT.   See analysis of ‘A’ above.

C. INCORRECT.   See analysis of A. above.

D.   CORRECT.   See analysis of ‘A.’ above.

<u>Supporting References</u>

1.   Dominion Nuclear Fleet Administrative Procedure OP-AA-200, “Equipment Clearance,” rev. 4.

<u>References Provided to Applicant</u>

None.

Answer: D
69.   G2.2.35 1

Which one of the following describes the Technical Specification definitions of
(1) COLD SHUTDOWN, and
(2) REFUELING SHUTDOWN?

A. (1) At least 1 % delta-k/k & Tave less than or equal to 200^{o}F
(2) At least 5 % delta-k/k, Tave less than or equal to 140 ^{o}F with fuel scheduled to be moved.

B. (1) At least 1 % delta-k/k & Tave less than or equal to 200^{o}F
(2) Reactor vessel head unbolted.

C. (1) At least 1.77 % delta-k/k & Tave less than or equal to 200^{o}F
(2) At least 5 % delta-k/k, Tave less than or equal to 140°F with fuel scheduled to be moved.

D. (1) At least 1.77 % delta-k/k & Tave less than or equal to 200^{o}F
(2) Reactor vessel head unbolted.

<u>K/A</u>

G2.2.35
Ability to determine TS mode of operation.

K/A Match Analysis
Q tests memory of TS Mode Definitions.

Answer Choice Analysis
A. Correct. See Tech Spec Definitions
B. Incorrect. Plausibility of part 2 exists with the potential confusion with REFUELING OPERATIONS.
C. Incorrect. Plausibility of part 1 exists because 1.77% is the SDM requirement for INTERMEDIATE SHUTDOWN.
D. Incorrect.

Supporting References
Tech Specs - Section 1 Definitions

References Provided to Applicant
None.

Answer: A

70. G2.2.39 2

Unit One is at Cold Shutdown and it is desired to perform a manual make-up to increase VCT level.

The following actions occur:
0930 - 1-CH-223 (PG isolation to blender) is opened for VCT make-up
0945 - VCT make-up is complete and the blender is secured

Which one of the following (1) describes the time at which 1-CH-223 is required to be closed in accordance with Tech Specs, and (2) whether sealing the valve after closure will comply with the conditions of the Tech Spec LCO 3.2, “Chemical and Volume Control System”?

A. (1) 1000
(2) Sealing the valve after closure will comply with the Tech Spec LCO.

B. (1) 1000
(2) Sealing the valve after closure will NOT comply with the Tech Spec LCO.

C. (1) 1045
(2) Sealing the valve after closure will comply with the Tech Spec LCO.

D. (1) 1045

(2) Sealing the valve after closure will NOT comply with the Tech Spec LCO.

K/A

Knowledge of less than one hour Technical Specification action statements for systems.

K/A Match Analysis

Applicant must have knowledge of a 15 minute LCO action statement in order to arrive at the correct answer.

Answer Choice Analysis

A. CORRECT. Tech Spec 3.2.E states that this and other valves may be opened for 15 minutes for makeup and/or planned dilution activities. Upon closing Tech Specs requires them to be locked, sealed, or otherwise secured.

B. INCORRECT. The second part is incorrect because the Tech Specs state that sealing the valve is acceptable. Plausible because the tech specs also state that locking the valve is acceptable; therefore, an applicant could have a credible misconception that the sealing the valve is not enough and think that the valve must be locked.

C. INCORRECT. See above.

D. INCORRECT. See above.

Supporting References

Modified from NRC 2004-301 exam.
Tech Spec 3.2.E

References Provided to Applicant

None.

Answer: A
71. G2.3.12 2

Current conditions on Unit 1:

- Core loading is in progress.
- Both loops of RHR are operable with 'B' RHR loop providing shutdown cooling.
- Reactor cavity level is 24 feet and slowly decreasing.

Based on the current conditions, which one of the following describes :

(1) the minimum level above the reactor vessel flange that must be maintained in the reactor cavity during REFUELING OPERATIONS in accordance with Technical Specifications

and

(2) what indications are available in the main control room to monitor this parameter?

A. (1) 26 feet
(2) RCS standpipe trend recorder

B. (1) 26 feet
(2) Cold calibrated pressurizer level

C. (1) 23 feet
(2) RCS standpipe trend recorder

D. (1) 23 feet
(2) Cold calibrated pressurizer level

<u>K/A</u>

Radiation Control
G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
(CFR 41.12 / 45.9 / 45.10) (RO – 3.2)

<u>K/A Match Analysis</u>

The question requires the applicant to understand a licensed operator's fuel handling responsibilities including knowledge of those plant conditions that would require stopping fuel movement.

<u>Answer Choice Analysis</u>

A. INCORRECT. *Plausible because 26' is the low level alarm point that is established during REFUELING OPERATIONS. The second half is plausible because a reactor cavity level of 23 feet is the point where two RHR loops are required to be operable. The reason for the level requirement is because with only* ***one RHR loop operable*** *the inventory in the reactor cavity would allow time to initiate emergency procedures not to swap RHR loops.*

B. INCORRECT. *Plausible because 26' is the low level alarm point that is established during REFUELING OPERATIONS. The second half is the correct reason for the limit.*

C. INCORRECT. *Plausible because the first half of the answer is the correct limit. The second half is plausible because a reactor cavity level of 23 feet is the point where two RHR loops are required to be operable. The reason for the level requirement is because with only* ***one RHR loop operable*** *the inventory in the reactor cavity would allow time to initiate emergency procedures not to swap RHR loops.*

D. CORRECT. T.S. 3.10.A.6 requires a minimum 23 feet above the reactor vessel flange in the reactor cavity and the reason is to ensure 99% of the iodine gas is removed in the event of a ruptured irradiated fuel assembly

<u>Supporting References</u>

1. Surry lesson plan, ND-92.5-LP-1, Refueling Overview, Rev. 17, Obj. D, pg.17

2. T.S. 3.10.A.7, pg 3.10-3

<u>References Provided to Applicant</u>

None.

Answer: D
72. G2.3.7 4

Corrective maintenance is scheduled to be performed on a blowdown radiation monitor sample cooler. The work will involve removing the relief valve on the Component Cooling (CC) side of the heat exchanger. HP has already posted the area as a 'Contaminated Area' although NO work has been done at this time.

You are assigned to tag out the cooler to support the maintenance and will have to enter the posted 'Contaminated Area'.

Which ONE of the following states the requirements to enter the area to execute the tagout?

A. Report to the Health Physics check-in desk and inform them of the actions to be taken. Obtain a DAD on the respective RWP. Anti-contamination clothing is NOT required for this work.

B. Report to the Health Physics check-in desk and inform them of the actions to be taken. Obtain a DAD on the respective RWP. Anti-contamination clothing is required for this work.

C. Report to the Health Physics check-in desk and inform them of the actions to be taken. A DAD is not required for this work but anti-contamination clothing is required for this work.

D. Since no work has been done and the area is clean, entry into the area is allowed with no further limitations or controls.

K/A

Generic: Radiation Control
G2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.
(CFR 41.12 / 45.10) (RO – 3.5)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly remember the requirements for generic RWPs, given a plausible operationally valid scenario.

Answer Choice Analysis

A. Incorrect – But plausible is the candidate believes that since the area is clean, that Anti-Cs are not required.

B. CORRECT.

Incorrect – But plausible if the candidate believes that a DAD is not required because the area is outside the area normally considered to be the RCA.

D. Incorrect - Area is normally a clean, non-RCA area. Posting for work yet to be performed may allow the candidate to conclude that no additional actions are required.

Supporting References

1. Dominion Station Administrative Procedure VPAP-2101, “Radiation Protection Program,” rev. 34.

References Provided to Applicant

None.

Answer: B
73. G2.4.19 2

Consider the following step from 1-E-0 (Reactor Trip or Safety Injection), and answer the question below:

*8. ___ CHECK RCP TRIP AND MINIFLOW RECIRC CRITERIA:

☐ a) Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS	☐ a) GO TO Step 9.
☐ b) RCS subcooling - LESS THAN 30°F [85°F]	☐ b) GO TO Step 9.
☐ c) Stop all RCPs	
☐ d) RCS pressure - LESS THAN 1275 psig [1475 PSIG]	☐ d) GO TO Step 9.
e) Close CHG pump miniflow recirc valves: ☐ • 1-CH-MOV-1275A ☐ • 1-CH-MOV-1275B ☐ • 1-CH-MOV-1275C	☐ e) Close 1-CH-MOV-1373.

Which ONE of the following 1) states the purpose of the " * " preceeding the step number, and 2) the significance of the bullets on substep "e" above?

A. (1) That the step is an Immediate Operator Action step.
(2) The bullets signify that the components must be operated in the order listed.

B. (1) That the step is a Continuous Action step
(2) The bullets signify that the components can be operated in any order.

C. (1) That the step is an Immediate Operator Action step
(2) The bullets signify that the components can be operated in any order.

D. (1) That the step is a Continuous Action step
(2) The bullets signify that the components must be operated in the order listed.

K/A

Knowledge of EOP layout, symbols, and icons.

K/A Match Analysis

Question requires knowledge of symbols used in the EOPs.

Answer Choice Analysis

A. INCORRECT. Immediate action is plausible because “[ ]” are used in the EOPs to designate immediate actions. Second part is also incorrect as this defines the use of lettered or numbered substeps.

B. CORRECT.

C. INCORRECT. Immediate action is plausible because “[ ]” are used in the EOPs to designate immediate actions. Second part is correct.

D. INCORRECT. First part is correct. Second part is also incorrect as this defines the use of lettered or numbered substeps.

Supporting References

ND-95.3-LP-2, EMERGENCY PROCEDURE WRITER'S FORMAT, Rev 13.

References Provided to Applicant

None.

Answer: B
74. G2.4.42 2

Initial Conditions:

- A large group of armed hostile intruders has gained access to multiple vital areas of the Surry Power Station.

Current Conditions:

- The Shift Manager has declared an emergency, and determined that the Local Emergency Operations Facility (LEOF) is unavailable due to the ongoing gun battle occurring on the station.

Based on the current conditions, which one of the following is (1) the backup facility for the LEOF; AND (2) personnel Accountability must be complete within __________ following a declaration of ALERT, SITE AREA EMERGENCY, or GENERAL

EMERGENCY, in accordance with the Surry Emergency Plan (SEP)?

A. (1) Technical Support Center (TSC)
   (2) 60 minutes

B. (1) Central Emergency Operations Facility (CEOF)
   (2) 30 minutes

C. (1) Technical Support Center (TSC)
   (2) 30 minutes

D. (1) Central Emergency Operations Facility (CEOF)
   (2) 60 minutes

K/A

Generic: Emergency Procedures/Emergency Plan
G2.4.42 Knowledge of emergency response facilities.
(CFR 41.10 / 45.11) (RO – 2.6)

K/A Match Analysis

This question matches the K/A statement by requiring the RO applicant to correctly remember the alternate sites for the EOF and the LMC, given a plausible operational scenario that would necessitate the use of E-plan response facilities offsite to the Surry plant. Applicants must also recall an important metric for Personnel Accountability.

Answer Choice Analysis

A. INCORRECT. Part (1) of the distractor is incorrect, part (2) is also incorrect but plausible as it is the closest emergency response facility. Part (2) is incorrect, because the Surry lesson plan for Emergency Plan Overview states: "Accountability must be complete within 30 minutes following declaration of Alert Site Area Emergency or General Emergency." 60 minutes is plausible because it is the normal time allowed to fully staff the EOF/TSC etc.

B. CORRECT.

C. INCORRECT. Part (1) incorrect see #1 above, part (2) correct.

D. INCORRECT. Part (1) is correct, part (2) incorrect. See above descriptions.

Supporting References

1. Surry Emergency Plan Procedure, “Surry Power Station Emergency Plan,” rev. 54 dtd 12/30/2008. Especially chapter 7.

2. Surry Lesson Plan ND-95.5-LP-1, "EMERGENCY PLAN OVERVIEW," rev. 8 dtd 07/29/08. See p. 18.

References Provided to Applicant

None.

Answer: B

75. G2.4.45 2

Unit One has experienced a Loss of Coolant Accident from 100% power and the control room team is performing 1-E-1 (Loss of Reactor or Secondary Coolant).

0200 - Plant conditions are as follows:
- All S/G narrow range levels are 33%
- RWST level is 25% and decreasing
- RCS Pressure is 4 psig
- Core Exit Thermocouple temperature is 229°F
- Containment Pressure peaked at 39 psia and is currently 17 psia (decreasing)

0215 - Plant conditions are as follows:
- A red path condition is identified for 1-FR-P.1 (Response to Imminent Pressurized Thermal Shock Condition).
- Annunciator 1A-G4 (LHSI PP 1A LOCKOUT OR OL TRIP) is received
- Annunciator 1D-E6 (CHG PP 1A 15H5 LOCKOUT) is received
- Annunciator 1A-A7 (RWST LO LVL) is received

Which ONE of the following states the required action to be taken by the team for these conditions?

A. Go to 1-FR-P.1 (Response to Imminent Pressurized Thermal Shock Condition)

B. Go to 1-ES-1.3 (Transfer to Cold Leg Recirculation)

C. Go to ECA-1.1 (Loss of Emergency Coolant Recirculation)

D. Continue in 1-E-1 until transition made to subsequent procedure

K/A

Plant Wide Generics
Ability to prioritize and interpret the significance of each annunciator or alarm.
(CFR: 41.10/43.5/45.3/45.12) (RO – 4.1)

K/A Match Analysis

The question requires the RO applicant to prioritize the listed alarms by significance based on the plant conditions and recognize that a failure of the semi-vital bus has occurred.

Answer Choice Analysis

A. INCORRECT. Plausible, as FR-P.1 is a procedure that is required to be performed, however, based on RWST level a transistion to ES-1.3 is required. FRs are not to be performed until the first 5 steps of ES-1.3 are complete.

B. CORRECT. The highes priority at this time is performance of ES-1.3, as indicated by RWST level and the RWST low level alarm.

C. INCORRECT. Plausible as a LHSI pump and HHSI pump were just locked out which significantly degrades the containment recirculation capability. Candidate could consider this entry criteria into ECA-1.1.

D. INCORRECT. Plausible if the candidate does not realize that ES-1.3 is a direct entry criteria and continues in E-1 until transition is made. Additionally, the candidate may not believe a transistion to FR-P.1 is required, as no actions are performed within FR-P.1 for a LBLOCA.

Supporting References

1. 1-ES-1.3 (Transfer to Cold Leg Recirculation)
2. 1-FR-P.1 (Repsonse to Imminent Pressurized Thermal Shock)
3. 1-ECA-1.1 (Loss of Emergency Coolant Recirculation)

References Provided to Applicant

None.

Answer: B

76.

Unit 1 Initial Conditions:
- Reactor Power = 100%

Unit 1 Current Conditions:
- 1-E-0, Reactor Trip or Safety Injection has been initiated
- RCP-1A seal failure exists and the pump has been secured
- RCS leakage is 200 gpm
- CETs = 550°F
- Containment pressure = 20 psia
- Subcooled margin = 50°F
- 2 RCPs are operating
- RVLIS dynamic head range = 30%

Based on the current conditions, which one of the following correctly states:
(1) the EAL classification required to be made by the shift manager

and

(2) the maximum time for notification of the NRC after the declaration is made, in accordance with EPIP-2.02, Notification of NRC, ?

(REFERENCE PROVIDED)

A. (1) Alert
(2) 15 minutes

B. (1) Alert
(2) 1 hour

C. (1) Site Area Emergency
(2) 15 minutes

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

77.

Current conditions:

- Unit 1 is at 100% power.
- Unit 2 is in COLD SHUTDOWN with the following charging system Alignment:
  - 2-CH-P-1B and 2-CH-P-1C control switches are in pull to lock.
- A leak has been found on 2-CH-447, Charging Pumps Cross-connect to Unit 1 Isolation Valve, and has been isolated for repairs

In accordance with Technical Specifications, which one of the following identifies:
(1) the status of LCO 3.2, Chemical and Volume Control for both units, and
(2) the basis for the LCO requirement?

Actions of LCO 3.2 are required on ____(1)____ due to the potential inability to _________(2)_________.

A. (1) Unit 1 but NOT Unit 2
(2) bring the plant to COLD SHUTDOWN conditions during specific postulated fire scenarios.

B. (1) Unit 1 but NOT Unit 2
(2) maintain a stable RCS makeup flowpath during specific postulated seismic scenarios.

C. (1) Unit 1 and Unit 2
(2) bring the plant to COLD SHUTDOWN conditions during specific postulated fire scenarios.

D. (1) Unit 1 and Unit 2.
(2) maintain a stable RCS makeup flowpath during specific postulated seismic scenarios.

78.

While Unit 1 was at 100% Power, multiple lightning strikes caused:
- 1-RC-PT-1445 (PRZR PRESS CNTRL CH 2) to fail HIGH.
- An automatic Main Generator trip.

- While performing immediate operator actions of 1-E-0, REACTOR TRIP OR SAFETY INJECTION, the Reactor Operator notes the following:
  - 1-RC-PT-1444 (PRZR PRESS CNTRL CH 1) = 1910 psig and INCREASING.
  - 1-RC-PT-1455 (PRZR PRESS PROTECT CH 1) = 1930 psig and INCREASING.
  - 1-RC-PT-1456 (PRZR PRESS PROTECT CH 2) = 1920 psig and INCREASING.

Current Conditions:
- Operators have transitioned to 1-ES-0.1, “REACTOR TRIP RESPONSE,” and are at the step to “CHECK PRZR PRESSURE CONTROL".
- No adjustments were made to <u>ANY</u> pressurizer pressure control component after the reactor trip.
- The Reactor Operator notes the following:
  - 1-RC-PT-1444 = 1990 psig and DECREASING.
  - 1-RC-PT-1455 = 2010 psig and DECREASING.
  - 1-RC-PT-1456 = 2000 psig and DECREASING.

Based upon the current conditions, (1) what is the NEXT required operator action to stop pressure from lowering, AND (2) what procedure(s) is/are required to be performed in parallel with 1-ES-0.1 to restore normal pressure control at 2235 psig?

A. (1) CLOSE BOTH of the open PRZR Spray Valves.
   (2) Perform 1-AP-31, “INCREASING OR DECREASING RCS PRESSURE,” to address BOTH the failed spray valve(s) AND the 1-RC-PT-1445 failure. Entry may also be made to 0-AP-53.00, "LOSS OF VITAL INSTRUMENTATION/CONTROLS," which will direct a transition to 1-AP-31.

B. (1) CLOSE the open PRZR PORV(s) or associated block valve(s).
   (2) Perform 1-AP-31 to address the 1-RC-PT-1445 failure ONLY. Entry may also be made to 0-AP-53.00, which will direct a transition to 1-AP-31.

C. (1) CLOSE BOTH of the open PRZR Spray Valves.
   (2) Perform 0-AP-53.00 to address BOTH the failed spray valve(s) AND the 1-RC-PT-1445 failure. A transition to 1-AP-31 will NOT be required.

D. (1) CLOSE the open PRZR PORV(s) or associated block valve(s).
   (2) Perform 0-AP-53.00 to address the 1-RC-PT-1445 failure ONLY. A transition to 1-AP-31 will NOT be required.

79.

Unit 1 Initial Conditions:
- Reactor power = 100%
- A steam line break develops inside containment

Current Conditions:
- Reactor is tripped
- RCS Pressure is 1780 psia and decreasing
- Containment pressure = 17.7 psia and increasing
- Pzr Level = 25% and decreasing
- Indicated RCS Subcooling (ICCM) =100°F
- AFW flow to each intact Steam Generator is 250 gpm.

The team has reached step 8 of 1-E-2 - CHECK IF SI FLOW SHOULD BE REDUCED

Based on the above conditions, which one of the following states: (1) which SI initiation signal will be generated if only 2 pressure switches are at the setpoint and (2) when transitioning from 1-E-2, to which procedure should the SRO transition?

A. (1) RCS Pressure Low
(2) 1-ES-1.1, SI TERMINATION

B. (1) RCS Pressure Low
(2) 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT

C. (1) Hi CLS
(2) 1-ES-1.1, SI TERMINATION

D. (1) Hi CLS
(2) 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT

80.

Initial Conditions:
- Unit 1 is experiencing a sustained Loss of All AC Power condition.
- The TDAFW pump shaft sheared on startup, and all efforts to cross-connect AFW with Unit 2 have failed.

Current Conditions:
- All emergency buses remain de-energized.
- Operators have just completed the step in ECA-0.0 to Check DC Bus Loads,and have placed both the DC emergency oil pump and the Air Side seal oil backup pump in PTL.
- Core Exit Thermocouples (CETCs) are 1202 °F and rising.
- STA reports the following Critical Safety Function Status Trees:
  - Core Cooling: RED
  - Heat Sink: RED
  - Containment: ORANGE
  - Inventory: YELLOW
  - Subcriticality: GREEN
  - Integrity: GREEN

Based upon the current conditions, what procedure is required to be used NEXT to mitigate the casualty?

A. 1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

B. 1-ECA-0.0, LOSS OF ALL AC POWER.

C. 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

D. 1-FR-C.1, RESPONSE TO INADEQUATE CORE COOLING.

81.

Unit 1 initial conditions:
- A large break LOCA occurred.
- The crew transitioned to 1-ES-1.3, Transfer to Cold Leg Recirculation.

Current conditions:
- RWST level is 12% and dropping.
- Recirc Mode Transfer (RMT) switches have been verfied in the “RMT” position.
- Amps and flow indications are oscillating on “A” and “B” LHSI pumps:
- Containment sump level is 54 inches.
- Crew has entered 1-ES-1.3, Attachment 1, Containment Sump Screen Blockage - Contingency Actions.

Based on current conditions, which one of the following describes the actions as required by Attachment 1 of 1-ES-1.3?

A. Trip both LHSI pumps, cross-tie Unit 1 charging system with Unit 2 RWST and return to the procedure step in effect of 1-ES-1.3.

B. Trip one LHSI pump, cross-tie LHSI pump suction with Unit 2 RWST and return to the procedure step in effect of 1-ES-1.3.

C. Trip both LHSI pumps, cross-tie Unit 1 charging system with Unit 2 RWST and go to ECA-1.1, Loss of Emergency Coolant Recirculation.

D. Trip one LHSI pump, cross-tie LHSI pump suction with Unit 2 RWST and go to ECA-1.1, Loss of Emergency Coolant Recirculation.

82.

Unit 1 initial conditions:
- The reactor failed to trip after receiving a trip signal
- SRO transitioned from 1-E-0, REACTOR TRIP OR SAFETY INJECTION, to 1-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.
- Reactor Power = 23% decreasing

Current plant conditions:
- Emergency Boration Initiated
- Reactor power = 3% decreasing
- Intermediate range channels indicate negative SUR
- Operators are verifying the reactor subcritical at the end of 1-FR-S.1

Based on the current plant conditions, which one of the following states (1) the emergency boration guidance in 1-FR-S.1, and (2) the next correct procedure transition as directed by 1-FR-S.1?

A. 1) Boration should continue. Obtaining adequate shutdown margin is not required prior to any procedure transition.
   2) Return to 1-E-0.

B. 1) Boration should continue. Obtaining adequate shutdown margin is not required prior to any procedure transition.
   2) Remain in 1-FR-S.1, and inititiate Attachment 1, VERIFYING APPLICABLE STEPS OF 1-E-0.

C. 1) Obtaining adequate shutdown margin is required prior to any procedure transition.
   2) Return to 1-E-0.

D. 1) Obtaining adequate shutdown margin is required prior to any procedure transition.
   2) Remain in 1-FR-S.1, and inititiate Attachment 1, VERIFYING APPLICABLE STEPS OF 1-E-0.

83.

The following sequence of events occurred on Unit 1:

- Time = 1400. Reactor Power is 10^{-8} Amps and stable for taking critical data.
- Time = 1401. IR N-35 has failed.
- Time = 1402. Reactor Operator reports N-35 Control Power AND Instrument Power fuses are blown.
- Time = 1403. Reactor Power is 10^{-8} Amps and stable.

Based on the given sequence of events, what offsite (i.e., external to Surry) notifications are required in accordance with VPAP-2802, "Notifications and Reports?"

Consider that all required internal notifications (i.e. to operations, engineering, and station management, etc.) are, or will be, made.

(REFERENCES PROVIDED)

	One hour or less External Notifications	Greater than one hour External Notifications
A.	None required	None required
B.	None required	Notification(s) required
C.	Notification(s) required	Notification(s) required
D.	Notification(s) required	None required

84.

Unit 1 current conditions:
- A steam generator (SG) tube leak exists on 'A' SG.
- The reactor is tripped.
- SI is not required.
- The crew has initiated 1-AP-24.01, Large Steam Generator Tube Leak.

Which one of the following completes the statements below concerning the tube leak on 'A' SG?

In accordance with 1-AP-24.01, SG level is required to be greater than _____(1)____ before stopping feed to the leaking SG AND Tave is required to be less than _____(2) before closing the associated main steam trip valve on the leaking SG.

A. (1) 12% Narrow Range
(2) 543°F

B. (1) 12% Narrow Range
(2) 547°F

C. (1) 22% Narrow Range
(2) 543°F

D. (1) 22% Narrow Range
(2) 547°F

85.

Unit 1 conditions:

- A LOCA has occurred
- The crew is performing Step 4 of 1-ES-1.1, SI TERMINATION, which requires the operators to stop all but one charging pump and place them in AUTO.
- With only one charging pump now running, RCS pressure begins to decrease.

Based on the above conditions, which one of the following states the required action (if any) and the correct procedure implementation?

A. Manually reinitiate SI and transition to 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

B. Manually restart a charging pump and monitor RCS pressure while continuing in 1-ES-1.1, SI TERMINATION.

C. Transition to 1-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

D. Continue in 1-ES-1.1, SI TERMINATION, to isolate HHSI to the cold legs, restart of a charging pump is not required.

86.

Initial conditions:
- Both units are at 100% power.
- The crew entered 1-AP-8.00, Loss of Normal Charging Flow
- Gas binding is suspected on the Unit 1 charging pumps.
- Charging and letdown have been secured on Unit 1.
- Venting of the charging pumps was attempted, but was unsuccessful.

Current conditions:
- Pressurizer level is 18% and dropping.

Based on the current conditions and in accordance with 1-AP-8.00, which one of the following describes:
(1) the reactor trip requirements for both units, and
(2) whether performance of Attachment 3, Charging Pump Cross-Connect, of 1-AP-8.00 is required?

A. (1) A reactor trip is required on Unit 1. A trip of Unit 2 reactor is not required.
(2) Performance of Attachment 3 is required.

B. (1) A reactor trip is required on Unit 1 and Unit 2.
(2) Performance of Attachment 3 is required.

C. (1) A reactor trip is required on Unit 1. A trip of Unit 2 reactor is not required.
(2) Performance of Attachment 3 is NOT required.

D. (1) A reactor trip is required on Unit 1 and Unit 2.
(2) Performance of Attachment 3 is NOT required.

87.

Unit 1 initial conditions:
- The Residual Heat Removal system is in service.
- At 10:03 A.M., a spurious Safety Injection (SI) actuation occurred on both trains of SI.
- The crew entered 1-AP-10.20, Response to Spurious Safety Injection with RCS Temperature Less Than 350°F.

Current conditions (10:05 A.M.):
- The SI Reset pushbuttons have been depressed on both trains.
- 1A-F3, SI INITIATED TRAIN A, is lit.
- A field operator has been sent to open the DC breaker for SI Train 'A'.

Based on the current conditions, which one of the following identifies…
(1) the charging pump(s) feeding the reactor IMMEDIATELY after the SI initiation, and
(2) the required procedural transition in accordance with 1-AP-10.20?

A. (1) One charging pump.
   (2) Exit 1-AP-10.20 and Go to 1-AP-10.19, Resetting Safety Injection.

B. (1) Three charging pumps.
   (2) Exit 1-AP-10.20 and Go to 1-AP-10.19, Resetting Safety Injection.

C. (1) One charging pump.
   (2) Initiate 1-AP-10.19, Resetting Safety Injection and continue performing 1-AP-10.20.

D. (1) Three charging pumps.
   (2) Initiate 1-AP-10.19, Resetting Safety Injection and continue performing 1-AP-10.20.

88.

Unit 1 initial conditions:
- Reactor power = 100%
- RCS pressure = 1900 psig decreasing rapidly

Current Conditions:
- Containment pressure = 24 psia (maximum) decreasing
- RCS pressure = 300 psig decreasing
- RWST level = 18% decreasing
- 1-E-1 LOSS OF REACTOR OR SECONDARY COOLANT in progress

Based on the above conditions, which one of the following states (1) the minimum operating containment spray pumps required to be operating as defined in FSAR Chapter 6, and (2) to what procedure 1-E-1 directs you to transition?

A. (1) One
(2) 1-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION

B. (1) One
(2) 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION

C. (1) Two
(2) 1-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION

D. (1) Two
(2) 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION

89.

Unit 2 initial conditions:
- A loss of offsite power has occurred.
- The crew has entered 2-E-0
- Emergency diesel generators (EDG) #2 and #3 have started and tied to their respective emergency buses.

Current conditions:
- 2-EP-BKR-25J3, EDG #3 Emergency Supply breaker has tripped.
- The following alarms are lit on annunciator panel
  - 0-VSP-B5 – EMERG GEN 3 TRBL
  - 0-VSP-A5 – EMERG GEN 3 DIFF
  - 2K-G4 – 4KV EMERG BUS EMERG SUP AUTO TRIP
- The crew entered 0-AP-17.05, EDG 3 – Emergency Operations.
- 14 minutes later the #3 EDG was identified as being shutdown.

Based on these current conditions, which one of the following identifies (1) the FIRST attachment to be performed in accordance with 0-AP-17.05, and (2) the condition that tripped breaker 2-EP-BKR-25J3?

A. (1) Attachment 1, Auxiliary Trip Relay Actuation - Contingency Actions.
   (2) Field voltage was lost to EDG #3.

B. (1) Attachment 1, Auxiliary Trip Relay Actuation - Contingency Actions.
   (2) EDG #3 differential lockout relay actuated.

C. (1) Attachment 3, Stripping the 2J Bus.
   (2) Field voltage was lost to EDG #3.

D. (1) Attachment 3, Stripping the 2J Bus.
   (2) EDG #3 differential lockout relay actuated.

90.

Unit 1 Current Conditions:
- Reactor Power is 35%
- An Instrument Air Dryer malfunction occurs
- Operators have attempted, but could not bypass the instrument air dryer
- Instrument Air Pressure is 45 psig and decreasing.
- Operators have just started taking actions in 1B-E6, IA LO HDR PRESS/IA COMPR 1 TBL

Based on the current conditions, which one of the following 1) states the correct procedures to use to address the reduced Instrument Air Pressure, and 2) following the reactor trip whether or not a four hour notification is required in accordance with VPAP-2802, Notifications and Reports.

(REFERENCE PROVIDED)

A. (1) Perform AP-40.00, "Non-recoverable Loss of Instrument Air," in conjunction with the EOP network of procedures.
   (2) Four hour notification is required.

B. (1) Do NOT perform AP-40.00, "Non-recoverable Loss of Instrument Air," in conjunction with the EOP network of procedures.
   (2) Four hour notification is required.

C. (1) Perform AP-40.00, "Non-recoverable Loss of Instrument Air," in conjunction with the EOP network of procedures.
   (2) Four hour notification is NOT required.

D. (1) Do NOT perform AP-40.00, "Non-recoverable Loss of Instrument Air," in conjunction with the EOP network of procedures.
   (2) Four hour notification is NOT required.

91.

The following sequence of events occurs on Unit 1:

- Time = 1800. During a power increase, the unit stabilizes power at 85% to perform a calorimetric. The delta flux target is 0.0%. Delta flux readings have remained in the target band for the past 48 hours.
- Time = 1801. A rod control circuit malfunction causes rods to insert.
- Time = 1802. Rod motion is stopped.

	Delta Flux	NI 41	NI 42	NI 43	NI 44
- TIme = 1803	-13.0%	85%	84%	85%	86%
- Time = 1817	-13.0%	85%	84%	85%	86%
- Time = 1837	-10.0%	85%	84%	85%	86%
- Time = 1905	-6.0%	85%	84%	85%	86%
- Time = 1910	-5.0%	85%	84%	85%	86%

Based on the given sequence of events, at time 1911, which one of the following answers (1) if reactor power is allowed to be raised above 90% in accordance with Technical Specification 3.12, CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS, AND (2) the reason(s) for the above answer, in accordance with TS 3.12 BASIS?

(REFERENCE PROVIDED)

A. (1) No.
(2) Radial Xenon distribution in the core has been affected to an extent that reactor power reduction is required and a subsequent power range high flux trip setpoint reduction to minimize the effects of xenon redistribution on the radial power distribution during load changes.

B. (1) No.
(2) Axial Xenon distribution in the core has been affected to an extent that reactor power reduction is required and a subsequent power range high flux trip setpoint reduction to minimize the effects of xenon redistribution on the axial power distribution during load changes.

C. (1) Yes.
(2) Axial Xenon distribution control at less than 90% power is not as significant as axial Xenon control at full power, and allowances were made in the accident analyses (which is the basis of the Delta Flux control procedures) for heat flux peaking factors for accidents occurring at less than 90% power.

D. (1) Yes.
(2) Radial Xenon distribution in the core was NOT affected sufficiently to change the heat flux peaking factors which can be reached on a subsequent return to full power within the target band.

92.

Unit 1 initial conditions:
- The reactor was tripped due to a tube rupture on ‘B’ steam generator (SG).
- A Safety Injection (SI) initiated.
- 1-E-3, Steam Generator Tube Rupture, is being performed.
- RCS cooldown to target temperature of 495°F has been completed.

Following the cooldown plant conditions are as follows:
- ‘B’ SG pressure is 1020 psig and stable.
- Core Exit Thermocouples (CETC) is 490°F.
- SI has been reset.
- LHSI pumps have been reset.
- RCS subcooling is 35°F.
- Operators are evaluating RCS subcooling in accordance with 1-E-3 before continuing with RCS depressurization.

Based on the current plant conditions, which one of the following describes (1) whether a procedure transition to 1-ECA-3.1, SGTR with Loss of Reactor Coolant - Subcooled Recovery is required AND (2) the reason for the subcooling evaluation in accordance with 1-E-3?

A. (1) Transition to 1-ECA-3.1 is required.
   (2) Subcooling cannot be assured and actions must be taken to re-establish subcooling.

B. (1) Transition to 1-ECA-3.1 is NOT required.
   (2) Subcooling cannot be assured and actions must be taken to re-establish subcooling.

C. (1) Transition to 1-ECA-3.1 is required.
   (2) Loss of RCS coolant from other than the tube rupture may be occurring.

D. (1) Transition to 1-ECA-3.1 is NOT required.
   (2) Loss of RCS coolant from other than the tube rupture may be occurring.

93.

Plant Conditions:

- Plant fire protection systems were manually disabled to perform unplanned maintenance.
- The disabled fire protection systems allowed a small fire to escalate into a large fire in the MCR.
- The Unit Supervisor determines that MCR must be evacuated.
- The Auxiliary Shutdown Panel is available.

Which one of the following correctly (1) describes the entry into 0-FCA-1.00, Limiting MCR Fire, and (2) states whether EOP actions or FCA actions take precedence when conflicting guidance is encountered?

A. (1) Directly enter 0-FCA-1.00 prior to going to 0-AP-48.00, Fire Protection - Operations Response.
   (2) EOP actions take precedence over FCA actions when conflicting guidance is encountered.

B. (1) Directly enter 0-FCA-1.00 prior to going to 0-AP-48.00, Fire Protection - Operations Response.
   (2) FCA actions take precedence over EOP actions when conflicting guidance is encountered.

C. (1) First enter 0-AP-48.00, Fire Protection - Operations Response, then enter 0-FCA-1.00.
   (2) EOP actions take precedence over FCA actions when conflicting guidance is encountered.

D. (1) First enter 0-AP-48.00, Fire Protection - Operations Response, then enter 0-FCA-1.00.
   (2) FCA actions take precedence over EOP actions when conflicting guidance is encountered.

94.

A condition has occured in the plant that requires the implementation of 10CFR-50.54(x) to protect public health and safety. Adequate time for approval exists before the action is required.

Which ONE of the following states 1) the required level of authorization prior to taking the deviating action and 2) what follow-up actions are required following the completion of the action?

A. 1) The highest level of authority on site is required to approve the action.
   2) Notification of the NRC is required.

B. 1) The highest level SRO licensed individual is required to approve the action.
   2) Notification of the NRC is NOT required.

C. 1) The highest level of authority on site is required to approve the action.
   2) Notification of the NRC is NOT required.

D. 1) The highest level SRO licensed individual is required to approve the action.
   2) Notification of the NRC is required.

95.

In accordance with 1-OP-FH-001, Controlling Procedure for Refueling, which one of the following identifies when subcriticality multiplication monitoring must be initiated during core reload?

A. After the fourth assembly has been loaded in the core.

B. After the sixth assembly has been loaded in the core.

C. After the eigth assembly has been loaded in the core.

D. After the tenth assembly has been loaded in the core.

96.

Initial Conditions:
- Unit 2 is in a refueling outage.

Current Conditions:
- The Shift Technical Advisor (STA) identifies that, due to multiple schedule changes and emergent switchyard work, there is a two-hour window that will occur several hours later on the current shift that would place the unit in a RED Shutdown Risk (SDR) condition.

Which ONE of the following states 1) whether entry into the RED SDR condition is permissible and 2) limitations or restrictions that exist for this condition?

A. 1) Yes, entry in the RED SDR condition is allowed.
   2) The RED condtion shall not exceed the duration of one shift.

B. 1) Yes, entry in the RED SDR condition is allowed.
   2) The RED condition can exist, providing administrative controls are in place to prevent further degradation.

C. 1) No, entry in the RED SDR condition is NOT allowed.
   2) A RED condition shall not exceed one hour, providing administrative controls are in place to prevent further degradation.

D. 1) No, entry in the RED SDR condition is NOT allowed.
   2) The activities that would place the station in this configuration must be rescheduled.

97.

Unit 1 is in Hot Shutdown and preparations are being made to conduct an infrequently performed test to verify the time dependence of reactor coolant flow following an intentional loss of a reactor coolant pump.

The test has been classified as an ICCE (Infrequently Conducted or Complex Evolution) Category I evolution.

Which ONE of the following states

1) who is required to review and approve this ICCE procedure in accordance with OP-AA-106 (Infrequently Conducted or Complex Evolutions)

and

2) an acceptable example of the individual who meets the general selection guidance for the seniority and qualifications of the Senior Operations Manager selected to provide oversight of the test in accordance with OP-AA-106.

A. (1) Manager - Nuclear Operations
   (2) Control Room Unit Supervisor with an active SRO license at Surry.

B. (1) Facility Safety Review Committee
   (2) Operations manager who formerly held an SRO license at Surry.

C. (1) Manager - Nuclear Operations
   (2) Operations manager who formerly held an SRO license at Surry.

D. (1) Facility Safety Review Committee
   (2) Control Room Unit Supervisor with an active SRO license at Surry.

98.

Current conditions:

- Unit 1 is at full power.

Which one of the following states:

1) the DOSE EQUIVALENT IODINE-131 limit for RCS activity in accordance with Technical Specification 3.1.D, Maximum Reactor Coolant Activity

AND

2) the assumed release duration through the Main Steam Safety Valves and Atmospheric Relief Valves during the postulated steam generator tube rupture in accordance with Technical Specification bases?

A. 1) Less than or equal to 0.1 µCi/cc.
   2) 60 minutes.

B. 1) Less than or equal to 0.1 µCi/cc.
   2) 30 minutes.

C. 1) Less than or equal to 1.0 µCi/cc
   2) 60 minutes.

D. 1) Less than or equal to 1.0 µCi/cc
   2) 30 minutes.

99.

An event has occured in the station and the emergency plan has been entered.

In accordance with EPIP-1.01 (Emergency Manager Controlling Procedure), The Shift Manager is required to announce:

- that he or she has assumed the Station Emergency Manager (SEM) position
- the emergency classification, the time of declaration
- AND _________(1)_____________ .

The event was a General Emergency, the initial notification of the applicable PAR _________(2)___________ .

A. (1) the EAL, and any pertinent upgrade criteria.
   (2) is required to be made at the same time the initial notification of the GE is made.

B. (1) the EAL. Announcing upgrade criteria is not required by EPIP-1.01.
   (2) is required to be made at the same time the initial notification of the GE is made.

C. (1) the EAL, and any pertinent upgrade criteria.
   (2) is NOT required to be made at the same time the initial notification of the GE is made.

D. (1) the EAL. Announcing upgrade criteria is not required by EPIP-1.01.
   (2) is NOT required to be made at the same time the initial notification of the GE is made.

100.

The following conditions exist:

- A manual Rx trip was initiated 10 minutes ago based on AP-16.00, Ecessive RCS Leakage, criteria
- Pressurizer level is off-scale low
- Pressurizer pressure is 1500 psig and decreasing
- All SG levels are 5% NR and slowly increasing
- All SG pressures are 1005 psig and stable
- All main steam line radiation monitors are reading .02 mr/hr
- MGPI Vent-Vent radiation monitor is reading 4.3 E6 cpm
- Containment pressure is 10.2 psia
- Containment sump level is 47%
- VSP-F-4, “AUX Building Sump HI Level,” is illuminated
- Safeguards Area Sump high level alarm is locked in

Upon exiting E-0, which one of the following is the correct procedure transitions for the event in progress?

A. Go to E-1 (Loss of Reactor or Secondary Coolant), ECA-1.2 (LOCA Outside Containment), then ECA-1.1 (Loss of Emergency Coolant Recirculation)

B. Go to E-1 (Loss of Reactor or Secondary Coolant), then ECA-1.1 (Loss of Emergency Coolant Recirculation), then ECA-1.2 (LOCA Outside Containment)

C. Go to ECA-1.1 (Loss of Emergency Coolant Recirculation), then ECA-1.2 (LOCA Outside Containment)

D. Go to ECA-1.2 (LOCA Outside Containment), then ECA-1.1 (Loss of Emergency Coolant Recirculation)

SRO REFERENCE MATERIAL

1. This package
2. SRO/RO Exam Package
3. EAL Matrices
4. VPAP-2802

a. At a power level greater than 90 percent of RATED POWER, if the indicated axial flux difference deviates from its target band, within 15 minutes either restore the indicated axial flux difference to within the target band or reduce the reactor power to less than 90 percent of RATED POWER.

b. At a power level less than or equal to 90 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference is within the limits shown on TS Figure 3.12-3. One minute penalty is accumulated for each one minute of operation outside of the target band at power levels equal to or above 50% of RATED POWER.

(2) If Specification 3.12.B.4.b.(1) is violated, then the reactor power shall be reduced to less than 50% power within 30 minutes and the high neutron flux setpoint shall be reduced to less than or equal to 55% power within the next four hours.

(3) A power increase to a level greater than 90 percent of RATED POWER is contingent upon the indicated axial flux difference being within its target band.

(4) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to TS Table 4.1-1 provided the indicated axial flux difference is maintained within the limits of TS Figure 3.12-3. A total of 16 hours of operation may be accumulated with the axial flux difference outside of the target band during this testing without penalty deviation.

c. At a power level less than or equal to 50 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of RATED POWER is contingent upon the indicated axial flux difference not being outside its target band for more than one hour accumulated penalty during the preceding 24-hour period. One half minute penalty is accumulated for each one minute of operation outside of the target band at power levels between 15% and 50% of RATED POWER.

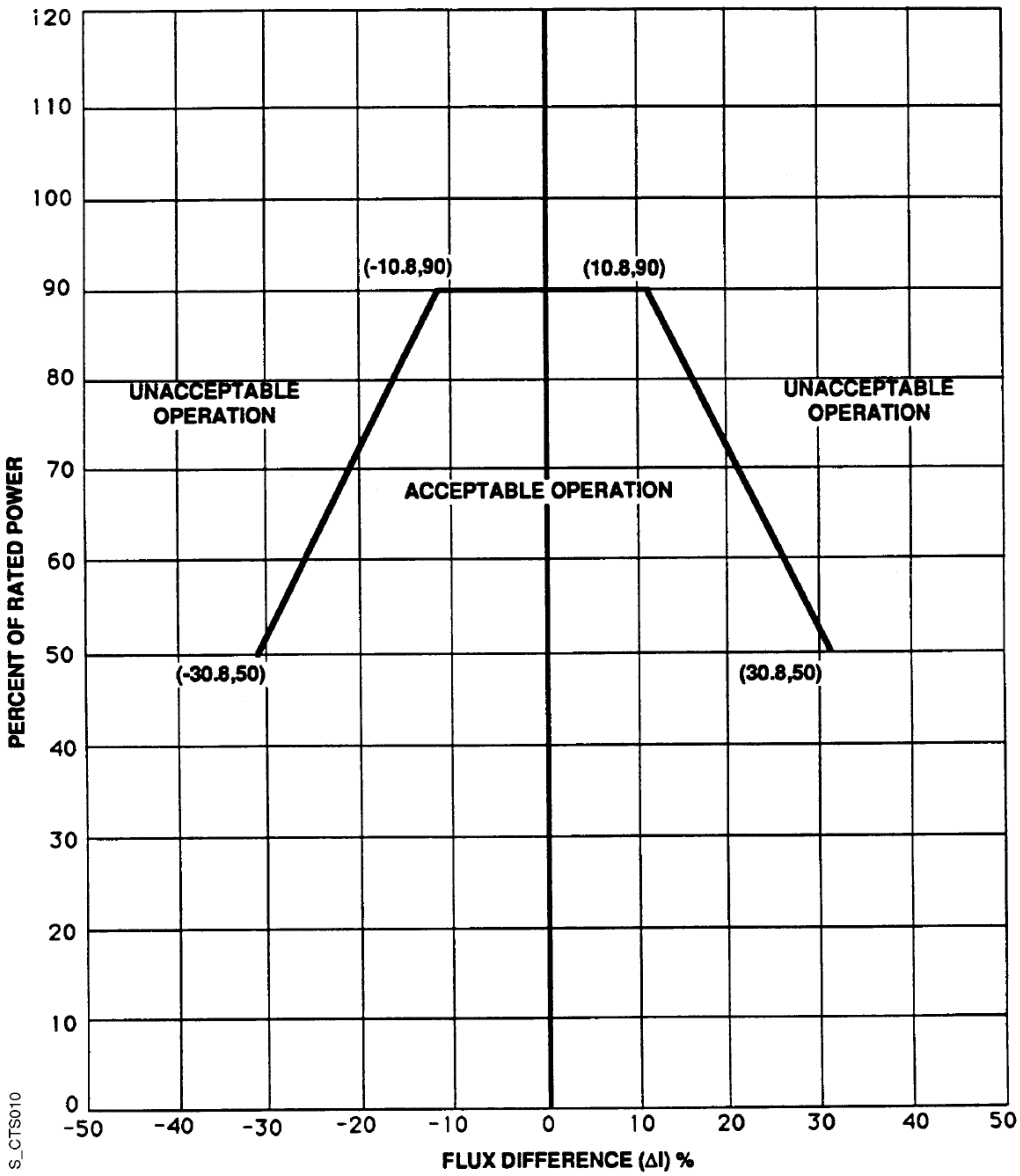
d. The axial flux difference limits for Specifications 3.12.B.4.a, b, and c may be suspended during the performance of physics tests provided:

(1) The power level is maintained less than or equal to 85% of RATED POWER, and

(2) The limits of Specification 3.12.B.1 are maintained. The power level shall be determined to be less than or equal to 85% of RATED POWER at least once per hour during physics tests. Verification that the limits of Specification 3.12.B.1 are being met shall be demonstrated through in-core flux mapping at least once per 12 hours.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in Specification 3.12.B.4.a and the flux difference time limits in Specifications 3.12.B.4.b and c. If the alarms are out of service temporarily, the axial flux difference shall be logged and conformance to the limits assessed every hour for the first 24 hours and half-hourly thereafter. The indicated axial flux difference for each excore channel shall be monitored at least once per 7 days when the alarm is OPERABLE and at least once per hour for the first 24 hours after restoring the alarm to OPERABLE status.

AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED POWER
SURRY POWER STATION



Dominion®

Station Administrative Procedure

Title: Notifications and Reports

Process / Program Owner: Director Nuclear Station Safety and Licensing

Procedure Number	Revision Number	Effective Date
VPAP-2802	**34**	**On File**

Revision Summary

Revised to incorporate 40 CFR 82, Protection of Stratospheric Ozone, reportability instruction.

- Added 5.8.26 (under Environmental Compliance Coordinator responsibilities) - "Notifying Electric Environmental Services of the failure to repair a commercial or comfort refrigeration unit containing greater than 50 pounds of refrigerant within 30 days; see Step 6.20.2."
- Added 5.13.26 (under Director Electric Environmental Services responsibilities) - "Notifying the EPA headquarters in accordance with 40 CFR 82.166(N)(i) of the failure to repair a commercial or comfort refrigeration unit containing greater than 50 pounds of refrigerant within 30 days; see Step 6.20.2."
- Added 6.20.2, 40 CFR 82, Protection of Stratospheric Ozone.

The following changes were in response to North Anna CA 149585 (CR 352954), North Anna Environmental Program Audit:

- Added 3.1.106 - North Anna CA 149585 (CR 352954), North Anna Environmental Program Audit.
- Added 6.3.2.e.4 NOTE "Items marked with an asterisk(*) on Attachment 1are required to be reported to the response agencies listed in block 10 of the attachment. **(Reference 3.1.106)**"
- Revised Attachment 1, Oil or Hazardous Substance Release Report - 721973(Apr 2010) - added asterisks referenced in 6.3.2.e.4 NOTE; added block 6 - "Material Discharged (for oil spills use the codes found in the SPCC Oil Spill Report form)"; added block 12 - "If No, is there a potential for release? Yes - No"

The following changes were administrative in nature:

- 4.4 and 4.23 - changed "Nuclear Mutual Limited" to "Nuclear Electric Insurance Limited"; deleted references to "Nuclear Mutual Limited" and "NML."
- Changed "Risk Services" to "Corporate Risk Management."
- Updated 6.1.1, Notifications - added "State Department of Emergency Management—(804) 674-2400, ask for EOC Duty Officer"
- Updated USNRC phone number and address.
- Updated 6.3.3.i - added third bullet: "Louisa/Surry County Administrator."
- Revised Attachment 1, Oil or Hazardous Substance Release Report - 721973(Apr 2010) - changed "Water Quality" and "Corporate Water Quality Department" to "Electric Environmental Services."

Approvals on File

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NOTE: Section 6 includes a CFR Part or Section number within many of its titles. Some of the CFR titles were edited to make the format consistent for this procedure. In other cases, referenced Parts or Sections do not have titles, so titles were created for this procedure. **Do not**, therefore, rely on these titles to cite CFR Part or Section titles. CFR Titles that appear in Section 3.0 **are** cited as they appear in the CFR.

1.0 PURPOSE

This procedure defines responsibilities, establishes requirements, and provides instructions to implement notification, report, and posting requirements applicable to Station activities, situations, and events.

This procedure is structured to guide you to specific information based on what you need to know for specific situations. Figure 1 suggests ways to find certain information quickly.

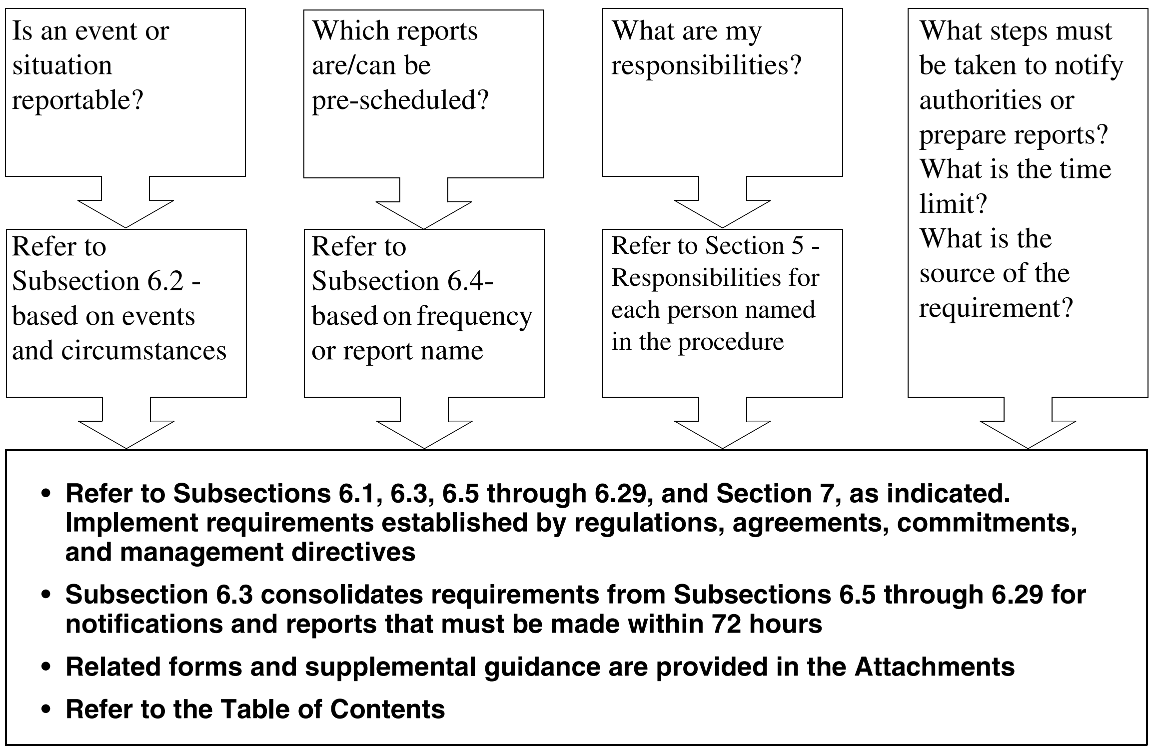

Figure 1
Optimal Use of this Procedure

2.0 SCOPE

2.1 This procedure applies to personnel with duties associated with the Station. Unless specifically indicated otherwise, references to the Station include the Independent Spent Fuel Storage Installation (ISFSI).

2.2 This procedure does **not** apply to:

- Actions controlled by Emergency Plan Implementing Procedures (EPIPs)
- Notifications and reports associated with non-nuclear-related insurance
- Internal Dominion notifications or reports that are not directly associated with external notifications or reports
- Reporting and posting requirements of nonregulatory agencies such as the Virginia Employment Commission, Internal Revenue Service, or Social Security Administration

2.3 Due to the staggered implementation of the Central Reporting System (CRS) for the Dominion nuclear fleet, the nuclear power station term for its station's corrective action system item: Plant Issue, Condition Report and/or Action Request are all being replaced by the CRS term, Condition Report. A Condition Report (CR) is the document used to identify Significant Conditions Adverse to Quality, Conditions Adverse to Quality, Adverse Trends, and other issues that do not meet expectations of management. The terms Action Request, Plant Issue (Deviation) and Deviation Report are synonymous with the term Condition Report. This statement is applicable to any procedure where the terms Deviation Report, DR, Plant Issue, or PI appear. Numbered items generated by the station's corrective action system prior to CRS implementation are not changed.

3.0 REFERENCES/COMMITMENT DOCUMENTS

3.1 References

3.1.1 10 CFR, Energy

3.1.2 18 CFR 12, Safety of Water Power Projects and Project Works

3.1.3 29 CFR 1900, Occupational Safety and Health Administration, Department of Labor

3.1.4 40 CFR, Protection of Environment

3.1.5 49 CFR, Transportation (Subchapter C—Hazardous Materials Regulations)

3.1.6 Emergency Planning and Community Right-To-Know Act of 1986, PL99-499

3.1.7 Code of Virginia § 10.1-1429, Notice of Release of Hazardous Substance

3.1.8 Code of Virginia § 62.1-44.34:19, Reporting of Discharge

3.1.9 Virginia State Water Control Board Regulation 9 VAC 25-31-10 § I, Reporting Requirements

3.1.10 Virginia State Water Control Board Regulation 9 VAC 25-90-10, Oil Discharge Contingency Plans and Administrative Fees for Approval

3.1.11 Virginia State Water Control Board Regulation 9 VAC 25-200, Water Withdrawal Reporting

3.1.12 NRC Regulatory Guide 1.16, Reporting of Operating Information—Appendix A: Technical Specifications, Revision 4, August 1975

3.1.13 NRC Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Revision 1, June 1974

3.1.14 NRC Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Revision 3, May 1983

3.1.15 NRC Regulatory Guide 5.62, Reporting of Safeguards Events, Revision 1, November 1987

3.1.16 NRC Regulatory Guide 10.1, Compilation of Reporting Requirements for Persons Subject to NRC Regulations, Revision 4, October 1981

3.1.17 NRC Radiological Assessment Branch, Branch Technical Position: An Acceptable Radiological Environmental Monitoring Program, Revision 1, November 1979

3.1.18 North Anna Power Station Technical Specifications

3.1.19 North Anna Independent Spent Fuel Storage Installation License and Technical Specifications

3.1.20 Surry Power Station Units 1 and 2 Operating Licenses and Technical Specifications

3.1.21 Surry Independent Spent Fuel Storage Installation License and Technical Specifications

3.1.22 North Anna Power Station VPDES Permit No. VA0052451

3.1.23 Surry Power Station VPDES Permit No. VA0004090

3.1.24 North Anna Power Station Virginia Air Pollution Control Board Registration, No. 40726, February 28, 1983

3.1.25 Surry Power Station Virginia Air Pollution Control Board Registration, No. 50336, August 15, 1988

3.1.26 Topical Report, Quality Assurance Program

3.1.27 North Anna Power Station Emergency Plan

3.1.28 North Anna Hydroelectric Project Emergency Action Plan

3.1.29 Surry Power Station Emergency Plan

3.1.30 North Anna Power Station Updated Final Safety Analysis Report

3.1.31 North Anna Independent Spent Fuel Storage Installation Safety Analysis Report

3.1.32 Surry Power Station Updated Final Safety Analysis Report

3.1.33 Surry Independent Spent Fuel Storage Installation Safety Analysis Report

3.1.34 Nuclear Electric Insurance Limited Policies

3.1.35 Inservice Inspection Manual

3.1.36 Nuclear Loss Prevention Standards Manual

3.1.37 Oil Discharge Contingency Plan, and Appendix I and Appendix L

3.1.38 Virginia Power Technical Report No. PE-0013, North Anna Power Station Response to Regulatory Guide 1.97

3.1.39 Virginia Power Technical Report No. PE-0014, Surry Power Station Response to Regulatory Guide 1.97

3.1.40 EPIP-1.01, Emergency Manager Controlling Procedure

3.1.41 EPIP-2.01, Notification of State and Local Governments

3.1.42 EPIP-2.02, Notification of NRC

3.1.43 LI-AA-101, License Basis Document Changes Process

3.1.44 LI-AA-101-1001, License Basis Document Changes Process Reference

3.1.45 LI-AA-110, Commitment Management

3.1.46 LI-AA-200, NRC Licensing Correspondence

3.1.47 LI-AA-301, Implementation of 10 CFR 21, Reporting of Defects and Noncompliance

3.1.48 LI-AA-500, NRC/INPO/WANO Performance Indicator and MOR Reporting

3.1.49 MS-AA-WHR-401, Receiving

3.1.50 MS-AA-WHS-131, Storage and Handling

3.1.51 OP-AA-100, Conduct of Operations

3.1.52 OP-AA-102, Operability Determination

3.1.53 OP-AP-105, Post Trip Review

3.1.54 PI-AA-100-1007, Operating Experience Program

3.1.55 PI-AA-200, Corrective Action

3.1.56 VPAP-0301, Design Change Process

3.1.57 VPAP-0602, Vendor Technical Manual Control

3.1.58 VPAP-1406, Nuclear Material Control

3.1.59 VPAP-1901, Industrial Safety and Health

3.1.60 VPAP-2101, Radiation Protection Program

3.1.61 VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry)

3.1.62 VPAP-2104, Radioactive Waste Process Control Program (PCP)

3.1.63 VPAP-2202, Control of Chemicals and Hazardous Substances

3.1.64 VPAP-2203, Oil Spill Prevention, Control, and Countermeasures (SPCC) Plan

3.1.65 VPAP-2602, Safety Parameter Display System (SPDS) (Surry)

3.1.66 VPAP-2606, Safety Parameter Display System (SPDS) (North Anna)

3.1.67 VPAP-2703, Control Room Simulator

3.1.68 VPAP-2803, UFSAR and ISFSI SAR Management

3.1.69 Emergency Telephone Directory

3.1.70 FEMA Guidance Memorandum PR-1, Policy on NUREG-0654/FEMA-REP-1 and 44 CFR 350 Periodic Requirements, October 1, 1985

3.1.71 NRC Generic Letter 91-02, Subject: Reporting Mishaps Involving LLW Forms Prepared for Disposal, December 28, 1990

3.1.72 NRC Generic Letter 91-03, Reporting of Safeguards Events

3.1.73 NRC IE Information Notice No. 79-30, Reporting of Defects & Noncompliances, 10 CFR 21

3.1.74 NRC Information Notice No. 89-89, Event Notification Worksheets, December 26, 1989

3.1.75 NUREG-0302, Remarks Presented (Questions/Answers Discussed) at Public Regional Meetings to Discuss Regulations (10 CFR Part 21) for Reporting of Defects and Noncompliance, July 12-26, 1977, Revision 1, October 1977

3.1.76 NUREG-0472, Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors, Revision 3, March 1982

3.1.77 NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, Revision 2, October 2000

3.1.78 NUREG-1304, Reporting of Safeguards Events, February 1988

3.1.79 NUREG/BR-0006, Instructions for Completing Nuclear Material Transaction Reports and Concise Note Forms, Revision 4, February 1, 2000

3.1.80 NUREG/BR-0007, Instructions for Completing Material Balance Report and Physical Inventory Listing, Revision 3, February 1, 2000

3.1.81 ANSI/IEEE C57.104-1978, Guide for Detection and Determination of Generated Gases in Oil-Immersed Transformers and Their Relation to the Serviceability of the Equipment

3.1.82 ASTM E 185, Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (1F)

3.1.83 Westinghouse letter to NRC, Serial No. NS-CE-1749, April 6, 1978

3.1.84 Varga, Steven A., NRC, to Stewart, W. L., Virginia Electric and Power Company, March 11, 1983

3.1.85 Slayton, A. E., Jr., State Office of Emergency and Energy Services, letter to Stewart, W. L., Virginia Power, April 17, 1984

3.1.86 Speck, Samuel W., FEMA, letter to Regional Directors, Subject: Guidance Memorandum PR-1, Policy on FEMA Guidance Memorandum PR-1, Policy on NUREG-0654/FEMA-REP-1 and 44 CFR 350 Periodic Requirements, October 1, 1985 and 44 CFR 350 Periodic Requirements, October 4, 1985

3.1.87 Stewart, W. L., to Distribution, Subject: NRC Correspondence, October 8, 1985

3.1.88 Rosbe, William L., and Amy R. Chester, Hunton and Williams, Memorandum to Marshall, B. M., Virginia Power, Subject: Summary of Federal and Virginia Reporting Requirements for a Release of a Hazardous Substance or Oil (and new SARA Reporting Requirements), January 9, 1987

3.1.89 Engle, Leon, NRC, letter to Stewart, W. L., Virginia Electric and Power Company, Subject: Projected Values of Material Properties For Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events, May 29, 1987

3.1.90 Engle, Leon B., NRC, letter to Cruden, D. S., Virginia Electric and Power Company, Subject: Compliance with ATWS Rule, 10 CFR 50.62, Surry Power Plant, Units No. 1 and No. 2 (Surry 1 & 2) and North Anna Power Station, Units No. 1 and No. 2 (NA-1&2) (TAC Nos. 59147, 59148, 59117 and 59118), May 26, 1988

3.1.91 Hegner, J. D., to Kemp, P. A., Subject: FERC Dam Reporting Requirements, September 10, 1991

3.1.92 NEI 99-02, Regulatory Assessment Performance Indicator Guideline

3.1.93 Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, AIF/NESP-036, 1986

3.1.94 Regulatory Guide 1.159, Assuring the Availability of Funds for Decommissioning Nuclear Reactors, dated August 1990

3.1.95 Edison, Gordon E. Sr., NRC, letter (Serial number 01-040) to Christian, David A., Virginia Electric and Power Company, Subject: North Anna Power Station, Units 1 and 2 and Independent Spent Fuel Storage Installation (ISFSI), and Surry Power Station, Units 1 and 2 and ISFSI - Approval of proposed variation in reporting schedule pursuant to 10 CFR 72.76(a) and 74.13(a)(1) (TAC numbers MB0328, MB0329, MB0330, and MB0331), December 29, 2000

3.1.96 Code of Virginia § 10.1-2300, Virginia Antiquities Act

3.1.97 ASME Code Case N-532, Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission

3.1.98 NEI 94-01, Revision 0, dated July 26, 1995, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J

3.1.99 NRC Regulatory Guide 1.163, Performance-Based Containment Leak Test Program dated September 1995

3.1.100 Revised NRC Order EA-03-009, Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, dated February 20, 2004

3.1.101 NEI 07-07, Industry Ground Water Protection Initiative - Final Guidance Document

3.1.102 Engineering Transmittal ET-NAF-08-0034, Applicability of 10 CFR to NUHOMS Important to Safety Components Classified NSQ

3.1.103 NOD Audit Finding 07-12-01C

3.1.104 North Anna CR 324829

3.1.105 North Anna Power Station Waterworks Operation Permit No. 2109600 and North Anna Power Station Waterworks Operation Permit No. 2109610

3.1.106 North Anna CA 149585 (CR 352954), North Anna Environmental Program Audit

3.2 Commitment Documents

3.2.1 QA Audit No. N-84-25, Finding No. 3

3.2.2 QA Audit No. N-86-03, Finding No. 01

3.2.3 QA Audit No. S89-14, Finding S89-14-05

3.2.4 Stewart, W. L., Virginia Electric and Power Company, to Denton, Harold R. (Attn: Varga, Steven A.), U.S. Nuclear Regulatory Commission, Subject: Reporting Related to NPDES Permits: Surry Power Station Unit Nos. 1 and 2, March 28, 1983

3.2.5 Grace, J. Nelson, U.S. Nuclear Regulatory Commission, to Stewart, W. L., Virginia Electric and Power Company, Subject: Report Nos. 50-280/86-23, 50-281/86-23, 50-338/86-22, and 50-339/86-22, December 11, 1986

3.2.6 Cruden, D. S., Virginia Electric and Power Company, to U.S. Nuclear Regulatory Commission, Subject: North Anna Power Station Units 1 and 2, NRC Inspection Report Nos. 50-338/87-38 and 50-339/87-38, Reply to a Notice of Violation, April 21, 1988, Serial No. 88-136A

3.2.7 NRC Generic Letter 91-02, Reporting Mishaps Involving LLW Forms Prepared for Disposal, December 28, 1990

3.2.8 Urquhart, George O'N., Commonwealth of Virginia Department of Emergency Services, to Cox, F. M., Virginia Power, February 3, 1988

3.2.9 Cartwright, W. R., to Slayton, Addison E., State Department of Emergency Services, December 1, 1989

3.2.10 Lough, W. Timothy, Ph.D., Commonwealth of Virginia State Corporation Commission, to Cunningham, John E., Virginia Power, December 8, 1989

3.2.11 Edmonds, Larry L., to SNSOC Secretary, Subject: Status of Corrective Action in Response to DR 89-2128, January 2, 1990

3.2.12 Berkow, Herbert N., NRC, to Stewart, W. L., Subject: Surry Units 1 and 2 and North Anna Units 1 and 2 - 10 CFR 50.72 Reporting Requirements, June 3, 1991

3.2.13 Crisp, Robert W., FERC, to Harrell, E. Wayne, June 18, 1991

3.2.14 Crisp, Robert W., FERC, to Cartwright, W. R., August 2, 1991

3.2.15 Crisp, Robert W., FERC, to Harrell, E. Wayne, August 20, 1991

3.2.16 O'Hanlon, J.P., Virginia Power, to Slayton, Addison E., Virginia Department of Emergency Services, February 4, 1994

3.2.17 Surry Deviation Report S-97-1634, Fire Protection Notification

3.2.18 CTS 02-97-2120 Item 001, DR N-96-2528, North Anna, 10 CFR 71.95 Criteria

3.2.19 CTS 02-97-2196 Item 001, DR N-97-0915, North Anna, AMSAC Operability Criteria

3.2.20 Site Specific Decommissioning Cost Studies update frequency has been established in North Carolina per a North Carolina Commission Final Order (at least every five years). In Virginia, rate case testimony committed to an update every four years.

3.2.21 Deviation S-1999-2415, Four-hour Notification Requirement Following Engineered Safety Feature (ESF) Actuation

3.2.22 S-2001-1779-R2, Emergency Telecommunications System (ETS) Functionality

3.2.23 N-2002-0047, Control of General License Devices

3.2.24 Surry Licensing Issue 51027, AMSAC Inoperability

3.2.25 Plant Issue (Deviation) S-2002-3048-R2, AMSAC Inoperability

3.2.26 Plant Issue (Deviation) S-2004-1119, Oil Spill Reporting

3.2.27 North Anna Technical Specifications Amendment Numbers 239 and 220/Surry Technical Specifications Amendment Numbers 240 and 239, Elimination of Requirements to Provide Monthly Operating Reports and Annual Occupational Radiation Exposure Reports, Issued March 22, 2005

3.2.28 Surry Technical Specification Amendment Numbers 244/243, Administrative Controls Changes (to support Consolidated QA Program); Plant Issues (Licensing Commitments) S-2004-3015 and S-2004-4898

3.2.29 Plant Issue (Licensing Commitment) S-2005-1715, VPP Annual Self-assessment Reporting Requirement

3.2.30 Surry Technical Specification Amendment Numbers 247/246, Revision of Accident Monitoring Instrumentation Listing, Allowed Outage Times, Requirements, and Surveillances; Plant Issue (Licensing Commitment) S-2005-3380

4.0 DEFINITIONS

4.1 Actuation, Component or System

A change in the state of a component or system (e.g., valve opens or closes, motor or pump starts, safety injection, auxiliary feedwater initiation).

4.2 Actuation, Invalid

An invalid actuation is one that does not meet the criteria for being valid and are initiated for reasons other than to mitigate the consequences of an event (e.g., as part of a planned evolution, with the system properly removed from service, or after the safety function has already been completed). Invalid actuations include instances where instrument drift, spurious signals, human error, or other invalid signals caused actuation (e.g., jarring a cabinet, an error in the use of jumpers or lifted leads, an error in the actuation of switches or controls, equipment failure, radio frequency interference).

4.3 Actuation, Valid

Actuation resulting from an intentional manual initiation or from a signal that was initiated in response to actual plant conditions or parameters satisfying the requirements for initiation, unless part of a preplanned test.

4.4 Adverse Condition (Nuclear Electric Insurance Limited)

- A condition which, if allowed to continue uncorrected, could:
- Cause physical damage in excess of $500,000, or
- Result in an outage longer than three weeks

or

- Transformer tests indicate:
  - •• Any gas that exceeds ANSI/IEEE C57.104-1978 probability norms by 10 percent or more
  - •• Total dissolved combustible gases greater than 700 ppm

4.5 Basic Component (10 CFR 21)

4.5.1 A plant structure, system, component, or part thereof that affects its safety function necessary to assure the:

- Integrity of the reactor coolant pressure boundary
- Capability to shut down the reactor and maintain it in a safe shutdown condition, or
- Capability to prevent or mitigate the consequences of an accident which could result in potential off-site exposures comparable to those referred to in 10 CFR 50.67(b)(2) or 10 CFR 100.11

Basic component includes safety-related design, analysis, inspection, testing, fabrication, replacement parts, or consulting services (whether performed by the component supplier or others) that are associated with component hardware. Commercial grade items are not basic components until after dedication (i.e., when dedicated for use as a basic component).

4.5.2 Basic Component - Independent Spent Fuel Storage Installation (ISFSI) - When applied to 10 CFR 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste”, basic component means a structure, system, or component, or part thereof that affects their safety function, that is directly procured by the licensee of a facility or activity subject to the regulations in 10 CFR 21 and in which a defect or failure to comply with any applicable regulation in 10 CFR Chapter I, order, or license issued by the Commission could create a substantial safety hazard.

ISFSI NUHOMS NSQ items which are vendor Transnuclear Quality Category B items (Reference 3.1.102) are basic components which require the application of 10 CFR 21. (Reference 3.1.103).

4.6 Bypassing (VPDES Permit)

Intentional diversion of waste streams from any portion of a treatment works.

4.7 Byproduct Material

Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or using special nuclear material.

4.8 Condition That Affects the Safety of a Project (18 CFR 12.10)

Any condition, event, or action at Lake Anna Dam which might compromise the safety, stability, or integrity of any project work, or the ability of any project work to function safely for its intended purposes, including navigation, water power development, or other beneficial public uses; or which might otherwise adversely affect life, health, or property. Conditions affecting the safety of a project include, but are not limited to:

- Unscheduled rapid draw-down of impounded water
- Failure of any facility that controls the release or storage of impounded water, such as a gate, a valve, or an emergency power supply
- Failure or unusual movement, subsidence, or settlement of any part of the dam or the adjacent areas
- Unusual concrete deterioration or cracking, including development of new cracks or the lengthening or widening of existing cracks
- Piping, slides, or settlements of materials in any dam, abutment, dike, or embankment
- Significant slides or settlements of materials in areas adjacent to the lake
- Significant damage to slope protection
- Unusual instrumentation readings
- New seepage or leakage or significant gradual increase in pre-existing seepage or leakage
- Sinkholes
- Significant instances of vandalism or sabotage
- Natural disasters, such as floods or earthquakes
- Any other signs of instability of any project work

4.9 Cut-off Level (10 CFR 26)

The value set for designating a fitness for duty test result as positive.

4.10 Defect (10 CFR 21)

- A deviation (see Subsection 4.11) in a basic component delivered to the Station for use, if, on the basis of evaluation, the deviation could create a substantial safety hazard

  or

- Installation, use, or operation of a basic component that contains a deviation as defined above

  or

- A condition or circumstance that involves a basic component that could contribute to exceeding a safety limit, as defined in the Technical Specifications

4.11 Deviation (10 CFR 21)

As used to define Defect, a departure from technical requirements included in a procurement document.

4.12 Discovery (10 CFR 21)

Completion of the documentation first identifying the existence of a defect or failure to comply potentially associated with a substantial safety hazard (the date the Shift Manager signs a Condition Report).

4.13 Discovery Date

The date an event or situation becomes known. If a Condition Report is dispositioned as not reportable, but subsequently determined to be reportable, the date the Condition Report is redispositioned as reportable.

4.14 Emergency (State Department of Environmental Quality (Air))

A sudden, unexpected event that could result in a safety hazard if not immediately attended to, is necessary to prevent equipment damage, or is necessary to avoid imposing unreasonable financial burden.

4.15 Emergency Response Facilities (ERFs)

ERFs include:

- Technical Support Center (TSC)
- Local Emergency Operations Center (LEOF)
- Central Emergency Operations Facility (CEOF)
- Control Room (CR)
- Operational Support Center (OSC)

4.16 Engineered Safety Feature (ESF) [Commitment 3.2.21]

A function (excluding actuation interlocks) listed in:

- North Anna Technical Specification Table 3.3.2 - 1; UFSAR Chapter 7.3
- Surry Technical Specification Tables 3.7 - 2, 3, & 4; UFSAR Tables 7.5 - 1

4.17 Engineering Judgment

A documented engineering analysis or a documented, verifiable statement by a technically qualified individual.

4.18 Evaluation (10 CFR 21.21)

The process of determining whether a defect could create a substantial hazard, or determining whether a failure to comply is associated with a substantial safety hazard.

4.19 Event Date

The date that an event or situation occurred. If an event date is unknown, the discovery date is used as the event date. For testing that is conducted later than the required time, it should be assumed that the discrepancy occurred at the time the testing was required unless there is firm evidence to indicate that it occurred at a different time.

4.20 Extremities

Hands and forearms; feet and ankles.

4.21 Failure to Comply (10 CFR 21.21)

Condition of failing to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of NRC relating to a substantial safety hazard.

4.22 Fitness for Duty Event, Significant (10 CFR 26)

- The sale, use, or possession of illegal drugs within the protected area
- An act by a person licensed in accordance with 10 CFR 55 to operate a power reactor, or by supervisory personnel subject to 10 CFR 26, that:
  - •• Involves the sale, use, or possession of a controlled substance
  - •• Results in a confirmed positive test
  - •• Involves use of alcohol within the protected area
  - •• Results in a determination of unfitness for scheduled work due to the consumption of alcohol

4.23 Incident (Nuclear Electric Insurance Limited)

- An event that results in physical damage to property that exceeds, or is expected to exceed, $100,000
- A fire that involves activation or malfunction of a fixed fire extinguishing or detection system

4.24 Licensed Material

Source material, special nuclear material, and by-product material.

4.25 Navigable Waters (40 CFR 110)

Waters of the United States, including the territorial seas. The term includes:

a. All waters that are currently used, were used in the past, or may be susceptible to use in interstate or foreign commerce, including all waters that are subject to the ebb and flow of the tide.

b. Interstate waters, including interstate wetlands.

c. All other waters such as intrastate lakes, rivers, streams (including intermittent streams), mudflats, sandflats, and wetlands, the use, degradation, or destruction of which would affect or could affect interstate or foreign commerce including any such waters:

   (1) That are or could be used by interstate or foreign travelers for recreational or other purposes.

   (2) From which fish or shellfish are or could be taken and sold in interstate or foreign commerce.

   (3) That are used or could be used for industrial purposes by industries in interstate commerce.

d. All impoundments of waters otherwise defined as navigable waters under this section.

e. Tributaries of waters identified in paragraphs a. through d., including adjacent wetlands.

f. Wetlands adjacent to waters identified in paragraphs a. through e., provided that waste treatment systems (other than cooling pounds meeting the criteria of this paragraph) are not waters of the United States.

Navigable waters at North Anna begin at the confluence of North Anna River with Pamunkey River, and at Surry begin at the end of the discharge canal concrete liner.

4.26 Notification (To Notify)

Conveyance of (to convey) required or pertinent information, often in response to certain events, and typically within a specified time after an event. May be oral (e.g., telephone conversation) or written (e.g., a facsimile, a report, a posted document) unless specifically indicated otherwise (e.g., mandatory use of the telephone).

4.27 Notification, Immediate

Communication initiated without administrative or circumstantial delay (subordinate to ensuring the Station is in a safe condition and preserving personnel safety).

4.28 Report

A document, often in a required format with specified content. The verb form, “to report,” is synonymous with “to notify.” See also: Notification.

4.29 Reportable

Having pre-determined attributes that require notification of someone outside Dominion.

4.30 Restricted Area

An area to which access is controlled to protect individuals against undue risks from exposure to radiation and radioactive materials.

4.31 Sheen (40 CFR 110)

An iridescent appearance on the surface of water.

4.32 Sludge (40 CFR 110)

An aggregate of oil or oil and other matter of any kind in any form other than dredged spoil that has a combined specific gravity equivalent to or greater than water.

4.33 Source Material

Uranium or thorium, or any combination thereof, in any physical form; or ores that contain by weight one-twentieth of one percent or more of uranium, thorium, or any combination thereof. Does not include special nuclear material.

4.34 Special Nuclear Material (SNM)

- Any material consisting of plutonium, uranium 233, uranium enriched in the isotope 233 or in the isotope 235
- Any other material that NRC, pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended, determines to be SNM, excluding source material or any material artificially enriched
- Any other material that pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended, has been determined to be SNM

4.35 Special Report

A report, in letter format, required by Technical Specifications or station procedures that describes events or situations judged by Dominion to warrant a written communication (e.g., Subsection 6.29).

4.36 State Waters

All water, on the surface and under the ground, wholly or partially within or bordering the Commonwealth of Virginia, or within its jurisdiction.

4.36.1 For purposes of reportability, Dominion does not classify the discharge canal as State Waters. The permanent oil boom in the discharge canal, near the first Waste Heat Treatment Facility lagoon, marks the end of the discharge canal. **(North Anna)**

4.36.2 For purposes of reportability, Dominion does not classify the cooling water discharge canal as State Waters. The permanent oil boom marks the end of the discharge canal. **(Surry)**

4.37 Substantial Safety Hazard (10 CFR 21—NUREG-0302)

A loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility regulated by the NRC under parts 30, 40, 50, 52, 60, 61, 63, 70, 71, or 72.

4.38 Transport (10 CFR 73)

Any land, sea, or air conveyance or modules for these conveyances such as rail cars or standardized cargo containers.

5.0 RESPONSIBILITIES

5.1 Asbestos Removal Contractors

Asbestos removal contractors are responsible for:

5.1.1 Preparing and, when required, submitting asbestos notification forms to the State Department of Labor and Industry; see Step 6.27.3.b.

5.1.2 Submitting asbestos notification forms to Electric Environmental Services; see Step 6.27.3.b.

5.2 Director Nuclear Station Safety and Licensing

The Director Nuclear Station Safety and Licensing is responsible for:

5.2.1 Notifying NRC of significant fitness for duty events (responsibility shared with the Plant Manager (Nuclear)); see Step 6.3.6.b.

5.2.2 Notifying the Director NL&OS, the Manager Nuclear Oversight, and the NRC Resident Inspector of events reportable in accordance with 10 CFR 50.72; ensuring that the MSRC receives a copy of associated reports sent to NRC (responsibilities shared with Plant Manager (Nuclear)); see Subsection 6.3.

5.2.3 Notifying the Senior Vice President Nuclear Operations and the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event (responsibility shared with Site Vice President, Plant Manager (Nuclear), and Shift Manager); see Step 6.27.2.a.

5.2.4 Notifying Corporate Risk Management if INPO downgrades the Station to Category 5, suspends Dominion membership, or upgrades the Station to Category 1 (shared with Site Vice President, Plant Manager (Nuclear), and Director NL&OS); see Step 6.28.5.

5.2.5 Notifying Corporate Risk Management if the Station operating license is revoked or suspended (shared with Site Vice President, Plant Manager (Nuclear), and Director NL&OS); see Step 6.28.6.a.

5.2.6 Approving Emergency Plan activation reports (responsibility shared with Site Vice President and Plant Manager (Nuclear)); see Steps 6.3.5.b.2.. and 6.3.7.b.

5.2.7 Approving reports of changes, tests, and experiments; see Steps 6.10.6, and 6.14.3.

5.2.8 Reviewing reports for instances of primary coolant activity exceeding limits; see Step 6.24.6. **(Surry)**.

5.2.9 Reviewing reports of changes in discharge or management of pollutants; see Step 6.27.3.j.

5.2.10 Reviewing reports regarding wastewater facility operators; see Step 6.27.3.m.

5.2.11 Approving quarterly (NRC/INPO/WANO) data prior to submittal to applicable regulatory agencies.

5.2.12 Approving reports of individual monitoring; see Step 6.6.7.

5.2.13 Approving radiological effluent release reports; see Step 6.10.3.

5.2.14 Approving various reports to the DEQ, as required, as one of the station's Authorized Signatories; see Steps 6.27.3.j. and 6.27.3.m.

5.2.15 Approving radiological environmental operating reports; see Steps 6.23.9 and 6.24.8.

5.2.16 Approving reports of Technical Specification (TS) bases changes; see Steps 6.23.15 and 6.24.16.

5.3 Plant Manager (Nuclear)

The Plant Manager (Nuclear) is responsible for:

5.3.1 Notifying NRC of significant fitness for duty events (responsibility shared with the Director Nuclear Station Safety and Licensing); see Step 6.3.6.b.

5.3.2 If the Director Nuclear Station Safety and Licensing is unavailable, notifying the Director NL&OS and the NRC Resident Inspector of events reportable in accordance with 10 CFR 50.72 and ensuring that the MSRC receives a copy of associated reports sent to NRC; see Subsection 6.3.

5.3.3 Notifying the Senior Vice President Nuclear Operations and the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event (responsibility shared with Site Vice President, Director Nuclear Station Safety and Licensing, and Shift Manager); see Step 6.27.2.a.

5.3.4 Notifying Corporate Risk Management if INPO downgrades the Station to Category 5, suspends Dominion membership, or upgrades the Station to Category 1 (shared with Site Vice President, Director Nuclear Station Safety and Licensing, and Senior Vice President Nuclear Operations); see Step 6.28.5.

5.3.5 Notifying Corporate Risk Management if the Station operating license is revoked or suspended (shared with Site Vice President, Director Nuclear Station Safety and Licensing, and Senior Vice President Nuclear Operations); see Step 6.28.6.a.

5.3.6 Approving Emergency Plan activation reports (responsibility shared with Site Vice President and Director Nuclear Station Safety and Licensing); see Steps 6.3.5.b.2.. and 6.3.7.b.

5.3.7 Reviewing reports of changes in discharge or management of pollutants; see Step 6.27.3.j.

5.3.8 Reviewing reports regarding wastewater facility operators; see Step 6.27.3.m.

5.3.9 Approving various reports to the DEQ, as required, as one of the station's Authorized Signatories; see Steps 6.27.3.j. and 6.27.3.m.

5.4 Auxiliary Boiler Operator

The auxiliary boiler operator is responsible for notifying Electric Environmental Services of certain boiler operation; see Step 6.27.3.f. **(Surry)**

5.5 Director Corporate Accounting

The Director Corporate Accounting is responsible for:

5.5.1 Forwarding copies of Dominion Annual Reports to the Senior Vice President Nuclear Operations; see Step 6.10.8.a.

5.5.2 Preparing reports to guarantee deferred liability insurance policy payment; see Step 6.17.4.a.

5.6 Director Operations Support

The Director Operations Support is responsible for:

5.6.1 Reviewing reports of planned special exposures; see Step 6.6.5.a.

5.6.2 Reviewing notifications for first use of low-level waste packages; see Step 6.13.2.b.

5.7 Director Nuclear Protection Services and Emergency Preparedness

The Director Nuclear Protection Services and Emergency Preparedness is responsible for:

5.7.1 Preparing the FERC Regional Engineer notifications of significant changes in upstream or downstream circumstances affecting the North Anna Hydroelectric Project Emergency Action Plan; see Step 6.18.4.a. **(North Anna)**

5.7.2 Revising the North Anna Hydroelectric Project Emergency Action Plan to incorporate Independent Consultant Reports; see Step 6.18.4.b. **(North Anna)**

5.7.3 Submitting Emergency Plan activation reports; see Steps 6.3.5.b.3. and 6.3.7.c.

5.7.4 Submitting Early Warning System Availability reports; see Step 6.27.2.c.

5.7.5 Preparing North Anna Hydroelectric Project Emergency Action Plan adequacy review submittal letters; see Step 6.18.4.c. **(North Anna)**

5.7.6 Preparing North Anna Hydroelectric Project Emergency Action Plan test exercise summary and critique submittal letters; see Step 6.18.5.b. **(North Anna)**

5.7.7 Initiating submittal of emergency plans revised in accordance with 10 CFR 50.54(q); see Step 6.10.5.c.

5.7.8 Notifying the Virginia Department of Emergency Management and Virginia Department of Health regarding voluntary reporting for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.8 Environmental Compliance Coordinator

The Environmental Compliance Coordinator is responsible for:

5.8.1 Notifying the State Department of Environmental Quality (Water) and, as appropriate, the National Response Center of oil releases (shared with the Director Electric Environmental Services); see Step 6.3.2.d. **(North Anna)**

5.8.2 Notifying appropriate agencies or individuals of hazardous material releases (shared with the Director Electric Environmental Services); see Step 6.3.2.e. **(North Anna)**

5.8.3 Notifying the State Department of Environmental Quality (Water) of VPDES permit violations (shared with the Director Electric Environmental Services); see Step 6.3.2.f.

5.8.4 Notifying Electric Environmental Services of excess smoke releases (shared with Shift Manager); see Step 6.3.4.b.

5.8.5 Notifying Electric Environmental Services when hazardous waste shipment manifests are not returned on time; see Step 6.20.7.b.

5.8.6 Notifying the Local Emergency Planning Coordinator and initiating changes to this procedure if the Station has an extremely hazardous substance in an amount greater than its threshold planning quantity; see Step 6.20.9

5.8.7 Notifying Electric Environmental Services if an osprey nest is disturbed or a raptor is injured or killed by electrocution; see Step 6.22.4

5.8.8 Notifying Electric Environmental Services when manifests for off-site waste shipments that contain asbestos are not returned; see Step 6.27.3.b.

5.8.9 Notifying Electric Environmental Services of pollution control equipment malfunctions; see Step 6.27.3.e.

5.8.10 Submitting new or revised Material Safety Data Sheets; see Step 6.20.10.a.

5.8.11 Preparing and submitting specified reports to the State Department of Environmental Quality (Water); see Step 6.27.3.i.

5.8.12 Preparing and submitting Groundwater Pumpage and Use reports; see Step 6.27.3.k. **(Surry)**

5.8.13 Preparing and submitting Operation Report Meter Readings reports; see Step 6.27.4.a. **(North Anna)**

5.8.14 Preparing and submitting Sewage Treatment Plant Operation reports; see Step 6.27.4.b. **(Surry)**

5.8.15 Preparing and submitting Waterworks Operation reports; see Step 6.27.4.c. **(Surry)**

5.8.16 Preparing and maintaining a record of oil releases to the ground up to 25 gallons; see Step 6.3.2.d. and its footnote 1.

5.8.17 Preparing DOT forms for transportation-related hazardous material events; see Step 6.21.1.a.

5.8.18 Preparing notifications of certain events related to the VPDES permit; see Step 6.26.1.b.

5.8.19 Preparing environmental operating reports; see Step; 6.26.3.a. **(North Anna)**

5.8.20 Preparing and submitting reports of water withdrawals; see Step 6.27.3.r.

5.8.21 Preparing and submitting Tier II information forms; see Step 6.20.10.b.

5.8.22 Reviewing follow-up reports for unusual or important environmental events; see Step 6.26.2.c. **(North Anna)**

5.8.23 Reviewing reports of changes in discharge or management of pollutants; see Step 6.27.3.j.

5.8.24 Reviewing reports regarding wastewater facility operators; see Step 6.27.3.m.

5.8.25 Notifying Electric Environmental Services of an inadvertent discovery of archeological, historical, or other cultural resource; see Step 6.1.1.b.

5.8.26 Notifying Electric Environmental Services of the failure to repair a commercial or comfort refrigeration unit containing greater than 50 pounds of refrigerant within 30 days; see Step 6.20.2.

5.9 Fitness for Duty Administrator, Station

The Fitness for Duty Administrator (Station) is responsible for:

5.9.1 Ensuring NRC is notified of significant fitness for duty events or NRC personnel believed to be unfit for duty; see Steps 6.3.2.c. and 6.8.1.

5.9.2 Preparing and distributing Attachment 4 for significant fitness for duty events; see Step 6.8.1.

5.9.3 Preparing fitness for duty event follow-up reports.

5.10 Fitness for Duty Administrator, Corporate

The Fitness for Duty Administrator (Corporate) is responsible for notifying applicable Station Fitness for Duty Administrators of significant fitness for duty events; see Step 6.8.1.a.

5.10.1 Preparing reports of false positive or false negative test results.

5.10.2 Preparing reports of unsatisfactory laboratory performance.

5.10.3 Preparing fitness for duty event follow-up reports.

5.10.4 Preparing annual program reports to NRC; see Step 6.8.4.a.

5.11 Fitness for Duty Program Manager

The Fitness for Duty Program Manager is responsible for:

5.11.1 Reviewing and forwarding fitness for duty program false positive test result reports.

5.11.2 Approving unsatisfactory performance testing result reports.

5.11.3 Reviewing program assessment reports; see Step 6.8.4.b.

5.12 Lake Anna Dam Operator

The Lake Anna Dam Operator is responsible for:

5.12.1 Notifying the Shift Manager of deaths or serious injuries at or near the dam; see Step 6.3.2.h.

5.12.2 Notifying the Shift Manager of conditions that affect the safety of the dam and its associated works; see Step 6.3.2.i.

5.13 Director Electric Environmental Services

The Director Electric Environmental Services is responsible for:

5.13.1 Notifying the State Department of Environmental Quality (Air) of excess smoke releases; see Step 6.3.4.b.

5.13.2 Submitting asbestos notification forms to the State Department of Environmental Quality (Air); see Step 6.27.3.b.

5.13.3 Notifying the State Department of Environmental Quality (Air) when manifests for off-site waste shipments that contain asbestos are not returned; see Step 6.27.3.c.

5.13.4 Notifying the State Department of Environmental Quality (Air) of pollution control equipment malfunctions; see Step 6.27.3.e.

5.13.5 Notifying the State Department of Environmental Quality (Air) regional inspector of auxiliary boiler operation; see Step 6.27.3.f. **(Surry)**

5.13.6 Notifying the State Department of Environmental Quality (Water) and, as appropriate, the National Response Center of oil releases (shared with the Environmental Compliance Coordinator); see Step 6.3.2.d. **(North Anna)**

5.13.7 Notifying appropriate agencies or individuals of hazardous material releases (shared with the Environmental Compliance Coordinator); see Step 6.3.2.e. **(North Anna)**

5.13.8 Notifying the State Department of Environmental Quality (Water) of VPDES permit violations (shared with the Environmental Compliance Coordinator); see Step 6.3.2.f.

5.13.9 Notifying the State Department of Environmental Quality (Water) of unplanned bypasses; see Step 6.3.6.f.

5.13.10 Notifying the Virginia Department of Environmental Quality regarding voluntary reporting for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.13.11 Notifying Nuclear Licensing and Operations Support of State Department of Environmental Quality (Water) approval of VPDES permit changes; see Step 6.27.3.o. **(Surry)**

5.13.12 Coordinating proposed VPDES permit changes with Nuclear Licensing and Operations Support; see Step 6.27.3.m. **(Surry)**

5.13.13 Submitting information packages to the EPA Regional Administrator; see Step 6.20.4.

5.13.14 Preparing and submitting hazardous waste reports; see Step 6.20.7.a.

5.13.15 Preparing and submitting exception reports when hazardous waste shipment manifests are not returned on time; see Step 6.20.7.b.

5.13.16 Preparing, approving, and submitting follow-up reports for hazardous material releases; see Step 6.22.3.

5.13.17 Preparing and submitting reports of changes in discharge or management of pollutants; see Step 6.27.3.j.

5.13.18 Preparing and submitting amendments or revisions to the Oil Discharge Contingency Plan; see Step 6.27.3.l.

5.13.19 Preparing and submitting reports regarding wastewater facility operators; see Step 6.27.3.m.

5.13.20 Preparing, approving, and submitting follow-up reports for VPDES permit noncompliances; see Step 6.27.3.n.

5.13.21 Preparing and submitting reports for tank-bottom-water pump and haul activities; see Step 6.27.3.p. **(Surry)**

5.13.22 Preparing and submitting temperature monitoring program reports; see Step 6.27.3.q. **(North Anna)**

5.13.23 Reviewing notifications of certain events related to the VPDES permit; see Step 6.26.1.b. **(North Anna)**

5.13.24 Preparing follow-up reports for unusual or important environmental events; see Step 6.26.2.b. **(North Anna)**

5.13.25 Reviewing environmental operating reports; see Step 6.26.3.a.

5.13.26 Notifying the EPA headquarters in accordance with 40 CFR 82.166(N)(i) of the failure to repair a commercial or comfort refrigeration unit containing greater than 50 pounds of refrigerant within 30 days; see Step 6.20.2.

5.14 Director Human Resources Employee Services/Safety & Health

The Director Human Resources Employee Services/Safety & Health is responsible for:

5.14.1 Notifying the FFD Program Manager of false positive test results.

5.14.2 Notifying the FFD Program Manager of unsatisfactory laboratory test performance.

5.14.3 Reviewing reports of false positive test results and other unsatisfactory laboratory test performance.

5.15 Director Nuclear Analysis and Fuel (NAF)

The Director NAF is responsible for:

5.15.1 Preparing and submitting notifications, and notifying of schedule changes, for spent fuel transport through a state; see Step 6.15.1.

5.15.2 Notifying the receiver of impending shipment of SNM of low strategic significance; see Step 6.15.2.a.

5.15.3 Notifying the shipper of receipt of SNM of low strategic significance; see Step 6.15.2.b.

5.15.4 Preparing, approving, and submitting DOE/NRC material balance reports (Director NLOS may approve as an alternate); see Step 6.16.2.

5.15.5 Preparing and submitting DOE/NRC material transaction reports (Director NLOS may approve as an alternate); see Step 6.16.3.

5.15.6 Preparing revised assessments for changes in projected values of RT_{PTS}; see Step 6.10.7.a.

5.15.7 Preparing proposed programs to satisfy 10 CFR 50, App. G; see Step 6.10.14.a.

5.15.8 Preparing summary reports of plant startup and power escalation testing; see Step 6.24.4.a. **(Surry)**

5.15.9 Preparing a core operating limits report for each refueling; see Steps 6.23.8.a. and 6.24.7.

5.15.10 Coordinating preparation of reports for fracture toughness test specimens; see Step 6.10.15.a.

5.15.11 Reviewing notifications for first use of spent nuclear fuel packages; see Step 6.13.2.c.

5.16 Director Corporate Engineering

The Director Corporate Engineering is responsible for:

5.16.1 Notifying Electric Environmental Services and Licensing (Station) of planned modifications to the dam; see Step 6.18.3.a. **(North Anna)**

5.16.2 Notifying the Shift Manager of emergency corrective measures identified by an independent consultant; see Step 6.18.6.b. **(North Anna)**

5.16.3 Preparing periodic containment leakage test reports; see Step 6.10.17.

5.16.4 Preparing reports of planned modifications to Lake Anna Dam; see Step 6.18.3.b. **(North Anna)**

5.16.5 Obtaining and forwarding independent consultant reports; see Step 6.18.7. **(North Anna)**

5.16.6 Preparing inservice inspection reports; see Steps 6.23.4 and 6.24.5.

5.17 Director Nuclear Licensing and Operations Support (NL&OS)

The Director NL&OS is responsible for:

5.17.1 Notifying appropriate corporate organizations for events reportable in accordance with 10 CFR 50.72; see Step 6.3.1.c.

5.17.2 Notifying NRC of VPDES permit changes; see Steps 6.26.1.b. and 6.27.3.o.

5.17.3 Notifying Corporate Risk Management if INPO downgrades a Station to Category 5, or suspends Dominion membership (shared with Site Vice President and Directors); see Step 6.28.5.

5.17.4 Notifying Corporate Risk Management of significant NRC actions affecting operating licenses or permission to operate (shared with Site Vice President and Directors); see Step 6.28.6.a.

5.17.5 Notifying the NRC Region II Branch Chief and Project Manager regarding voluntary reporting for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.17.6 Posting notices to workers; see Steps 6.5.1.c. and 6.7.1.a.

5.17.7 Reviewing and submitting reports of planned special exposures; see Step 6.6.5.

5.17.8 Reviewing and submitting changes to the QA Topical Report; see Step 6.10.5

5.17.9 Reviewing and submitting revisions to the Emergency Plan; see Step 6.10.5.c.

5.17.10 Reviewing, approving and submitting reports of insurance or financial security; see Step 6.10.5.d.

5.17.11 Preparing and submitting reports of program to manage fuel after the reactor operating license expires; see Step 6.10.5.f.

5.17.12 Preparing and submitting reports of bankruptcy petitions; see Step 6.10.5.g.

5.17.13 Reviewing and submitting revised assessments for changes in projected values of RT_{PTS}; see Step 6.10.7.

5.17.14 Reviewing and submitting proposed programs to address fracture toughness requirements; see Step 6.10.14.

5.17.15 Reviewing and submitting reports for exceeding off-site radiological limits; see Step 6.10.16.

5.17.16 Reviewing and submitting notifications for first use of radioactive material packages; see Step 6.13.2.

5.17.17 Reviewing and submitting notices for certain events that involve bodily injury or property damage; see Step 6.17.1.

5.17.18 Reviewing and submitting notices to indicate material change in, renewal or replacement of liability insurance policies; see Steps 6.17.2 and 6.17.3.

5.17.19 Reviewing, approving and submitting reports to guarantee deferred liability insurance policy payment; see Step 6.17.4.

5.17.20 Reviewing and submitting DOT Forms F 5800; see Step 6.21.2.

5.17.21 Reviewing and submitting reports of planned removal or significant changes to equipment that controls radioactivity in effluents; see Step 6.23.5. **(North Anna)**

5.17.22 Reviewing and submitting notifications of certain events related to the VPDES permit; see Step 6.26.1.b. **(North Anna)**

5.17.23 Sending NRC documents and correspondence that must be posted at the Station to Licensing (Station); see Step 6.5.1.e.

5.17.24 Reviewing reports for exceeding the limits in the Offsite Dose Calculation Manual; see Step 6.10.16.a.

5.17.25 Reviewing, approving and submitting fitness for duty program assessment reports; see Step 6.8.4.c.

5.17.26 Submitting revisions to the Security Plan per 10 CFR 50.54(p)(1); see Step 6.10.5.b.

5.17.27 Submitting revisions to the Emergency Plan per 10 CFR 50.54(q) or 10 CFR 72.44(f); see Step 6.10.5.c.

5.17.28 Submitting Dominion Annual Reports; see Step 6.10.8.b.

5.17.29 Submitting 10 CFR 50, Appendix H reports; see Step 6.10.15.b.

5.17.30 Submitting notices of material changes in proof of financial protection or other financial information to comply with 10 CFR 140; see Step 6.17.2.b.

5.17.31 Submitting startup reports; see Step 6.24.4.d.

5.17.32 Submitting follow-up reports for notifications of license violations; see Step 6.23.6.b. **(North Anna, Unit 2)**

5.17.33 Approving and submitting core operating limits reports; (Director NSS&L may approve as an alternate) see Steps 6.23.8.d. and 6.24.7.

5.17.34 Submitting copies of VPDES permit violation reports; see Steps 6.26.1.a. and 6.27.3.n.

5.17.35 Submitting copies of proposed VPDES permit changes; see Steps 6.26.1.b. and 6.27.3.o.

5.17.36 Preparing, obtaining Senior Financial Officer responsible for Decommissioning Trust Fund concurrence and submitting decommissioning reports; see Step 6.10.13.

5.17.37 Preparing and submitting to Corporate Treasury, Site Specific Cost Studies for decommissioning; see Steps 6.10.13 and 6.27.1.c.

5.17.38 Ensuring annual updates to the Decommissioning Trust Fund Status are submitted by Corporate Treasury; see Step 6.27.1.

5.17.39 Approving and submitting ECCS Evaluation Model changes report; see Step 6.10.4.

5.18 Director Nuclear Oversight

The Director Nuclear Oversight is responsible for reviewing submittals of changes to the QA Topical Report; see Step 6.10.5.a.

5.19 Director Nuclear Training

The Director Nuclear Training is responsible for:

5.19.1 Reviewing notifications of change in operator or senior operator status; see Step 6.10.12.b.

5.19.2 Preparing and reviewing reports of simulator status; see Step 6.11.2.

5.20 Director Corporate Risk Management

The Director Corporate Risk Management is responsible for:

5.20.1 Notifying insurers of certain changes in INPO rating and membership; see Step 6.28.5.

5.20.2 Notifying insurers of operating license suspension or revocation; see Step 6.28.6.b.

5.20.3 Preparing insurance or financial security reports; see Step 6.10.5.d.

5.20.4 Preparing notices for certain events that involve bodily injury or property damage; see Step 6.17.1.a.

5.20.5 Preparing notices of material changes in proof of financial protection or other financial information to comply with 10 CFR 140; see Step 6.17.2.a.

5.20.6 Preparing notices to indicate renewal or replacement liability insurance policies; see Step 6.17.3.a.

5.21 EPIX (Equipment Performance Information Exchange) Coordinator

The EPIX Coordinator is responsible for determining whether component failures or malfunctions are reportable to EPIX; see Step 6.10.11.e. **[Commitment 3.2.3]**

5.22 Senior Vice President Nuclear

The Senior Vice President Nuclear is responsible for:

5.22.1 Reissuing Shift Supervisor Responsibility Directives; see Step 6.29.7.

5.22.2 Submitting reports of fitness for duty program false positive test results.

5.22.3 Submitting reports of unsatisfactory performance testing results.

5.22.4 Submitting letters that describe chemical test program changes.

5.22.5 Approving notifications for proposed programs to address fracture toughness requirements; see Step 6.10.14.c.

5.22.6 Approving notifications for first use of NRC pre-approved radioactive material packages; see Step 6.13.2.d.

5.22.7 Approving notifications of spent fuel transport through a state; see Step 6.15.1.c.

5.23 Shift Manager

The Shift Manager is responsible for:

5.23.1 Notifying the NRC Operations Center of specified events; see Steps 6.3.1.b., 6.3.4.a., and 6.3.6.a.

5.23.2 Notifying Dominion personnel of events reportable in accordance with 10 CFR 50.72; see Step 6.3.1.c.

5.23.3 Notifying Dominion personnel of potential oil discharge events; see Step 6.3.2.d.

5.23.4 Notifying the State Department of Environmental Quality (Water) and, as appropriate, the National Response Center and the U.S. Coast Guard, of oil releases; see Step 6.3.2.d. **(Surry)**

5.23.5 Notifying appropriate agencies or individuals of hazardous material releases; see Step 6.3.2.e. **(Surry)**

5.23.6 Notifying appropriate agencies of transportation-related hazardous material releases; see Step 6.3.2.g.

5.23.7 Notifying FERC and initiating deviation reports for deaths or serious injuries at or near the dam; see Step 6.3.2.h. **(North Anna)**

5.23.8 Notifying FERC of conditions that affect the safety of the dam and its associated works; see Step 6.3.2.i. **(North Anna)**

5.23.9 Notifying the NRC Operations Center of specified events (responsibility shared with Site Vice President); see Step 6.3.3.

5.23.10 Notifying the Environmental Compliance Coordinator of smoke releases and notifying Electric Environmental Services of excess smoke releases (shared with Environmental Compliance Coordinator); see Step 6.3.4.b.

5.23.11 Notifying NRC of unusual or important events with potentially significant environmental impact; see Step 6.3.6.c. **(North Anna)**

5.23.12 Notifying the Supervisor Licensing (Station) of planned removal, and subsequent restoration of warning and safety devices from service at the main dam; see Step 6.18.8.a. **(North Anna)**

5.23.13 Notifying the Supervisor Licensing (Station) that voluntary communications are required for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.23.14 Notifying Station management of any event that may be of media significance; see Step 6.27.2.a.

5.23.15 Notifying the Senior Vice President Nuclear Operations and the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event (responsibility shared with Site Vice President, Plant Manager (Nuclear), Director, and Manager Nuclear Operations); see Step 6.27.2.a.

5.24 Shift Technical Advisor (STA)

The STA is responsible for notifying Station management of events reportable in accordance with 10 CFR 50.72; see Step 6.3.1.c.

5.25 Station Coordinator Emergency Preparedness

The Station Coordinator Emergency Preparedness is responsible for:

5.25.1 Preparing, and after approval, forwarding emergency plan activation reports; see Steps 6.3.5.b.1. and 6.3.7.a.

5.25.2 Ensuring that copies of the current North Anna Hydroelectric Project Emergency Action Plan are posted; see Step 6.18.5.a. **(North Anna)**

5.25.3 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.26 Site Vice President

The Site Vice President is responsible for:

5.26.1 Notifying the NRC Operations Center of specified events (responsibility shared with Shift Manager); see Step 6.3.3.

5.26.2 Designating an Director to notify NRC of significant fitness for duty events; see Step 6.3.6.b.

5.26.3 Notifying the Senior Vice President Nuclear Operations and the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event (responsibility shared with Directors, Manager Nuclear Operations, and Shift Manager); see Step 6.27.2.a.

5.26.4 Notifying Corporate Risk Management if INPO downgrades the Station to Category 5, suspends Dominion membership, or upgrades the Station to Category 1 (shared with Site Vice President, Director Nuclear Station Safety and Licensing, and Director NL&OS); see Step 6.28.5.

5.26.5 Notifying Corporate Risk Management if the Station operating license is revoked or suspended (shared with Site Vice President, Director Nuclear Station Safety and Licensing, and Director NL&OS); see Step 6.28.6.a.

5.26.6 Notifying the County Administrator regarding voluntary reporting for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.26.7 Approving and submitting reports for exceeding untreated liquid or gaseous waste discharge limits; see Step 6.10.16.b.

5.26.8 Approving Emergency Plan activation reports (responsibility shared with Directors); see Steps 6.3.5 and 6.3.7.b.

5.26.9 Approving Licensee Event Reports (LERs); see Step 6.10.11.c.

5.26.10 Approving follow-up reports for unusual or important environmental events; see Step 6.26.2.d. **(North Anna)**

5.26.11 Approving additional information reports for adverse conditions; see Step 6.28.2.e.

5.26.12 Approving discretionary special reports; see Subsection 6.29.

5.26.13 Reviewing certain reports prepared in accordance with 10 CFR 50.54; see Step 6.10.5.

5.26.14 Reviewing notifications of change in operator or senior operator status; see Steps 6.10.12.c. and 6.10.12.d.

5.26.15 Reviewing and approving immediate notification follow-up reports for conditions that affect the safety of Lake Anna Dam; see Step 6.18.1.d. **(North Anna)**

5.26.16 Reviewing and approving reports of planned modifications to Lake Anna Dam; see Step 6.18.3.c. **(North Anna)**

5.26.17 Reviewing and approving notification letters for removal and restoration of Lake Anna Dam safety devices; see Step 6.18.8.c. **(North Anna)**

5.26.18 Approving reports of deaths or serious injuries at or near the Lake Anna Dam; see Step 6.18.2.d. **(North Anna)**

5.26.19 Approving notifications of significant changes in upstream or downstream circumstances affecting the North Anna Hydroelectric Project Emergency Action Plan; see Step 6.18.4.a. **(North Anna)**

5.26.20 Approving submittals of North Anna Hydroelectric Project Emergency Action Plan adequacy reviews and North Anna Hydroelectric Project Emergency Action Plan revisions; see Step 6.18.4.c. **(North Anna)**

5.26.21 Approving submittals of North Anna Hydroelectric Project Emergency Action Plan test exercise summaries and critiques; see Step 6.18.5.c. **(North Anna)**

5.26.22 Reviewing follow-up reports for hazardous material releases; see Step 6.22.3.b.

5.26.23 Reviewing, approving and submitting environmental operating reports; see Step 6.26.3. **(North Anna)**

5.26.24 Approving and submitting notifications of change in operator or senior operator status; see Steps 6.10.12.e. and 6.10.12.f.

5.26.25 Submitting special reports related to Reactor Vessel Overpressure Mitigating System actuations; see Step 6.24.13.a. **(Surry)**

5.26.26 Submitting special reports related to inoperable explosive gas monitoring instrumentation; see Step 6.24.13.b. **(Surry)**

5.26.27 Submitting reports related to inoperable accident monitoring instrumentation; see Step 6.24.17 **(Surry)**

5.26.28 Submitting special reports related to waste gas holdup system oxygen concentration; see Step 6.24.13.c. **(Surry)**

5.26.29 Submitting special reports related to quadrant to average power tilt; see Step 6.24.13.d. **(Surry)**

5.26.30 Submitting copies to NRC (if NRC is not notified of an event) of reports provided to other agencies regarding unusual or important environmental events; see Step 6.26.2.e.

5.26.31 Approving steam generator tube reports (Vice President Nuclear Engineering may approve as an alternate); see Steps 6.23.7 and 6.24.14.

5.26.32 Approving reports for the Reactor Pressure Vessel Head inspection results; see Steps 6.23.14 and 6.24.14.

5.26.33 Approving inservice inspection reports (Director NSS&L or Director NLOS may approve as alternates); see Steps 6.23.4 and 6.24.5.

5.27 Facility Safety Review Committee (FSRC)

FSRC is responsible for:

5.27.1 Arbitrating disagreements on issues of reportability, as specified in PI-AA-200.

5.27.2 Approving Substantial Safety Hazard Evaluations, determining whether defects or failures to comply are potentially reportable, and notifying others as appropriate; see Step 6.7.2.

5.27.3 Approving revised assessments for changes in projected values of RT_{PTS}; see Step 6.10.7.c.

5.27.4 Approving startup reports; see Step 6.24.4.c. **(Surry)**

5.27.5 Approving special reports to NRC; see Steps 6.23.1 and 6.24.1.

5.27.6 Approving core operating limits reports; see Steps 6.23.8.b. and 6.24.7.

5.27.7 Reviewing certain reports submitted in accordance with 10 CFR 50.54; see Step 6.10.5.

5.27.8 Reviewing proposed programs to address fracture toughness requirements; see Step 6.10.14.b.

5.27.9 Reviewing reports for exceeding untreated liquid or gaseous waste discharge limits; see Step 6.10.16.b.

5.27.10 Reviewing periodic containment leakage test reports; see Step 6.10.17.d.

5.27.11 Reviewing and approving events reportable in accordance with 10 CFR 50.73; see Step 6.10.11.c. Submitting results to the Senior Vice President Nuclear Operations and MSRC; see Step 6.24.2. **(Surry)**

5.27.12 Reviewing Safety Limit Violation Reports; see Step 6.24.3 **(Surry)**

5.27.13 Reviewing special reports for Reactor Vessel Overpressure Mitigating System use to mitigate RCS pressure transients; see Step 6.24.13.a. **(Surry)**

5.27.14 Reviewing special reports related to inoperable explosive gas monitoring instrumentation; see Step 6.24.13.b. **(Surry)**

5.27.15 Reviewing reports related to inoperable accident monitoring instrumentation; see Step 6.24.17 **(Surry)**

5.27.16 Reviewing special reports related to waste gas holdup system oxygen concentration; see Step 6.24.13.c. **(Surry)**

5.27.17 Reviewing special reports related to quadrant to average power tilt; see Step 6.24.13.d. **(Surry)**

5.27.18 Reviewing discretionary reports to NRC; see Subsection 6.29.

5.28 Manager Nuclear Training

The Manager Nuclear Training is responsible for preparing notifications of changes in operator or senior operator license status; see Step 6.10.12.a.

5.29 Manager Nuclear Operations

The Manager Nuclear Operations is responsible for:

5.29.1 Notifying Station management of events reportable in accordance with 10 CFR 50.72; see Step 6.3.1.c.

5.29.2 Notifying Station management of potentially media significant events; see Step 6.27.2.a.

5.29.3 Notifying the Senior Vice President Nuclear Operations and the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event (responsibility shared with Site Vice President, Directors, and Shift Manager); see Step 6.27.2.a.

5.29.4 Preparing notifications of changes in operator or senior operator license status; see Step 6.10.12.b.

5.29.5 Reviewing reports of deaths or serious injuries at or near the dam; see Step 6.18.2.c. **(North Anna)**

5.29.6 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.30 Manager Radiological Protection

The Manager Radiological Protection is responsible for:

5.30.1 Notifying Director Operations Support of excess package contamination or radiation; see Step 6.3.2.b.

5.30.2 Notifying NRC and the final carrier of package excess surface contamination or excess radiation (shared with Supervisor Licensing (Station)) and initiating a Condition Report; see Step 6.3.2.b.

5.30.3 Preparing reports of planned special exposures; see Step 6.6.5.a.

5.30.4 Preparing reports of individual monitoring; see Step 6.6.7.a.

5.30.5 Preparing, approving, and submitting the annual reconciliation report to the National Source Tracking System; see Step 6.6.8.

5.30.6 Preparing radiological effluent release reports; see Steps 6.10.3.a. and 6.10.16.a.

5.30.7 Preparing notifications for first use of radioactive material packages; see Step 6.13.2.a.

5.30.8 Preparing reports of significant reductions in low-level waste packaging effectiveness or defects in low-level waste packaging; see Step 6.13.3.c.

5.30.9 Preparing reports of significant reductions in spent fuel packaging effectiveness or defects in spent fuel packaging; see Step 6.13.3.f.

5.30.10 Preparing radiological environmental operating reports; see Step 6.23.9.a.

5.30.11 Preparing special reports for mishaps involving low level waste forms; see Step 6.29.5.

5.30.12 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.31 Manager Nuclear Security

The Manager Nuclear Security is responsible for:

5.31.1 Providing clarification and guidance to Shift Managers for reportability determinations, as specified in PI-AA-200.

5.31.2 Providing clarification and guidance, as requested, to support Licensing (Station) preparation of safeguards LERs; see Step 6.10.11.d.

5.31.3 Preparing reports of changes to the Station physical security plan, training and qualification plan, or safeguards contingency plan; see Step 6.14.12.a.

5.31.4 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.32 Manager Site Engineering

The Manager Site Engineering is responsible for:

5.32.1 Supporting preparation of LERs; see Step 6.10.11.d.

5.32.2 Assisting with preparation of reports for discretionary reporting items; see Subsection 6.29.

5.32.3 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.33 Supervisor, Fuel Performance Analysis

The Supervisor, Fuel Performance Analysis is responsible for"

5.33.1 Approving nuclear material transaction reports; see Step 6.16.3.

5.33.2 Entering data into INPO's Consolidated Data Entry (CDE) System; see Subsection 6.30.

5.34 Supervisor ISI/IST/Materials Engineering (Station)

The Supervisor ISI/IST/Materials Engineering (Station) is responsible for:

5.34.1 Preparing steam generator tube reports; see Steps 6.23.7 and 6.24.14.

5.34.2 Preparing results of Reactor Pressure Vessel Head related nonvisual nondestructive (NDE) examinations; see Steps 6.23.14 and 6.24.15.

5.35 Supervisor ISI/DBD/UFSAR Engineering

The Supervisor ISI/DBD/UFSAR Engineering is responsible for:

5.35.1 Preparing inservice inspection reports; see Steps 6.23.4 and 6.24.5.

5.35.2 Preparing UFSAR updates for submittal to the NRC; see Step 6.10.9.

5.35.3 Preparing ISFSI FSAR updates for submittal to the NRC; see Step 6.14.5.

5.36 Supervisor Licensing (Station)

The Supervisor Licensing (Station) is responsible for:

5.36.1 Notifying the NRC Regional Office of failures related to safeguards and security, or failures to notify NRC of planned removal or significant change in radioactive effluent control equipment; see Step 6.3.6.d. **(North Anna)**

5.36.2 Submitting discretionary special reports; see Subsection 6.29.

5.36.3 Confirming fire suppression nonfunctionality notifications; see Step 6.3.6.a.

5.36.4 Posting notices to Station workers; see Steps 6.5.1 and 6.7.1.a.

5.36.5 Preparing and submitting reports of changes, tests, and experiments; see Step 6.10.6. **[Commitment 3.2.1]** and see Step 6.14.3.

5.36.6 Preparing, obtaining approvals, and submitting LERs; see Step 6.10.11.c.

5.36.7 Preparing and submitting immediate notification follow-up reports for conditions that affect the safety of Lake Anna Dam; see Step 6.18.1.b. **(North Anna)**

5.36.8 Preparing and submitting FERC Database reports for deaths or serious injuries at or near the dam; see Step 6.18.2.b. **(North Anna)**

5.36.9 Notifying the FERC Regional Engineer of significant changes in upstream or downstream circumstances affecting the North Anna Hydroelectric Project Emergency Action Plan; see Step 6.18.4.a. **(North Anna)**

5.36.10 Preparing and submitting notification letters for removal and restoration of Lake Anna Dam safety devices; see Step 6.18.8 **(North Anna)**.

5.36.11 Preparing follow-up reports for license violations; see Step 6.23.6.b. **(North Anna)**

5.36.12 Preparing reports for instances of primary coolant activity exceeding limits; see Step 6.24.6. **(Surry)**.

5.36.13 Preparing special reports for Reactor Vessel Overpressure Mitigating System use to mitigate RCS pressure transients; see Step 6.24.13.a. **(Surry)**

5.36.14 Preparing special reports related to inoperable explosive gas monitoring instrumentation; see Step 6.24.13.b. **(Surry)**

5.36.15 Preparing reports related to inoperable accident monitoring instrumentation; see Step 6.24.17 **(Surry)**

5.36.16 Preparing special reports related to waste gas holdup system oxygen concentration; see Step 6.24.13.c. **(Surry)**

5.36.17 Preparing special reports related to quadrant to average power tilt; see Step 6.24.13.d. **(Surry)**

5.36.18 Obtaining from the EPIX Coordinator a determination of EPIX reportability (to include in LERs) of equipment failures or malfunctions; see Step 6.10.11.e.

5.36.19 Notifying NRC and the final carrier of package excess surface contamination or excess radiation levels (shared with Manager Radiological Protection) and initiating a Condition Report; see Step 6.3.2.b.

5.36.20 Notifying the Department of Transportation of transport incidents that involve radioactive material; see Step 6.3.2.g.

5.36.21 Reviewing quarterly (NRC/INPO/WANO) data prior to submittal to applicable regulatory agencies; see Subsection 6.30.

5.36.22 Notifying the NRC Senior Resident Inspector and designated Dominion personnel regarding voluntary reporting for radioactive contamination of groundwater and radioactive spills and/or leaks; see Step 6.31.1.

5.36.23 Submitting steam generator tube reports to the NRC; see Steps 6.23.7 and 6.24.14.

5.36.24 Submitting reports to the NRC for Reactor Pressure Vessel Head inspection results; see Steps 6.23.14 and 6.24.14.

5.36.25 Reviewing and submitting reports of individual monitoring; see Step 6.6.7.

5.36.26 Reviewing and submitting radiological effluent release reports; see Step 6.10.3.

5.36.27 Submitting inservice inspection reports (Corporate Licensing may submit as an alternate); see Steps 6.23.4 and 6.24.5.

5.36.28 Reviewing and submitting radiological environmental operating reports; see Steps 6.23.9 and 6.24.8.

5.36.29 Preparing and submitting reports of Technical Specification (TS) bases changes; see Step 6.23.15.

5.36.30 Reviewing and submitting reports for instances of primary coolant activity exceeding limits; see Step 6.24.6 **(Surry)**

5.36.31 Preparing, obtaining approvals, and submitting reports to the NRC related to general license ISFSI; see Step 6.14.13.

5.37 Supervisor Maintenance Support

The Supervisor Maintenance Support is responsible for:

5.37.1 Notifying the Supervisor Nuclear Site Safety (Station) of certain potential adverse conditions; see Step 6.28.2.a.

5.37.2 Notifying the Supervisor Nuclear Site Safety (Station) of significant additional information related to certain potential adverse conditions and preparing reports to document such information; see Step 6.28.2.c.

5.37.3 Notifying the Supervisor Nuclear Site Safety (Station) of, and preparing reports to document, certain potential incidents; see Step 6.28.3.a.

5.38 Supervisor Nuclear Site Safety

The Supervisor Nuclear Site Safety is responsible for:

5.38.1 Preparing and ensuring submittal of initial and follow-up notification letters to nuclear insurers for inspection report compliance recommendations; see Step 6.28.1.

5.38.2 Determining reportability of potential adverse conditions and notifying the insurer of adverse conditions; see Step 6.28.2.

5.38.3 Reviewing and submitting additional information reports for adverse conditions; see Step 6.28.2.

5.38.4 Determining reportability of potential incidents and notifying the insurer of incidents; see Step 6.28.3.

5.38.5 Notifying the OSHA of accidental deaths or multiple injuries; see Step 6.3.5.c.3.

5.38.6 Notifying the NEIL of fire system impairments and corrections; see Step 6.28.4.a.

5.38.7 Posting OSHA form No. 300 - Log of Work-Related Injuries and Illnesses; see Step 6.19.1.a.

5.39 Supervisor Station Nuclear Safety

The Supervisor Station Nuclear Safety is responsible for ensuring that appropriate assignments are made through the Corrective Action Program to ensure compliance with 10 CF 21; see Step 6.7.2

5.40 Senior Vice President Nuclear Operations

The Senior Vice President Nuclear Operations is responsible for:

5.40.1 Approving reports of planned special exposures; see Step 6.6.5.c.

5.40.2 Approving reports of changes to the Station physical security plan, training and qualification plan, or safeguards contingency plan; see Step 6.14.12.c.

5.40.3 Approving notices for certain events that involve bodily injury or property damage; see Step 6.17.1.c.

5.40.4 Approving notices to indicate renewal or replacement liability insurance policies; see Step 6.17.3.c.

5.40.5 Approving DOT Forms F 5800; see Step 6.21.1.c.

5.40.6 Approving follow-up reports for license violations; see Step 6.23.6.b.

5.40.7 Approving notifications of certain events related to the VPDES permit; see Step 6.26.1.b. **(North Anna)**

5.40.8 Notifying the State Corporation Commission Staff of unplanned outages; see Step 6.27.1.a.

5.40.9 Reporting outage information to the State Corporation Commission; see Step 6.27.1.b.

5.41 Boric Acid Corrosion Control (BACC) Coordinator

The Boric Acid Corrosion Control (BACC) Coordinator is responsible for preparing results of Reactor Pressure Vessel Head related visual inspections; see Steps 6.23.14 and 6.24.15.

5.42 Supervisor Nuclear Spent Fuel (Nuclear Analysis and Fuel)

The Supervisor Nuclear Spent Fuel is responsible for notifying the Supervisor Licensing (Station) of the date planned for first storage of spent fuel under an ISFSI general license, and after using a cask to store spent fuel; see Step 6.14.13.

6.0 INSTRUCTIONS

6.1 General

This Section presents required notifications and reports on the basis of initiating mechanisms. Non-scheduled initiating mechanisms are those that cannot be, or are not easily, pre-scheduled. Non-scheduled mechanisms are further classified according to event or condition, or according to time limitations for fulfilling the required action, or both. Scheduled reports are those whose completion can be pre-scheduled. Subsections 6.2, Non-Scheduled Notifications and Reports, and 6.4, Scheduled Reports, summarize requirements and implementation processes for both groups. Subsections 6.5 through 6.29 provide the details for each requirement.

NOTE: PI-AA-200, Corrective Action, establishes responsibilities and *processing* requirements for initiating and obtaining determinations of reportability for most non-periodic events. **[Commitment 3.2.2]**

6.1.1 Notifications

a. Voice or fax notifications or confirmations by dialable telephone, to individuals or organizations outside Dominion, shall be to the numbers listed in the:

- Applicable Emergency Plan Implementing Procedure
- Emergency Telephone Directory

Voice notification numbers that may not be included in the above listed documents are:

- NRC Director, Spent Fuel Project Office—(301) 415-8500
- National Response Center (EPA and U.S. Coast Guard)—(800) 424-8802
- U.S. Coast Guard—(804) 441-3314 **(Surry)**
- FERC Regional Engineer—(678) 245-3069
- Department of Transportation (DOT)—(800) 424-8802 or (202) 426-2675
- Office of Pesticides & Toxic Substances—(215) 597-8598

6.1.1 **Notifications (continued)**

- State Department of Environmental Quality—
  Air/Water/Waste Regional Office—
  (703) 583-3800 or (after hours) DEM (800) 468-8892 **(North Anna)**
  Air/Water/Waste Regional Office—
  (804) 527-5020 or (after hours) DEM (800) 468-8892 **(Surry)**
  Groundwater Notification—
  (703) 583-3810 or (703) 583-3813 **(North Anna)**
  Groundwater Notification—
  (804) 698-5053 or (804) 527-5038 **(Surry)**
- State Corporation Commission—(804) 371-9611
- Area Director of Occupational Safety and Health Administration (OSHA)—(804) 371-2327
- Nuclear Electric Insurance Limited (NEIL) — (877) 634-5911
- American Nuclear Insurers (ANI)—(860) 682-1301
- Local County Administrator—
  - Louisa County—(540) 967-0401
  - Surry County—(757) 294-5271
- State Department of Emergency Management—(804) 674-2400, ask for EOC Duty Officer

Fax numbers that may not be included in the above listed documents are:

- NRC Operations Center—(301) 816-5151
- NRC Regional Office—(404) 997-4900
- State Department of Environmental Quality —
  - Air/Water/Waste/Pollution Response Regional Office-
    (804) 527-5106 **(Surry)**
  - Air/Water/Waste/Pollution Response Regional Office-
    (703) 583-3871 **(North Anna)**
- Nuclear Electric Insurance Limited—
  (302) 888-3008
- American Nuclear Insurers—(860) 659-0002

b. Notifications to other departments inside Dominion for consideration of additional action(s) to be taken include Electric Environmental Services.

6.1.2 **Reports**

a. Individuals or organizations responsible for preparing a report shall collect, interpret, and ensure the accuracy and validity of information required for a report in accordance with this procedure and with applicable implementing procedures.

b. Individuals or organizations responsible for reviewing a report shall conduct a technical, administrative, and regulatory review, as appropriate.

c. Documents to be submitted to NRC shall be sent to:

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

d. Documents to be submitted to the NRC Regional Office shall be sent to:

USNRC
Region II
Marquis One Tower
245 Peachtree Center Avenue, NE, Suite 1200
Atlanta, GA 30303-1257

e. Documents to be submitted to the REIRS Project Manager shall be sent to:

REIRS Project Manager
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

f. Documents to be submitted to the Office of Nuclear Material Safety and Safeguards shall be sent to:

Director, Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

g. Documents to be submitted to the Division of Low-Level Waste Management and Decommissioning shall be sent to:

Director, Division of Low-Level Waste Management and Decommissioning
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

h. Documents to be submitted to the FERC Regional Office shall be sent to:

Federal Energy Regulatory Commission
Atlanta Regional Office
3700 Crestwood Parkway, NW, Suite 950
Duluth, GA 30096

i. Documents to be submitted to the Virginia Department of Emergency Management shall be sent to:

Virginia Department of Emergency Management
10501 Trade Court
Richmond, VA 23236-3713

j. Documents to be submitted to the State Department of Environmental Quality shall be sent to:

Air
Gregory L. Clayton, Director
State Department of Environmental Quality (Air)
300 Central Road, Suite B
Fredericksburg, VA 22401 **(North Anna)**

Robert L. Beasley, Director
State Department of Environmental Quality (Air)
Arboretum 5, Suite 250
9210 Aboretum Parkway
Richmond, VA 23236 **(Surry)**

Water
Northern Virginia Regional Office (NVRO)
13901 Crown Court
Woodbridge, VA 22193 **(North Anna)**

Water Regional Office
P.O. Box 11143
Richmond, VA 23230-1143 **(Surry)**

k. Documents to be submitted to the Virginia Department of Health shall be sent to:

Office of Water Programs
Environmental Engineering Field Office
131 Walker Street
Lexington, VA 24450-2431 **(North Anna)**

Virginia Department of Health
Southeast Virginia Regional Office
5700 Thurston, Suite 203
Virginia Beach, VA 23455 **(Surry)**

l. Documents to be submitted to American Nuclear Insurers shall be sent to:

American Nuclear Insurers
95 Glastonbury Boulevard, Suite 300
Glastonbury, CT 06033

m. Documents to be submitted to Nuclear Electric Insurance Limited shall be sent to:

David Scott or Greg Wilks
Nuclear Electric Insurance Limited
Manufacturers Hanover Plaza
1201 Market Street, Suite 1200
Wilmington, DE 19801

n. Documents to be submitted to the South Carolina Department of Health and Environmental Control shall be sent to:

South Carolina Department of Health and Environmental Control
2600 Bull Street
Columbia, SC 29209

o. Documents submitted to the local County Administrator shall be sent to:

Louisa County Administrator
P.O. Box 160
Louisa, VA 23093

Surry County Administrator
P. O. Box 65
Surry, VA 23883

6.2 Non-Scheduled Notifications and Reports

NOTE: Reportability determinations for items included in Step 6.2.1 are initiated and processed in accordance with PI-AA-200, Corrective Action.

6.2.1 Critical, Significant, and Potentially Significant Events or Conditions

NOTE: Notifications required by activation of the Emergency Action Plan for Lake Anna Dam are established and controlled by the Plan. However, see Step 6.3.4.a.4.

a. **Emergency Plan Activation**—See Steps **6.3.2**, **6.3.5,** and **6.3.7**.

NOTE: Operability/functionality (availability) is established by the controlling procedure (e.g., Technical Specifications, Station Administrative Procedure). Requirements in this procedure to report inoperable/nonfunctional systems or equipment generally rely on other procedures to establish the basis for determining operability/functionality.

b. **Systems and Components**

- Reactor trip—See Steps **6.3.3**, 6.3.4.a., 6.10.11, 6.27.1.a., and 6.27.2
- Inoperable/nonfunctional (including unavailable or out of service) systems or components—See Steps **6.3.3**, 6.7.2, 6.10.2, 6.10.11, 6.24.13.b., 6.24.17, 6.28.2, 6.28.3, 6.29.1, 6.29.2, 6.29.3, 6.29.4 and 6.29.6
- Fire detection, suppression, or barrier inoperability/nonfunctionality—See Steps **6.3.5.d.**, **6.3.6.a.**, 6.25.1 and 6.28.4
- Defective systems or components—See Steps **6.3.4.1.**, 6.7.2, 6.10.2, 6.10.11 and 6.27.3.e.
- Unacceptable containment leak rate test results—See Step 6.10.17
- Significant changes in projected values of RT_{PTS}—See Step 6.10.7
- Reactor Vessel Overpressure Mitigating System is used to mitigate an RCS transient—See Step 6.24.13.a. **(Surry)**
- Unscheduled outages—See Steps 6.27.1.a. and 6.27.2
- Dissolved gases in transformers exceed limits—See Step 6.28.2
- Conditions affecting the safety of Lake Anna Dam or its works—See Steps **6.3.2.i.** and 6.18.1
- Planned removal from service, and restoration to service, of Lake Anna Dam safety devices—See Step 6.18.8

c. **Operating Limitations**

- Technical Specification safety limit exceeded—See Steps **6.3.2.a.5.**, 6.10.2, 6.23.3 (**North Anna**), and 6.24.3 (**Surry**).
- Limiting Condition for Operation not met—See Steps **6.3.4.1.**, 6.10.2, and 6.10.11.
- Departure from license conditions or Technical Specifications permitted by 10 CFR 50.54(x)—See Steps **6.3.3.a.** and 6.10.11.
- Excess oxygen in waste gas holdup system—See Step 6.24.13.c. **(Surry)**
- Excessive quadrant to average power tilt—See Step 6.24.13.d. **(Surry)**

d. **Radiation or Exposure Events**

- Accidental criticality—See Steps **6.3.3.c.**, 6.17.1 and 6.27.2
- Personnel contamination—See Steps **6.3.2.a.4.**, 6.6.4, 6.17.1 and 6.27.2
- Radiation overexposures—See Steps **6.3.2.a.4.**, 6.6.4, 6.17.1 and 6.27.2
- Planned special exposures—See Step 6.6.5
- At receipt, contaminated or excessively radioactive packages—See Step **6.3.2.b.**
- Radioactive effluent releases—See Steps **6.3.2.a.4.**, **6.3.6.c.**, 6.6.4, 6.10.11, 6.10.16, 6.17.1, 6.26.2, 6.27.2, and 6.28.3
- Radioactive materials transport incident—See Steps **6.3.2.g.** and 6.28.3
- Twenty Four Hour Notification—See Step 6.3.6.a.1.
- Groundwater contamination—See Step **6.3.4** and Subsection 6.31.

e. **Security or Safeguards Events**

- Attempted or actual unauthorized entry—See Steps **6.3.3.f.**, 6.15.3 and 6.27.2
- Acts, attempts, or threats to interrupt normal operation—See Steps **6.3.3.e.**, 6.15.3 and 6.27.2
- Loss, theft, or attempted theft of special nuclear material—See Steps **6.3.2.a.3.**, **6.3.3.c., 6.3.3.e.**, 6.6.2.b., 6.15.3, 6.16.1 and 6.27.2
- Involving byproduct, source, or special nuclear material—See Steps **6.3.3.e.**, 6.6.2, and 6.15.3
- Attempted or actual introduction of contraband—See Steps **6.3.3.h.**, **6.3.6.b.**, 6.8.1
- Loss of shipment of special nuclear material or spent fuel—See Step **6.3.3.d.**
- Violations of requirements of NRC-approved physical security, guard training and qualification, and safeguard contingency plans

f. **Fitness for Duty Events**

- Significant Fitness for Duty events—See Steps **6.3.6.b.**, 6.8.1 and 6.27.2
- NRC employee suspected to be unfit for duty—See Step **6.3.2.c.**
- Drug and Alcohol testing Errors—See Step 6.8.3.

g. **Environmental Events**

- Toxic gas releases—See Steps **6.3.6.c.**, 6.26.2.b. (**North Anna**) and 6.27.2.a.
- Oil or hazardous material spills or releases—See Steps **6.3.2.d.**, **6.3.2.e.**, **6.3.6.c.**, 6.20.4, 6.26.2.b., 6.27.2.a., 6.27.3.l., and 6.27.3.n. **(North Anna)**
- Smoke releases from Station—See Step **6.3.4.b.**
- Significant increase in nuisance organisms or conditions **(North Anna)**—See Steps **6.3.6.c.** and 6.26.2
- Failure to comply with VPDES permit requirements—See Steps **6.3.2.f.**, **6.3.6.f.**, **6.3.6.e.**, 6.26.1.a. and 6.27.3.n.
- Unplanned bypass of waste treatment facilities—See Steps **6.3.6.f.** and 6.27.3.n.
- Unpermitted, unusual, or extraordinary discharge—See Steps **6.3.6.e.** and 6.27.3.n.
- Unanticipated or emergency discharge of waste water or chemical substances—See Steps **6.3.6.c.** (**North Anna**), **6.3.6.e.**, 6.26.2.b. (**North Anna**), 6.27.2 and 6.27.3.n.
- Bird of prey death or injury by electrocution—See Step 6.22.4
- Disturbance of an osprey nest—See Step 6.22.4
- Excessive bird impactions (**North Anna**)—See Step 6.26.2
- Fish kills—See Steps **6.3.6.c.**, 6.26.2.b. and 6.27.2 **(North Anna)**
- On-site plant or animal disease outbreaks—See Steps **6.3.6.c.**, 6.26.2.b. and 6.27.2 **(North Anna)**
- Mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973—See Steps **6.3.6.c.**, 6.26.2 and 6.27.2.a. **(North Anna)**

h. **ISFSI-Unique Events**

- A defect in any spent fuel storage cask structure, system, or component important to safety—See Step **6.3.5.a.7.**
- A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask—See Step **6.3.5.a.7.**

i. **Miscellaneous Events or Conditions**

- Special circumstances that may be considered media significant—See Steps **6.3.4.a.4.**, 6.11.3, and 6.27.2.a.
- Unusual or unplanned occurrences that may be of concern to nearby residents—See Step 6.27.2.a.
- Station fires—See Steps 6.17.1, 6.27.2.a., 6.28.2 and 6.28.3
- Demonstrations, picketing, civil disturbances, strikes, work stoppages—See Steps **6.3.3.e.**, **6.3.4.a.4.**, and 6.27.2.a.
- Earthquakes, storms, floods, forest or brush fires—See Steps **6.3.2.i.**, 6.10.11, 6.18.1 **(North Anna)**, 6.27.2.a. and 6.28.3
- Injuries or accidental deaths—See Steps **6.3.2.g., 6.3.5.c.**, 6.17.1 and 6.27.2.a.
- Transportation of contaminated injured person—See Step **6.3.5.a.5.**
- Deaths or serious injuries at, or alleged to be related to, Lake Anna Dam—See Steps **6.3.2.h., 6.3.5.c.**, 6.18.2 and 6.27.2.a. **(North Anna)**
- Transport incidents involving radioactive or hazardous materials—See Steps **6.3.2.g.**, 6.17.1, 6.21.2, and 6.28.3
- Unanalyzed condition that significantly compromises Station safety—See Step **6.3.5.2.**
- Failure to notify NRC of planned removal or significant changes to equipment that controls amount of radioactivity in effluents—See Step **6.3.6.d.** **(North Anna)**
- Ambulance transport of personnel to an off-site medical facility—See Step 6.27.2
- Mishaps involving low-level waste forms—See Step 6.29.5
- A failure to comply, potentially associated with a significant safety hazard—See Step 6.7.2
- Nonreceipt of hazardous waste shipment manifest from receiver—See Steps 6.20.7.b. and 6.27.3.b.
- Planned or emergency removal of asbestos or asbestos containing material—See Step 6.27.3.b.
- Actual or expected unavailability of licensed waste treatment operator—See Step 6.27.3.m.
- Pump and haul of bulk-storage-tank bottom waters—See Step 6.27.3.p. **(Surry)**
- Operation of auxiliary boiler—See Step 6.27.3.f. **(Surry)**
- Licensed material package effectiveness reduction or with safety-significant defects—See Step 6.13.3

6.2.2 **Special Commitments; Administrative Matters**

a. **Outages and Refueling**

- Outages—See Steps 6.27.1.a. and 6.27.1.b.
- Refueling—See Step 6.23.8
- Removal of Reactor Vessel Material Surveillance Program coupons—See Step 6.10.15
- Restart after refueling, fuel movement, license modification authorizing a power level increase, or Station modifications—See Step 6.24.4 **(Surry)**
- Inservice inspections—See Steps 6.23.4, 6.23.7.a. **(North Anna)**, and 6.24.5 **(Surry)**

b. **Legal & Commercial**

1. **Program & Procedure Changes**
   - Changes to the security plans without prior NRC approval—See Step 6.10.5.b.
   - Revisions to the Emergency Plan or implementing procedures without prior NRC approval—See Step 6.10.5.c.
   - Changes to Chemical Test Program procedures.
   - Significant changes in the operation of equipment that controls the amount of radioactivity in effluents **(North Anna)**—See Step 6.23.5.
   - Changes in discharge or management of pollutants—See Step 6.27.3.j.
   - Significant changes from upstream or downstream conditions addressed in the North Anna Hydroelectric Project Emergency Action Plan **(North Anna)**—See Step 6.18.4
   - Decreased availability of private personnel or equipment to prevent or mitigate a worst-case oil release—See Step 6.27.3.l.

2. **Station Changes**
   - Major changes to radioactive liquid, gaseous, or solid waste treatment systems—See Step 6.10.3
   - Introduction of an extremely hazardous substance in an amount greater than its threshold planning quantity—See Step 6.20.9
   - A change in type of product stored or handled at the Station for which an Material Safety Data Sheet (MSDS) has not been submitted—See Step 6.27.3.l.
   - A substantial increase in the maximum oil storage capacity at the Station—See Step 6.27.3.l.

3. **Movement of Radioactive Materials**
   - Shipment or receipt of SNM—See Steps 6.15.1, 6.15.2, 6.15.3, 6.15.4, and 6.16.3
   - First use of radioactive material packaging—See Step 6.13.2

4. **NRC Licences, Orders, & Inspections**
   - Change in operator or senior operator status—See Step 6.10.12
   - Receipt of NRC notices of violation that involve radiological working conditions, proposed impositions of civil penalty, orders for imposing requirements, orders modifying, suspending, revoking a license, orders imposing a civil penalty, and responses thereto. See Steps 6.5.1.e. and 6.5.1.f.
   - Issuance of an NRC shutdown order—See Step 6.28.6
   - Issuance of Dominion Annual Report—See Steps 6.10.8 and 6.14.11
   - Five years before expiration of reactor operating license—See Step 6.10.5.f.
   - Three years before the predicted date that fracture toughness levels will no longer satisfy 10 CFR 50, App. G, Section IV.A—See Step 6.10.14.
   - Suspension or revocation of an NRC operating license—See Step 6.28.6
   - A change of licensee for the Station—See Step 6.27.3.l.

5. **Permits, Orders, & Evaluations**
   - Proposed changes to the VPDES permit—See Step 6.26.1.b. **(North Anna)** or Step 6.27.3.o. **(Surry)**
   - Changes or additions to the VPDES permit or State certification—See Step 6.26.1.b.**(North Anna)** or Step 6.27.3.o. **(Surry)**
   - Stay of a VPDES permit or State certification appeal—See Step 6.26.1.b. **(North Anna)**
   - Modifications to Lake Anna Dam or its works—See Step 6.18.3
   - Suspension from INPO—See Step 6.28.5
   - Classification as INPO Category 5—See Step 6.28.5

6. **Insurance & Financial**
   - Material change in proof of financial protection or financial information previously filed—See Step 6.17.2
   - Expiration, renewal, or replacement of 10 CFR 140 financial protection—See Step 6.17.3
   - Filing of Chapter 11 petition by or against any component of Dominion Resources—See Step 6.10.5.g.

c. **Individual Requests or Directives**
   - Worker and former worker radiation exposure data—See Steps 6.5.2 and 6.5.3.
   - Radiation overexposures—See Step 6.5.4.
   - Terminating employees & workers—See Step 6.5.5.

6.3 Immediate to 72-Hour Notifications

This subsection consolidates requirements for situations or events addressed by Subsections 6.5 through 6.29, for which notifications or reports are required within 72 hours.

6.3.1 General Requirements

a. When this subsection (Subsection 6.3) designates someone other than the Shift Manager or a member of Station management to notify a government agency, that person shall ensure the Shift Manager or a member of Station management is advised before making the notification. See also Step 6.3.4.a.4.

NOTE: Notifications for events that exceed an Emergency Action Level, as specified in EPIP-1.01, Emergency Manager Controlling Procedure, are controlled by EPIP-2.01, Notification of State and Local Governments and EPIP-2.02, Notification of NRC. See also Steps 6.3.5 and 6.3.7. **[10 CFR 50.72(a)(3), 10 CFR 50.72(c)(1), 10 CFR 50.72(c)(2)]**

NOTE: When it is discovered that an event or condition had existed, but the basis for the emergency class no longer exists at the time of this discovery and no other reasons exist for an emergency declaration, then declaration of an emergency class is not required. See Step 6.3.3.i. for notification requirements.

b. For events reportable to the NRC Operations Center, the Shift Manager shall:

1. Complete NRC Form 361, Event Notification Worksheet.
2. Fax the Event Notification Worksheet to the NRC Operations Center. See Step 6.1.1.
3. Using the Emergency Notification System (ENS), verify that NRC received the fax.
4. Be prepared to read the entire contents of the Event Notification Worksheet to the NRC Operations Center officer.
5. Ensure the NRC Operations Center officer has a clear understanding of the issues, and that all questions regarding the notification have been answered.
6. If the ENS is nonfunctional, use commercial telephone service, other dedicated telephone service, or any other method that ensures the NRC Operations Center is notified as soon as practical. See Step 6.1.1. **[10 CFR 50.72(a)(2)]**

7. Maintain an open, continuous communications channel with the NRC Operations Center, when requested by NRC. **[10 CFR 50.72(c)(3) & 10 CFR 73.71(a)(3)]**

c. For events that are reportable in accordance with 10 CFR 50.72 and 10 CFR 72.75:

- Immediately, the Shift Manager shall notify the Manager Nuclear Operations or the Operations Manager On Call, and the STA
- Within one hour, the Manager Nuclear Operations or Operations Manager On Call shall notify the Site Vice President and the Plant Manager (Nuclear)
- Within one hour, the STA shall notify the Director Nuclear Station Safety and Licensing
- Within one hour, the Director Nuclear Station Safety and Licensing (if absent, the Plant Manager (Nuclear)) shall notify the Manager Nuclear Oversight of reactor trips; for other events that are reportable in accordance with 10 CFR 50.72 and 10 CFR 72.75, this notification shall be made within 24 hours
- Within 24 hours, the Director Nuclear Station Safety and Licensing (if absent, the Plant Manager (Nuclear)) shall notify the NRC Resident Inspector.
- Within 24 hours, the Director Nuclear Station Safety and Licensing (if absent, the Plant Manager (Nuclear)) shall notify the Director NL&OS
- Within 24 hours, the Site Vice President, a Director, Manager Nuclear Operations, or Shift Manager shall notify the Senior Vice President Nuclear Operations
- When notified, the Director NL&OS shall promptly notify appropriate corporate organizations, including Public Relations, Medical, Corporate Risk Management, and Power Supply, as applicable

6.3.2 **Immediate Notifications**

NOTE: Some conditions, indicated by “See EPIP-1.01,” may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, Emergency Plan Implementing Procedures (EPIPs) control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

NOTE: Upon NRC request, the designated responsible person must maintain an open, continuous communications channel with the NRC Operations Center. **[10 CFR 50.72(c)(3)]**

a. The Shift Manager shall notify the NRC Operations Center via the ENS of:

1. Any further degradation in the level of safety of the plant or other worsening plant conditions, after telephone notifications to NRC as specified in Step 6.3.2 or Step 6.3.3. See EPIP-1.01. **[10 CFR 50.72(c)(1)]**
2. The results of ensuing evaluations or assessments of plant conditions, the effectiveness of response or protective measures taken, and information related to plant behavior that is not understood, after telephone notifications to NRC as specified in Step 6.3.2 or Step 6.3.3. **[10 CFR 50.72(c)(2)]**
3. Lost, stolen or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10 CFR 20, Appendix C, under circumstances in which it appears persons in unrestricted areas could be exposed. See also Steps 6.6.2.b. and 6.6.2.c. **[10 CFR 20.2201(a)(i)]**

NOTE: The requirements of Step 6.3.2.a.4. do not apply to doses that result from planned special exposures, that are within the limits for planned special exposures, and that are reported in accordance with Step 6.6.5. **[10 CFR 20.2202(e)]**

4. Events that involve by-product, source, or special nuclear material possessed by Dominion that may have caused or threatens to cause: **[10 CFR 20.2202(a)]**
   - An individual to receive:
     - •• A total effective dose equivalent of ≥ 25 rems
     - •• An eye dose equivalent of ≥ 75 rems
     - •• A shallow-dose equivalent to the skin or extremities of ≥ 250 rads
   - Release of radioactive material inside or outside a restricted area, so that, if an individual had been present for 24 hours, they could have received an intake five times the occupational annual limit on intake

   If the event involves radiological overexposure, the DEM shall be notified as specified in Step 6.27.2. See also Step 6.6.3.c.

5. A Technical Specifications safety limit violation. See also Steps 6.23.3 **(North Anna)** and 6.24.3 **(Surry)**. **[10 CFR 50.36(d)(1)(i)(A)]**

6. Upon declaration of an emergency as specified in the approved emergency plan regarding ISFSI events. **[10 CFR 72.75(a)]**

b. If:

- Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87(i) **[10 CFR 20.1906(d)(1)]**

  or

- External radiation levels exceed the limits of 10 CFR 71.47 **[10 CFR 20.1906(d)(2)]**

1. Radiological Protection shall notify Supervisor Licensing (Station) and the Shift Manager.

2. Radiological Protection or Supervisor Licensing (Station) shall notify (see Step 6.3.1.a.) the final delivering carrier and, by telephone and telegram, mailgram, or facsimile, the NRC Operations Center. See Step 6.1.1.

3. The notifier in Step 6.3.2.b.2. shall initiate a Condition Report as specified in PI-AA-200, including documentation of its notifications on the Condition Report.

c. If an NRC employee or NRC Contractor is believed to be under the influence of any substance or otherwise unfit for duty, the Fitness for Duty Administrator (Station) or a Station Management staff member shall notify (see Step 6.3.1.a.) the NRC Regional Administrator by telephone followed by written notification (e.g., e-mail or fax). If the Regional Administrator cannot be reach, notification shall be made to the NRC Operations Center. **[10 CFR 26.77(c)]**

NOTE: Use Table 1, Summary of Reporting Requirements for Non-Radiological Releases To the Environment, to supplement Step 6.3.2.d. for reporting requirements. The Environmental Compliance Coordinator or Electric Environmental Services should be consulted when assessing oil release reportability.

d. If oil may have been released from the Station into state waters that:
- Violates applicable water quality standards (i.e., **any** oil in water) **[40 CFR 110.3]**
- Causes a film or sheen upon or discoloration of the surface of the water or adjoining shorelines **[40 CFR 110.3]**
- Causes a sludge or emulsion to be deposited beneath the surface of the water or upon adjoining shorelines **[40 CFR 110.3]**

or

If oil can reasonably be expected to enter, or there is a substantial threat that oil will enter, state waters or storm drains **(Reference 3.1.8)**

or

If more than 25 gallons[1] of oil has been or can reasonably be expected to be released to soil, including a spill within containment facilities[2] **(Reference 3.1.8)**:

or

1. Notice is considered to have been given to the State Water Control Board for oil releases to the ground up to 25 gallons if and only if the Environmental Compliance Coordinator prepares and maintains a record of such oil releases for five years, and the oil is cleaned up.
2. Oil tank dikes and transformer vaults are typical containment facilities.

Table 1
Summary of Reporting Requirements for Non-Radiological Releases To the Environment[a]

Material	Released To		Amount[b]	Report To[c]	See
Oil	Soil	Outside containment facilities[d]	> 25 gallons	DEQ & LEPC	6.3.2.d.
	Solid surface	Potential to reach soil or water	Case basis		
	State Waters[e]		Any discernible amount[k]	NaRC, DEQ & LEPC	
Hazardous Substance[f, g]	Land	Off-site	≥ RQ	NaRC, DEQ & LEPC	6.3.2.e. 6.3.2.g.[h]
	Water	On-site[i]	< RQ	DEQ	
			≥ RQ	NaRC, DEQ, & LEPC	
		Off-site	< RQ	DEQ	
			≥ RQ	NaRC, DEQ,	
Hazardous Waste[j]	Land	On-site[i]	≥ RQ	NaRC & DEQ	
		Off-site	Any amount	NaRC, DEQ & LEPC	
	Water	On-site[i]	< RQ	DEQ	
			≥ RQ	NaRC, DEQ, & LEPC	
		Off-site	Any amount	NaRC, DEQ, & LEPC	

a. Step 6.3.2.e. explains "releases to the environment" when hazardous substances or hazardous wastes are involved.
For oil, "releases" includes spilling, leaking, pumping, pouring, emitting, emptying, discharging, injecting, escaping, leaching, or disposing. Oil releases solely within a workplace (an enclosed building with a concrete floor) are not subject to the RQ unless oil reaches a floor drain connected to a pathway to the environment. All outdoor releases are subject to the RQ.
b. RQ = reportable quantity as specified in VPAP-2202, Control of Chemicals and Hazardous Substances.
c. DEQ = State Department of Environmental Quality; NaRC = National Response Center*; LEPC = Local Emergency Planning Coordinator. *At Surry, if the NaRC is notified, the U.S. Coast Guard must also be notified. DEM (Department of Emergency Management (Emergency Operations Center)) is notified (instead of DEQ) on nights, weekends, and after hours. Environmental "immediate notification" is defined as "as soon as possible, but not to exceed 24 hours." Phone numbers for the agencies are found in 6.1.1. Attachment 1, Oil or Hazardous Substance Release Report, should be used for spill information requested by the agencies. NRC notification is required within 4 hours of notifying any of these agencies.
d. Oil tank dikes and transformer vaults are typical containment facilities.
e. Includes releases to storm drains or comparable conduits to state waters. See also 4.36, State Waters.
f. Hazardous substances of concern to the Station are identified in VPAP-2202, Control of Chemicals and Hazardous Substances.
g. See Footnote 1. on page 73 if there is an on-site RQ release of a volatile substance.
h. 6.3.2.g. is applicable for transportation-related events.
i. An on-site hazardous material spill that, due to location, size, or substance properties, poses imminent or likely danger of an RQ release to the environment, must be reported to the same entities as offsite spills.
j. Hazardous waste is defined in the Environmental Protection Plan.
k **If the oil spill that reaches navigable waters is: a) greater than 1000 gallons or b) the second spill that is greater than 42 gallons in a 12 month period, then contact Electric Environmental Services (EES). Additional reporting by EES to the EPA will be required. [Commitment 3.2.26]**

If any spill reaches a solid surface, including surfaces inside secondary containment systems and inside buildings, and (1) if there is the potential for oil to reach surface water, and/or (2) if there is the potential for greater than 25 gallons of oil to reach soil

1. The individual who observes or suspects such an event or condition shall notify the Shift Manager.
2. The Shift Manager shall notify the Manager Nuclear Operations, the Environmental Compliance Coordinator, or Environmental Policy & Compliance, as available.
3. If the discharge is to storm drains or state waters, the Environmental Compliance Coordinator or Electric Environmental Services (see Step 6.3.1.a.) **(North Anna)** the Shift Manager **(Surry)** shall notify the National Response Center, the State Department of Environmental Quality (Water) (DEQ), the LEPC, and **(Surry)** the U.S. Coast Guard. If the discharge is to land, DEQ and the LEPC shall be notified. Notifications shall be documented on Attachment 1, Oil or Hazardous Substance Release Report. See 6.1.1.a. See also Steps 6.3.4.a.4., 6.20.4, and 6.27.3.n.[3]

NOTE: The Environmental Compliance Coordinator or Electric Environmental Services should be consulted when assessing hazardous material release reportability.

e. If a regulated, hazardous material release to the environment[1] exceeds a reporting threshold as specified in Table 1[2]:

   1. The individual who becomes aware of the release or potential release shall notify the Shift Manager. See EPIP-1.01.

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1. Reportable Quantity (RQ) is the amount of a regulated, hazardous material *released to the environment* during a 24-hour period that must be reported in accordance with federal agency requirements. RQ only applies to a release to the environment, so will not apply for every release of a regulated, hazardous material. For example, a hazardous substance spill that is contained *entirely* on-site, even if more than the RQ, is **not** reportable because it is not a release to the environment. However, if an RQ amount evaporates or is absorbed in soil, the spill has not been contained entirely on-site, and thereby becomes a reportable release to the environment.
   If the VPDES or other permit authorizes discharge of a hazardous material, a discharge is **not** reportable as a release to the environment unless a discharge amount or concentration exceeds the permit-authorized limit or the discharge is via a pathway not specified during the permit application and approval process. Permit-authorized discharges are reportable only as required by the applicable permit (e.g., the monthly Discharge Monitoring Report, per Step 6.27.3.i., required by the VPDES permit).
   If an amount or concentration does exceed a permit-authorized limit or is discharged via a pathway other than specified during the permit application and approval process, the RQ and associated reporting requirements will apply.

2. The Shift Manager shall notify the Manager Nuclear Operations, the Environmental Compliance Coordinator, or Electric Environmental Services, as available.

3. The Environmental Compliance Coordinator or Electric Environmental Services (see Step 6.3.1.a.) **(North Anna)** Shift Manager **(Surry)** shall notify the agencies listed in the "Report To" column of Table 1. If a reportable release involves off-site transportation (including storage incident to such transportation), the Shift Manager shall also notify the 911 operator, local and state police, and the National Response Center. Notifications shall be documented on Attachment 1, Oil or Hazardous Substance Release Report. See Step 6.1.1.a. See also Steps 6.3.2.g., 6.3.4.a.4., 6.22.3.b. and 6.27.3.n. **[CERCLA Sec. 304(b)(1); 40 CFR 302]**

NOTE: Items marked with an asterisk(*) on Attachment 1 are required to be reported to the response agencies listed in block 10 of the attachment. **(Reference 3.1.106)**

4. Notifications shall include (to the extent known) **[CERCLA Sec. 304(b)(2)]**:
   - The chemical name or identity of the substance involved in the release
   - Whether the substance is on the list referred to in section 302(a) of CERCLA, 40 CFR 302.
   - An estimate of the quantity of substance released to the environment
   - The time and duration of the release
   - The medium or media into which the release occurred
   - Any known or anticipated acute or chronic health risks associated with the emergency and, where appropriate, advice regarding medical attention necessary for exposed individuals
   - Proper precautions to take as a result of the release, including evacuation
   - The name and telephone number of the Dominion contact

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2. Table 1 does not mention PCBs because no PCBs are in use at the Station. The Environmental Compliance Coordinator or Electric Environmental Services should be contacted for further instructions if any question arises concerning PCBs being introduced on-site and any consequent reporting.

3. If the discharge occurs in the Main Switchyard the Dominion System Operator Transmission shall be notified. If the discharge is from the transformer belonging to Rappahannock Electric Cooperative at the Dam then that company shall be notified **(North Anna)**

f. If the Station does not comply with one or more limitations, standards, monitoring, or management requirements specified in the VPDES permit (if oil is involved, go to Step 6.3.2.d.; if hazardous materials are involved, go to Step 6.3.2.e.) and such noncompliance:

- May adversely affect State waters

  or

- May endanger public health[1]

As soon as possible, the Environmental Compliance Coordinator or Electric Environmental Services shall notify (see Step 6.3.1.a.) the State Department of Environmental Quality (Water) by telephone with the following information **[VPDES Permit]**:

- A description and cause of noncompliance
- The period of noncompliance, including exact dates and times or anticipated time when the noncompliance will cease
- Actions taken or planned to reduce, eliminate, and prevent recurrence

See also Steps 6.3.4.a.4., 6.27.2.a.1., and 6.27.3.n.

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1. Applicable regulations use, but do not define, the terms “adversely affect” and “endanger public health.” These terms must be interpreted on a case-by-case basis by individuals with aquatic ecology expertise and thorough familiarity with current regulatory agency reporting and enforcement policy. Such individuals will also determine how soon a specific event must be reported to avoid enforcement (i.e., within minutes of an event, or some longer time within the not-to-exceed 24-hour limit established by the VPDES Permit).

g. If an incident occurs during transport (including loading, unloading, and temporary storage) of:

- Radioactive materials in which fire, breakage, spillage, or suspected radioactive contamination occurs (see also Step 6.28.3) **[49 CFR 171.15(b)(2)]**
- Hazardous materials in which any of the following is a direct result of the hazardous materials: **[49 CFR 171.15(b)(1)]**
  - •• A person is killed
  - •• A person requires hospitalization because of injuries
  - •• An evacuation of the general public occurs lasting one or more hours
  - •• One or more major transportation arteries or facilities are closed or shut down for one hour or more
  - •• The operational flight pattern or routine of an aircraft is altered
- A situation exists (e.g., a continuing danger to life exists at the scene of the incident) that, in the judgment of the carrier or Dominion, should be reported even though it does not meet one of the previous criteria **[49 CFR 171.15(b)(5)]**

Supervisor Licensing (Station) shall notify (see Step 6.3.1.a.) DOT by telephone, or confirm carrier notification of DOT by telephone. See also Steps 6.3.2.e. and 6.21.2. The notification shall include the **[49 CFR 171.15(a)]**:

- Notifier's name
- Name and address of carrier represented by the notifier
- Phone number where the notifier can be contacted
- Date, time, and location of incident
- The extent of injuries, if any
- Classification, name, and quantity of radioactive or hazardous materials involved, if available
- Type of incident and nature of radioactive or hazardous material involvement and whether a continuing danger to life exists at the scene

h. If a serious accident or a death occurs at or immediately above or below Lake Anna Dam[1] or is alleged to be related to the existence or operation of the dam:

   1. The Lake Anna Dam Operator shall notify the Shift Manager and provide information necessary to prepare Attachment 2, FERC Public Safety Database Report.

   2. The Shift Manager shall initiate a Condition Report in accordance with PI-AA-200.

   3. The Shift Manager should notify the FERC Regional Engineer of the condition by telephone. See Step 6.1.1.a. See also Steps 6.3.4.a.4., 6.3.5.c., and 6.18.2.b. **(North Anna)**

i. If a condition is identified that affects the safety of Lake Anna Dam or its associated works (see Subsection 4.8), but does not require entry into the North Anna Hydroelectric Project Emergency Action Plan:

   1. The Lake Anna Dam Operator shall notify the Shift Manager and provide relevant supporting information.

   2. The Shift Manager shall notify, by telephone, the FERC Regional Engineer of the condition and initiate a Condition Report in accordance with PI-AA-200. See Step 6.1.1.a. See also Steps 6.3.4.a.4. and 6.18.1.b. **(North Anna)** **[18 CFR 12.10(a)]**

1. Incidents which involve other parts of the lake are excluded. **[18 CFR 12.10(b)(4)]**

6.3.3 One-hour Notifications

NOTE: Some conditions, indicated by "See EPIP-1.01," may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

As soon as practical, but within one hour, the Shift Manager, Station Emergency Manager, or Site Vice President shall notify the NRC Operations Center of:

a. Deviation from Technical Specifications (permitted by 10 CFR 50.54(x)) to protect the health and safety of the public, when no action consistent with license conditions and Technical Specifications can provide adequate or equivalent protection. **[10 CFR 50.72(b)(1)]**

b. An automatic safety system that does not function as required during operation. See EPIP-1.01. **[10 CFR 50.36(d)(1)(ii)(A)]**

NOTE: Notifications required by Steps 6.3.3.c., 6.3.3.d., and 6.3.3.e., are exempt from the requirement that Safeguards Information be transmitted only by protected telecommunications circuits approved by NRC.

c. An accidental criticality or loss of SNM. See EPIP-1.01.
[10 CFR 70.52 (a), 10 CFR 72.74(a), 10 CFR 74.11a]

NOTE: Step 6.3.3.d. notifications need not duplicate Step 6.3.3.e. notifications. **[10 CFR 74.11(c), 10 CFR 72.74(c)]**

d. A loss of any **[10 CFR 73.71(a)(1), 10 CFR 73.67(e)(3)(vii), 10 CFR 73.67(g)(3)(iii)]**:

- SNM shipment
- Spent fuel shipment

or

Availability of supplemental information after initial notification. **[10 CFR 73.71(a)(5)]** (See also Step 6.15.3.a.3.)

or

Recovery of or accounting for such lost shipment.
See also Step 6.15.3.a.2. **[10 CFR 73.71(a)(1), 10 CFR 73.67(e)(3)(vii), 10 CFR 73.67(g)(3)(iii)]**

NOTE: Steps 6.3.3.e., 6.3.3.f., 6.3.3.g., 6.3.3.h.notifications need not duplicate Step 6.3.3.d. or 10 CFR 50.72 notifications. **[10 CFR 72.74(c), 10 CFR 73.71(e), 10 CFR 74.11(c)]**

e. A reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause (See also Step 6.15.3.b.2.).
[10 CFR 73.71(b)(1), 10 CFR 73 App. G.I, 10 CFR 70.52 (a), 10 CFR 72.74(a), 10 CFR 74.11(a)]:

- Theft, loss, or unlawful diversion of SNM
- Significant physical damage to the Station, nuclear fuel, or carrier of nuclear fuel
- Interruption of normal operation through unauthorized use of or tampering with its machinery, components, or controls, including the security system

f. Unauthorized entry into a protected area, material access area, controlled access area, vital area, or transport.

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

NOTE: Fitness-for-duty events are reported in accordance with 10 CFR 26 instead of 10 CFR 73.71. See Steps 6.3.6.b. and 6.8.1. **[10 CFR 26.73(c)]**

h. Actual or attempted introduction of contraband into a protected area, material access area, or transport.

i. Discovery that an undeclared or misclassified event or condition met all the following criteria: **[10 CFR 50.72(a)(1)(i)]**

- Exceeded an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure
- The basis for the emergency class no longer exists at the time of discovery
- No other reasons exist for an emergency declaration

In addition, the following shall be notified:

- Department of Emergency Management (at approximately the same time)
- Director Nuclear Protection Services and Emergency Preparedness
- Louisa/Surry County Administrator

6.3.4 **Four-hour Notifications**

NOTE: Some conditions, indicated by “See EPIP-1.01,” may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

a. As soon as practical, but within four hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

NOTE: If a unit enters a limiting condition for operation (LCO) and a unit shutdown is started due to the LCO, the event **is** reportable even if shutdown is not completed. LCOs terminated by a unit shutdown for an unrelated reason are still reportable if the condition would not have been corrected within the LCO time limit for shutdown.

1. Initiation of plant shutdown (reduction of power or temperature) required by Technical Specifications. The initiation of plant shutdown does not include mode changes required by Technical Specifications if initiated after the plant is already in a shutdown condition. See EPIP-1.01. **[10 CFR 50.72(b)(2)(i), 10 CFR 50.36(d)(1)(i)(A), 10 CFR 50.36 (d)(2)(i), NUREG 1022 Item 3.2.1]**
2. Any event that results or should have resulted in ECCS discharge into the RCS as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. **[10 CFR 50.72(b)(2)(iv)(A)]**
3. Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when actuation results from and is part of a pre-planned sequence during testing or reactor operation. **[10 CFR 50.72(b)(2)(iv)(B)]**

NOTE: "Notification to other government agencies has been or will be made" is not necessarily an automatic notification to the NRC. Refer to NUREG – 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, for discussions and examples or contact Station Licensing if clarification is needed. **[NUREG-1022, Section 3.2.12]**

4. 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. **[Commitment 3.2.12]** **[10 CFR 50.72(b)(2)(xi)]**

5. ISFSI Non-emergency Four-Hour Notifications shall include, if available at time of notification: **[10 CFR 72.75(e)(3)]**
   - The caller's name and call back telephone number
   - A description of the event, including time and date
   - The exact location of the event
   - The quantities, and chemical and physical forms of the spent fuel, HLW or reactor related Greater than Class C (GTCC) waste involved
   - Any personnel radiation exposure data
6. An action taken in an emergency that departs from a license condition, technical specification, or certificate of compliance when the action is immediately needed to protect the public health and safety and no licensed action that provides adequate or equivalent protection is immediately apparent—see Step 6.14.7.f. **[10 CFR 72.75(b)(1)]**
7. An event at the ISFSI that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination. **[10 CFR 72.75(c)(3)]**
8. Groundwater Protection Voluntary Communication Notifications to other government agencies may be reportable under 10 CFR 50.72(b)(2)(xi) requirement for a 4-hour notification to the NRC operations center based upon the following guidance:
   - If a licensee is notifying a local, state, or other federal agency in accordance with an existing law, regulation, or ordinance, then the licensee should make its notification to the NRC under the 50.72 notification requirement.
   - If a licensee is informally communicating with a local, state, or other federal agency (i.e., not under a specific law, regulation or ordinance), then the licensee has discretion as to whether to informally communicate with NRC (e.g., through the site resident inspector and/or regional NRC office) or formally through the 50.72 notification process. If due to the site-specific circumstances or heightened sensitivity to the issue at that site, the issue is likely to produce strong media interest, then the licensee should consider notifying NRC under the 50.72 requirement because this is actually the underlying intent of the regulation.

b. Any person at the Station who observes smoke originating from Station equipment being released into the outdoor atmosphere shall notify the Shift Manager as soon as possible.

1. If the smoke is not from a fire and there are no certified visible emissions evaluators available to determine the opacity of the smoke being released to the outdoor atmosphere, the Shift Manager or other Station personnel shall take the appropriate steps to determine the source, cause, and duration of the smoke being released.
    - Once all of the pertinent information regarding the release of smoke has been obtained, the Electric Environmental Services (ESS) must be notified immediately.
    - The ESS will report the release of smoke into the outdoor atmosphere to the appropriate DEQ regional office as soon as practical, but no later than four daytime business hours of the occurrence, with all of the pertinent information. If the DEQ regional office determines that it is necessary to obtain smoke readings after receiving all of the pertinent information, the ESS will dispatch a certified visible emissions evaluator to the Station to determine the opacity of the smoke being released into the outdoor atmosphere.
2. The ESS will prepare and submit any written reports to the DEQ regional office regarding the release of smoke into the outdoor atmosphere.

6.3.5 **Eight-hour Notifications**

a. As soon as practical, but within eight hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

1. Any condition that results in the condition of the Station, including its principal safety barriers, being seriously degraded. **[10 CFR 50.72(b)(3)(ii)(A)]**
2. Any event or condition that results in the Station being in an unanalyzed condition that significantly degrades plant safety. **[10 CFR 50.72(b)(3)(ii)(B)]**

3. Any event or condition that results in valid actuation of any of the following systems, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation: **[10 CFR 50.72(b)(3)(iv)(A)]**
   - Reactor Protection System (RPS) - (RPS actuation with the reactor critical may be reportable within 4 hours under 10 CFR 50.72(b)(2)(iv)(B), see Step 6.3.4.a.3.)
   - General containment isolation signals affecting containment isolation valves in more than one system or multiple Main Steam Isolation Valves (MSIVs)
   - Emergency Core Cooling Systems (ECCS) including HHSI and LHSI (Actual discharges are reportable within 4 hours under 10 CFR 50.72(b)(2)(iv)(A), see Step 6.3.4.a.2.)
   - Auxiliary Feedwater System
   - Containment heat removal and depressurization systems including Containment spray and fan cooler systems
   - Emergency Diesel Generators (EDGs)

4. Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:
   - Shut down the reactor and maintain it in a safe shutdown condition
   - Remove residual heat
   - Control the release of radioactive material; or
   - Mitigate the consequences of an accident. See EPIP-1.01. **[10 CFR 50.72(b)(3)(v)]**

5. Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment. See also Step 6.27.2. **[10 CFR 50.72 (b)(3)(xii)]** Could also be a 4 hour report in accordance with 10 CFR 72.75 (b)(5).

6. An event that results in a major loss of emergency assessment capability[1], off-site response capability, or off-site communications capability, e.g., unavailability of any of the following (see Attachment 3, Emergency Response Unavailability, for unavailability criteria)[2]:
   - Safety Parameter Display System[3] (SPDS)
   - Emergency response facilities[4] (see Subsection 4.15)
   - Emergency communications facilities and equipment[5]
   - Prompt Notification System, including sirens
   - Plant monitors necessary for accident monitoring

   See EPIP-1.01. **[10 CFR 50.72(b)(3)(xiii)]**

7. Any instance of:
   - A defect in any spent fuel storage cask structure, system, or component that is important to safety **[10 CFR 72.75(c)1]**

     or

   - A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask **[10 CFR 72.75(c)2**

   See EPIP-1.01.

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1. A major loss of emergency assessment capability includes events that significantly impair fulfillment of the Emergency Plan, including safety assessment capability (e.g., loss of a significant portion of Control Room indications). Loss of on-site meteorological information does not constitute a major loss of assessment capability and should not be reported under this part.
2. Engineering judgment may be needed to assess the significance of losing certain equipment.
3. The SPDS is one function of the Plant Computer System (PCS). Unavailability of the SPDS can result from malfunctions ranging from those limited to the SPDS only, to those that cause a total loss of the PCS. Unavailability of only the SPDS, whether from a limited malfunction or a total loss of PCS, for less than eight hours is not reportable; but, unavailability of SPDS (up to and including a total loss of the PCS) along with unavailability of other assessment capability at the same time may be reportable. Scheduled PCS outages or operation of the PCS in the Simulator mode are not reportable if the SPDS can be made available in less than one hour (the time frame of one hour is commensurate with the required time for activation of an Emergency Response Facility, when required). [Reference 3.1.104]
4. EOF loss is reportable only if **both** the LEOF **and** the CEOF are unavailable.
5. A momentary loss of off-site response capability or emergency communications (e.g., the backup power supply fails while security computer and emergency communications are temporarily connected to perform a surveillance test) is **not** reportable.

b. If an Alert, Site Area Emergency, or General Emergency is declared:

1. The Station Coordinator Emergency Preparedness shall prepare a Summary Report from information in completed Emergency Plan Implementing Procedures, Control Room logs, and interviews with persons involved with the declaration and response, as appropriate. See Attachment 6, Example DEM Summary Report.

2. The Site Vice President, Director Nuclear Station Safety and Licensing, or Plant Manager (Nuclear) shall approve the report.

3. Within 8 hours after termination of the event, Nuclear Emergency Preparedness shall ensure the report is delivered to the State Coordinator of the Virginia Department of Emergency Management. **[NAEP 4.4; SEP 4.4]**

c. If, on Dominion property or at Lake Anna Dam, there is a Dominion employee or contractor fatality (regardless of the time between the injury and death, or the length of an illness) or an event in which three or more Dominion employees or contractors are hospitalized:

1. The Shift Manager shall notify Supervisor Nuclear Site Safety (Station) with the following information:
   - Number of fatalities
   - The employer of those killed
   - The circumstances of the event
   - The extent of injuries

2. Nuclear Site Safety (Station) shall notify OSHA as specified in Step 6.3.5.c.3. See also Step 6.3.4.a.4.

3. Within eight hours after the occurrence, the Supervisor Nuclear Site Safety (Station) (as specified in Step 6.3.5.b.2.) shall notify See Step 6.3.1.a.) the Area Director of OSHA by telephone or facsimile. See Step 6.1.1.a. See also Step 6.3.4.a.4. **[29 CFR 1904.8]**

d. Whenever fire protection systems, portions of a system, or equipment are impaired or reduced in status for other than scheduled maintenance or scheduled testing activities (meaning an unplanned failure or state of degradation), the Shift Manager shall notify the Supervisor Nuclear Site Safety (Station). **[Commitment 3.2.17] (Surry)**

North Anna notification to the Supervisor Nuclear Site Safety (Station) is within 48 hours per TRM requirements.

6.3.6 **Twenty-four Hour Notifications**

a. As soon as practical, but within 24 hours, the Shift Manager shall notify the NRC Operations Center with the ENS of **[10 CFR 20.2202(b)]**:

NOTE: The requirements of Step 6.3.6.a.1. do not apply to doses that result from planned special exposures, that are within the limits for planned special exposures, and that are reported in accordance with Step 6.10.11.c. **[10 CFR 20.2202(e)]**

1. An event that involves licensed material possessed by Dominion that may have caused or threatens to cause:
   - An individual to receive, in a period of 24 hours:
     - •• A total effective dose equivalent exceeding 5 rems
     - •• An eye dose equivalent exceeding 15 rems
     - •• A shallow-dose equivalent to the skin or extremities exceeding 50 rems
   - Release of radioactive material inside or outside a restricted area, so that, if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

   If an event involves radiological overexposure, DEM must be notified as specified in Step 6.27.2. See also Step 6.6.3.c.

2. ISFSI Twenty-Four Hour Notifications shall include, if available at time of notification: **[10 CFR 72.75(e)(3)]**
   - The caller's name and call back telephone number
   - A description of the event, including time and date
   - The exact location of the event
   - The quantities, and chemical and physical form of the spent fuel or HLW involved
   - Any personnel radiation exposure data

3. An unplanned contamination event that requires access to the contaminated area by workers or the public to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area [10 CFR 72.75(c)(1)]

4. An event in which safety equipment is disabled or fails to function as designed when: **[10 CFR 72.75(d)(1)]**
   - The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposure to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident,

     and

   - No redundant equipment was available and operable to perform the required safety function

5. An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive material that could exceed regulatory limits or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases)—see Step 6.14.7.f. **[10 CFR 72.75(d)(1)(i)]**

b. Within 24 hours after discovery of a significant FFD violation or programmatic failure, or drug and alcohol testing errors, the Shift Manager or Station Management Staff member shall report them by telephone to the NRC Operations Center (See Steps 6.1.1 and 6.8.3). **[10 CFR 26.719]**

   1. The notifier shall document the notification in Section B of Attachment 4, Significant Fitness for Duty Violation or Programmatic Failure/Drug or Alcohol Testing Errors NRC 24 Hour Notification.

2. The notifier shall return the completed original of Attachment 4 to the Fitness for Duty Administrator (Station) for further processing. See Step 6.8.1.

c. Within 24 hours, the Shift Manager shall notify NRC by telephone, telegraph, or facsimile, of any occurrence of an unusual or important event—causally related to Station operation—that indicates or could result in significant environmental impact. See also Step 6.26.2.b. **(North Anna)** **[NAPS EPP 4.1 & 5.4.2]**

d. Within 24 hours after discovery, Licensing (Station) shall notify (see Step 6.3.1.a.) the NRC Regional Office by telephone of failure to notify NRC of planned removal or significant change in the normal operation of equipment that controls the amount of radioactivity in Station effluents **(North Anna)**.

[NAPS Unit 1 License, 2.C(3)(b); Unit 2 License, 2.C(3)(a).]

By the first business day after discovery, Licensing (Station) shall confirm the telephone notification by telegram, mailgram, or facsimile to the NRC Regional Office. See also Step 6.23.6.

e. If any unpermitted, unusual, or extraordinary discharge[1] enters or could be expected to enter State waters, as soon as possible, but not later than 24 hours after discovery, Electric Environmental Services shall notify (see Step 6.3.1.a.) the State Department of Environmental Quality (Water). See also Steps 6.3.4.a.4., 6.3.2.f., and 6.27.3.n. **[VPDES Permit]**

f. If an unplanned bypass (i.e., intentional diversion of waste streams) occurs from any portion of a treatment works, as soon as possible, but not later than 24 hours after the bypass occurs, Electric Environmental Services shall notify (see Step 6.3.1.a.) the State Department of Environmental Quality (Water). **[VPDES Permit]**

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1. Unusual or extraordinary discharge includes, but is not limited to: a) unplanned bypasses, b) upsets, c) spillage of materials resulting directly or indirectly from processing operations or pollutant management activities, d) breakdown of processing or accessory equipment, e) failure of or taking out of service, sewage or industrial waste treatment facilities, auxiliary facilities, or pollutant management activities, or f) flooding or other acts of nature. [**VPDES Permit**]

6.3.7 **Seventy-two Hour Notifications**

If a Notification of Unusual Event is declared:

a. The Station Coordinator Emergency Preparedness shall prepare a Summary Report from information in completed Emergency Plan Implementing Procedures, Control Room logs, and interviews with persons involved with the declaration and response, as appropriate. See Attachment 6, Example DEM Summary Report.

b. The Site Vice President, Director Nuclear Station Safety and Licensing, or Plant Manager (Nuclear) shall approve the report.

c. Nuclear Emergency Preparedness shall ensure the report is delivered to the State Coordinator of DEM within 72 hours after the declaration. **[NAEP 4.4; SEP 4.4]**

6.4 Scheduled Reports

NOTE: Table 2, Directory of Periodic Reports—By Report Name, provides an alphabetical summary of the reports included in this subsection.

6.4.1 **Monthly**

a. Discharge Monitoring Report—See Step 6.27.3.i.

b. Operation Report Meter Readings—See Step 6.27.4.a. **(North Anna)**

c. Sewage Treatment Plant Operation—See Step 6.27.4.b. **(Surry)**

d. Waterworks Operation—See Step 6.27.4.c. **(Surry)**

6.4.2 **Quarterly**

a. Groundwater Pumpage and Use Report—See Step 6.27.3.k. **(Surry)**

b. Temperature Monitoring Data—See Step 6.27.3.q. **(North Anna)**

c. Reactor Oversight Process (ROP) Report—See Subsection 6.30.

d. Operating Data Report—See Subsection 6.30.

6.4.3 **Annual**

a. Changes to the Topical Report, Quality Assurance Program without prior NRC approval—See Step 6.10.5.a.

b. Early Warning System Availability—See Step 6.27.2.c.

c. Environmental Operating Report—See Step 6.26.3. **(North Anna)**

d. Facility changes, tests, and experiments—See Steps 6.10.6. and 6.14.3.

e. Guarantee of deferred premium payment—See Step 6.17.4.

f. North Anna Hydroelectric Project Emergency Action Plan adequacy review—See Step 6.18.4.c. **(North Anna)**

g. North Anna Hydroelectric Project Emergency Action Plan test exercise summary and critique—See Step 6.18.5.b. **(North Anna)**

h. Final Safety Analysis Report Update—See Step 6.10.9.

i. Reports of Individual Monitoring—See Step 6.6.7.

j. Personnel monitoring reports—See Step 6.6.7 **(North Anna)**.

k. Primary coolant activity level exceeding technical specification limits—See Step 6.24.6. **(Surry)**

l. Radiation exposure data to individuals—See Step 6.5.2.

m. Radiological Effluent Release Report—See Step 6.10.3.

n. Radiological Environmental Operating reports—See Step 6.23.9 **(North Anna)**.

o. Property damage insurance or financial security—See Step 6.10.5.d.

p. Shift Manager Responsibility Directive—See Step 6.29.7.

q. Water Withdrawals—See Step 6.27.3.r.

r. Tier II information forms—See Step 6.20.10.b.

s. Financial report —See Steps 6.10.8 and 6.14.11.

t. Emergency Core Cooling System (ECCS) Evaluation Model Changes— See Step 6.10.4.

u. Decommissioning Fund Status Report— See Step 6.27.1.c.

v. Material Balance Report—See Step 6.16.2

w. Voluntary Protection Program Annual Self-Assessment—See Step 6.27.5.b.

x. Fitness for Duty Program Assessment Report—See Step 6.8.4.

6.4.4 **Other**

a. Status of simulator performance tests (every four years)—See Step 6.11.2.

b. Independent consultant report for Lake Anna Dam (every five years)—See Step 6.18.7.

c. EPA Form 8700-13A for hazardous wastes (every even-numbered year)—See Step 6.20.7.a.

d. Decommissioning Fund Status Report (biennial)— See Step 6.10.13.

e. Site Specific Decommissioning Cost Estimate Update (every four years)— See Steps 6.10.13 and 6.27.1.c.

f. ISFSI Safety Analysis Report Update (every 24 months from the date of the issuance of the license)—See Step 6.14.5.

g. Steam Generator Tube Inspection Report (180 days after initial entry into Mode 4 following completion of an inspection per Technical Specification 5.5.9) - See Step 6.23.7.b. **(North Anna)**

Report Name	Addressed At
Decommissioning Reporting and Record Keeping	6.10.13 & 6.27.1.c.
Discharge Monitoring	6.27.3.i.
EAP Test Exercise Summary and Critique **(North Anna)**	6.18.5.b.
ECCS Evaluation Model Changes	6.10.4
Effluent Releases **(ISFSI)**	6.10.3.a.
Early Warning System Availability	6.27.2.c.
Environmental Operating **(North Anna)**	6.26.3
EPA Form 8700-13A for hazardous wastes	6.20.7.a.
Final Safety Analysis Report Update	6.10.9
Fracture Toughness of Reactor Coolant Pressure Boundary	6.10.14
Groundwater Pumpage and Use **(Surry)**	6.27.3.k.
Guarantee of Deferred Premium Payment	6.17.4
Independent Consultant Report (Lake Anna Dam)	6.18.7
ISFSI Safety Analysis Report Update	6.14.5
Material Status and Physical Inventory Listing	6.16.2
Nuclear Liability Financial Protection, Proof of	6.17.2
Operation Report Meter Readings **(North Anna)**	6.27.4
Personnel Exposure and Monitoring	6.6.7
Primary Coolant Activity Exceeds Limits **(Surry)**	6.24.6.
Property Insurance, Present Levels and Sources of	6.10.5.d.
QA Topical Report Changes	6.10.5.a.
Radiation Exposure to Individuals	6.5.2
Radiological Effluent Release	6.10.3
Radiological Environmental Operating **(North Anna)**	6.23.9
Sewage Treatment Plant Operation **(Surry)**	6.27.4.b.
Shift Supervisor Responsibility Directive **(North Anna)**	6.29.7
Steam Generator Tube Inspection Reports	6.23.7 & 6.24.14
Temperature Monitoring Data **(North Anna)**	6.27.3.q.
Tier II information forms	6.20.10.b.
Water Withdrawals	6.27.3.r.
Waterworks Operation **(Surry)**	6.27.4.c.

Table 2
Directory of Periodic Reports—By Report Name

6.5 10 CFR 19, Notices, Instructions, and Reports to Workers; Inspections

6.5.1 10 CFR 19.11—Posting of Notices to Workers

a. Licensing (Station) shall post a notice that describes the following documents and states where current copies may be examined:

- 10 CFR 19
- 10 CFR 20
- The license, license conditions, or documents incorporated into a license by reference, including amendments
- Plant procedures applicable to licensed activities

NOTE: Steps 6.5.1.b. and 6.5.1.c. provides the notice to employees required by 10 CFR 30, 40, 50, 70.7(e), 72.10(e)(1), and 150.

b. Licensing (Station) shall post Form NRC-3, “Notice to Employees,” at the Station.

c. Nuclear Licensing and Operations Support shall post Form NRC-3, “Notice to Employees,” on the NL&OS bulletin board.

d. The postings required by Steps 6.5.1.a. and 6.5.1.b. shall be at these locations:

North Anna

- HP dosimetry issue hallway
- Materials building entrance
- Processing Center (TSB) entrance
- Secondary security access control building (when activated)
- Security building hallway
- Training building hallway

Surry

- Administration building
- Clean change room
- Machine shop bulletin board
- Radwaste facility
- Secondary access area
- Service building hallway bulletin board

e. Within one business day after receipt or dispatch, Nuclear Licensing and Operations Support shall provide Licensing (Station) a copy of NRC:
   - Notices of Violation
   - Proposed impositions of civil penalty
   - Orders for imposing requirements
   - Orders modifying, suspending, revoking a license
   - Orders imposing a civil penalty, and responses thereto

f. Upon receipt of documents that involve radiological working conditions identified in Step 6.5.1.e. above, Licensing (Station) shall post copies at the following locations within two business days after receipt from or dispatch to NRC and shall remain for five business days or until action to correct the violation is completed, whichever is later.
   - HP dosimetry issue hallway **(North Anna)**
   - Secondary security access control building (when activated) **(North Anna)**
   - Security building hallway **(North Anna)**
   - The same locations specified for Surry at Step 6.5.1.d. **(Surry)**

g. NRC Documents identified in Step 6.5.1.e. above that do not involve radiological working conditions may be referenced in a memorandum posted in the locations identified in Step 6.5.1.f.

h. Licensing (Station) shall document item postings and removal.

i. Licensing (Station) shall replace posted items that are defaced or altered.

6.5.2 **10 CFR 19.13(b)—Radiation Exposure Data to Individuals**

Annually, Radiological Protection shall inform each worker of their exposure to radiation or radioactive material as shown in records maintained by Radiological Protection.

6.5.3 **10 CFR 19.13(c)—Radiation Exposure Data to Former Employees/Workers**
At the request of an individual formerly engaged in licensed activities controlled by the Station, Radiological Protection shall provide the individual a report of their exposure to radiation or radioactive material.

a. The report shall be provided within 30 days after the request or within 30 days after the Station determines the individual's exposure, whichever is later.

b. The report shall cover the period during which the individual's activities involved exposure to radiation or radioactive materials licensed by NRC.

c. The report shall include the dates and locations of the individual's participation in licensed activities during the period.

6.5.4 **10 CFR 19.13(d)—Radiation Exposure Data for Overexposures**
If an individual's exposure data is sent to NRC to comply with 10 CFR 20.2202, 20.2203, 20.2204, or 20.2206 (see Step 6.6.4), Radiological Protection shall provide the same exposure data to the employee or worker, in writing, no later than its transmittal to NRC.

6.5.5 **10 CFR 19.13(e)—Radiation Exposure Data to Terminating Individuals**

a. At the request of a terminating worker, Radiological Protection shall provide the worker or worker's designee a report of the radiation dose received by that worker in connection with Dominion operations during the current calendar year or fraction thereof.

b. If actual dose has not been determined, Radiological Protection shall provide a written dose estimate. Dose estimates shall be clearly identified as estimates.

6.6 10 CFR 20, Standards for Protection Against Radiation

6.6.1 **10 CFR 20.1906, Procedures for Receiving and Opening Packages**

a. **Removable Radioactive Surface Contamination**
See Step 6.3.2.b.

b. **External Radiation Levels**
See Step 6.3.2.b.

6.6.2 **10 CFR 20.2201, Reports of Theft or Loss of Licensed Material**

a. **Immediate Notification**
See Step 6.3.2.a.3.

b. **Thirty-day Notification**

Within 30 days after it becomes known that there is any lost, stolen, or missing licensed material—if licensed material in a quantity greater than 10 times the quantity specified in 10 CFR 20.1001-20.2401, Appendix C, is still missing—the Shift Manager shall notify the NRC Operations Center by telephone. **[10 CFR 20.2201(a)(ii)]**

c. **Thirty-day Report**

If a telephone notification is made in accordance with Step 6.3.2.a.3. or Step 6.6.2.b., within 30 days after the telephone notification, a report shall be submitted by an LER, as specified in Step 6.10.11.c. The report shall include: **[10 CFR 20.2201(b)]**

- A description of the licensed material involved, including kind, quantity, and chemical and physical form
- A description of the circumstances under which the loss or theft occurred
- A statement of disposition or probable disposition of the licensed material involved
- Radiation exposures to individuals, circumstances under which the exposures occurred, and the possible total effective dose equivalent to persons in unrestricted areas
- Actions that have been taken, or will be taken, to recover the material
- Procedures or measures that have been or will be adopted to prevent a recurrence of the loss or theft of licensed material

d. **Supplemental Reports**

Within 30 days, a supplemental report shall be submitted, by an LER, if any substantial, new information becomes known. **[10 CFR 20.2201(d)]**

e. **Personal Identity Information**

The names of individuals who may have received radiation exposure shall be stated in a separate and detachable part of reports. **[10 CFR 20.2201(e)]**

6.6.3 **10 CFR 20.2202, Notifications of Incidents**

a. **Immediate Notification**

See Step 6.3.2.a.4.

b. **Twenty-four Hour Notification**

See Step 6.3.6.a.1.

c. **Thirty-day Reports**

1. Reports of events that require notifications in accordance with Step 6.6.3.a. or Step 6.6.3.b. shall be submitted to NRC as specified in Step 6.6.4.

2. The names of individuals who have received radiation exposure shall be stated in a separate and detachable part of reports submitted to NRC. **[10 CFR 20.2202(c)]**

6.6.4 **10 CFR 20.2203, Reports of Overexposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Limits**

a. Within 30 days, a report shall be submitted by an LER, as specified in Step 6.10.11.c., for **[10 CFR 20.2203(a) and 10 CFR 20.2203(c)]:**

1. Any incident for which Step 6.6.3 requires NRC notification.

2. Doses in excess of any of the following (see VPAP-2101, Radiation Protection Program):
   - The occupational dose limits for adults in 10 CFR 20.1201
   - The occupational dose limits for a minor in 10 CFR 20.1207
   - The limits for an embryo/fetus of a declared pregnant woman in 10 CFR 20.1208
   - The limits for an individual member of the public in 10 CFR 20.1301
   - Any applicable limit in the license
   - Any ALARA constraints for air emissions established under 10 CFR 20.1101(d)

3. Levels of radiation or concentrations of radioactive material in:
   - A restricted area in excess of any applicable limit in the license
   - An unrestricted area in excess of 10 times any applicable 10 CFR 20 or license limit (whether or not involving exposure of any individual that exceeds 10 CFR 20.1301 limits)

NOTE: Reports submitted to NRC in accordance with Step 6.10.11.b.10. or Step 6.10.11.b.11. fulfill the effluent release reporting requirements of Step 6.6.4.a.4. **[10 CFR 20.2203(c)]**

4. Levels of radiation or releases of radioactive material that exceed 40 CFR 190 limits, or license conditions related to 40 CFR 190 limits. [**10 CFR 20.2203(a)(4)]**

b. Each report required by 10 CFR 20.2203(a) shall describe the extent of exposure of individuals to radiation or radioactive materials, including **[10 CFR 20.2203(b)]:**
   - Estimates of each individual's dose
   - Levels of radiation and concentrations of radioactive material involved
   - The cause of the elevated exposures, dose rates, or concentrations
   - Corrective steps taken or planned to prevent a recurrence, including the schedule for achieving conformance with applicable limits, generally applicable environmental standards, and associated license conditions

c. For each individual exposed, the report shall include, in a separate and detachable part, the individual's name, Social Security number, date of birth, and an estimate of the individual's exposure. **[10 CFR 20.2203(b)(2)]**

d. If an event involves radiological overexposure or exposure to radioactive materials, see Step 6.5.4.

e. If an event involves radiological exposure and the DEM has not already been notified, see Step 6.27.2.

6.6.5 **10 CFR 20.2204, Reports of Planned Special Exposures**

If a planned special exposure is conducted in accordance with VPAP-2101, Radiation Protection Program:

a. Radiological Protection shall prepare a report to inform NRC that a planned special exposure was conducted in accordance with 10 CFR 20.1206, and indicating the date the planned special exposure occurred and:
   - A description of the exceptional circumstances that required use of a planned special exposure
   - The name of the member of management who authorized the planned special exposure and a copy of the signed authorization
   - What actions were necessary
   - Why the actions were necessary
   - How doses were maintained ALARA
   - What individual and collective doses were expected, and what doses were actually received in the planned special exposure

b. Nuclear Licensing and Operations Support shall review the report.

c. The Senior Vice President Nuclear Operations shall approve the report.

d. Within 30 days after the planned special exposure is conducted, Nuclear Licensing and Operations Support shall submit the report to the NRC Regional Office.

6.6.6 **10 CFR 20.2205, Reports to Individuals of Exceeding Dose Limits**

a. Copies of reports to the NRC, pursuant to the provisions of 10CFR20.2203, 20.2204, or 20.2206, of any exposure of identified occupationally exposed individual(s), or identified member(s) of the public to radiation or radioactive material, shall be provided to the individual(s).

b. The reports shall be transmitted to the individual(s) at a time no later than the transmittal to the NRC.

6.6.7 **10 CFR 20.2206, Reports of Individual Monitoring**

NOTE: This report can be submitted in conjunction with the Primary Coolant Activity Level Exceeding T.S. Limits Report—See Step 6.24.6 **(Surry)**

a. Radiological Protection shall prepare a calendar year report of the results of individual monitoring performed for each individual for whom monitoring was required in accordance with 10 CFR 20.1502. The report may include data for individuals for whom monitoring was provided, but not required.

b. The report shall be prepared on Form NRC 5 or electronically, including all the information required by Form 5.

c. Licensing (Station) shall review the report.

d. The Director Nuclear Station Safety and Licensing shall approve the report.

e. By the end of April, Licensing (Station) shall submit the report for the preceding year to the REIRS Project Manager, Office of Nuclear Regulatory Research.

6.6.8 **10 CFR 20.2207, Reports of Transactions Involving Nationally Tracked Sources**

NOTE: Each licensee is required to reconcile its on-site inventory of nationally tracked sources with the information previously reported to the National Source Tracking System. Each licensee must compare the information contained in the system to its own inventory, including a check of the model and serial number of each source. This reconciliation does not require the licensee to conduct an additional physical inventory of its sources.

a. Radiological Protection shall perform annual reconciliation in January.

b. Licensing (Station) shall review the report.

c. The Manager Radiological Protection shall approve the report.

d. Reconciliation reporting can be performed electronically or in hard copy form.

e. By January 31 of each year, Radiological Protection shall submit to the National Source Tracking System confirmation that the data in the National Source Tracking System is correct.

6.7 10 CFR 21, Reporting of Defects and Noncompliance

6.7.1 10 CFR 21.6, Posting requirements

a. Licensing (Station) and Nuclear Licensing and Operations Support, respectively, shall post current copies of these documents at the Station and at Innsbrook:

- The regulations in 10 CFR 21
- Section 206 of the Energy Reorganization Act of 1974
- Procedures that implement the requirements of 10 CFR 21

NOTE: A notice—posted in lieu of the specified documents—with content similar to that described in LI-AA-301, Implementation of 10 CFR 21, Reporting of Defects and Noncompliance, will fulfill the requirements of Step 6.7.1.a.

b. Notices shall be posted at these locations:

Innsbrook Technical Center

- Nuclear Licensing and Operations Support bulletin board

North Anna

- HP dosimetry issue hallway
- Materials building entrance
- Processing Center (TSB) entrance
- Secondary security access control building (when activated)
- Security building hallway
- Training building hallway

Surry

- Administration Building
- Clean Change Room wall
- Machine Shop bulletin board
- Radwaste facility
- Secondary Access Area
- Service Building hallway bulletin board

6.7.2 **10 CFR 21.21, Notification of Failure to Comply or Existence of a Defect**

LI-AA-301, Implementation of 10 CFR 21 Reporting of Defects and Noncompliance, provides the method by which potential defects or failures to comply (noncompliance) associated with a basic component are evaluated in order to identify a defect that could create a substantial safety hazard as specified in 10 CFR 21, were it to remain uncorrected, and to identify the reporting and record keeping requirements associated with said evaluations.

6.8 10 CFR 26, Fitness for Duty Programs

6.8.1 **Significant Events**

a. If the Corporate Fitness for Duty Administrator becomes aware of a significant fitness for duty event that involves corporate personnel subject to 10 CFR 26, the Administrator shall determine at which stations the individual has unescorted access. Immediately, the Corporate Fitness for Duty Administrator shall notify either or both Station Fitness for Duty Administrators who shall coordinate a single notification.

b. If a Station Fitness for Duty Administrator becomes aware of a significant fitness for duty event that involves personnel with unescorted access to the Station, they shall:

   1. Immediately prepare Section A of Attachment 4, Significant Fitness for Duty Violation or Programmatic Failure/Drug or Alcohol Testing Errors NRC 24 Hour Notification.

   2. Immediately, submit the form to the Site Vice President or a Director.

c. See Step 6.3.6.b.

d. Promptly after Attachment 4 is returned by the notifier, the Station Fitness for Duty Administrator shall send a copy of Attachment 4 to the Fitness for Duty Program Manager, the Site Vice President, and the Director NL&OS.

6.8.2 **NRC Employee Potentially Unfit for Duty**

See Step 6.3.2.c.

6.8.3 Drug and Alcohol Testing Errors

a. Within 30 days of completing an investigation of any testing errors or unsatisfactory performance discovered in performance testing at either a Dominion testing facility or an HHS-certified laboratory, in the testing of quality control or actual specimens, or through the processing of FFD policy violation and MRO reviews, as well as any other errors or matters that could adversely reflect on the integrity of the random selection or testing process, a report of the incident and the corrective actions taken or planned shall be submitted to the NRC. **[10 CFR 26.719(c)(1)]**

b. The FFD Administrator (Corporate) shall prepare a report that describes the testing errors or unsatisfactory performance.

c. The Fitness for Duty Program Manager shall review the report and forward it to the Senior Vice President Nuclear.

d. The Senior Vice President shall submit the report to the NRC Regional Office.

e. Within 24 hours after discovery of a false positive error on a blind performance test sample submitted to an HHS certified laboratory, notification shall be made to Station Management by the Fitness for Duty Administrator (Corporate). The Shift Operations Manager or a Station Management Staff member shall report the error to the NRC Operations Center in accordance with Step 6.3.6.b. **[10 CFR 26.719(c)(2)]**

f. Within 24 hours after discovery of a false negative on a quality assurance check of validity screening tests, notification shall be made to Station Management by the Fitness for Duty Administrator (Corporate) The Shift Operations Manager or a Station Management Staff member shall report the error to the NRC Operations Center in accordance with Step 6.3.6.b. **[10 CFR 26.719(c)(3)]**

g. Non reportable indicators of FFD programmatic weaknesses shall be documented in the Corrective Action Program, trended and corrected.

6.8.4 FFD Program Annual Performance Data Report

a. As soon as possible after December 31, the FFD Program Annual Performance Data Report for January through December shall be submitted to the NRC annually before March 1 of the following year. The report shall contain the following information: **[10 CFR 26.717]**

- The FFD program performance data must include:
  - Random testing rate
  - Drugs tested with cutoff levels
  - Populations tested with results sorted by population
  - Test conditions
  - Substances identified
  - Number of subversion attempts
  - Summary of management actions
  - Number of work hour Waivers issued with summary of distribution per individual within each category
  - Summary of corrective actions resulting from the analyses of the Waiver data, including fatigue assessments
- Summary of management actions implement to correct identified program weaknesses
- Number of terminations and administrative actions taken against individuals
- Test results of positive initial drug tests by processing stage (e.g., initial test, testing at lab, MRO determination)

b. The Fitness for Duty Manager shall review the report.

c. The Director NL&OS shall submit the report to the NRC Regional Office before March 1 of the following year.

6.9 10 CFR 31, General Domestic Licenses for Byproduct Material [Commitment 3.2.23]

10 CFR 31.5 covers certain detecting, measuring, gauging or controlling devices and certain devices for producing light or an ionized atmosphere. Examples include self-luminous exit signs, smoke detectors, metal analyzers, dew point meters, and fluorotracers.

6.9.1 A report shall be submitted to the NRC within 30 days, or as specified in any request, in accordance with 31.5 in the event of:

- Theft, loss, damage, or indication of possible failure or damage,
- Transfer or disposal of device,
- Detection of removable contamination in excess of specified limits,
- Request from the NRC for annual registration,
- Request from the NRC for additional information, or
- Change of address of device location.

6.9.2 Radiation incidents, loss, or theft shall be reported in accordance with 10 CFR 20.2201 and 2202.

6.10 10 CFR 50, Domestic Licensing of Production and Utilization Facilities

6.10.1 10 CFR 50.9, Completeness and Accuracy of Information

Any information identified as having a significant implication for public health and safety or common defense and security, not already required to be provided to NRC by other reporting or updating requirements, shall be reported to the NRC Regional Office within two business days after the information is identified. See also Step 6.29.6.

6.10.2 10 CFR 50.36, Technical Specifications

a. Shutdown because a safety limit is exceeded—see Steps 6.3.2.a.5., 6.23.3 and 6.24.3 **[10 CFR 50.36(d)(1)(i)(A) & (d)(7)]**

NOTE: Specific Safety Limit Violation report requirements are included at Step 6.23.3. **(North Anna)**

b. Automatic safety system—see Steps 6.3.3.b. and 6.10.2.a. **[10 CFR 50.36(d)(1)(ii)(A) & (d)(7)]**

c. Shutdown due to an LCO—see Steps 6.3.4.1. and 6.10.2.a. **[10 CFR 50.36(d)(2) & (d)(7)]**

6.10.3 **10 CFR 50.36a, Technical Specifications on Effluents**

a. Radiological Protection shall prepare an annual Radiological Effluent Release report covering the operation of the unit in the previous year that demonstrates compliance with requirements as specified in VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry) , and VPAP-2104, Radioactive Waste Process Control Program (PCP). The material shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV. B. 1. **[NAPS TS 5.6.3; SPS TS 6.6.B.3]** The report shall:

- Summarize on a quarterly basis, in the format of Regulatory Guide 1.21, Appendix B[1], the quantities of radioactive liquid and gaseous effluents released from each unit[2] **[NAPS TS 5.6.3; SPS TS 6.6.B.3]**
- Summarize on an annual basis, in the format of Regulatory Guide 1.21, Appendix B[1], the quantity of solid waste released from each unit[2]**[NAPS TS 5.6.3; SPS TS 6.6.B.3]**
- State the quantity of each principal radionuclide released to unrestricted areas in liquid and gaseous effluents (including the ISFSI) **[10 CFR 72.44(d)(3), SPS ISFSI TS App. C, 1.4.1, and NAPS ISFSI TS 5.5.2]**
- Discuss releases for which a report was required by Step 6.10.16
- Provide an assessment[3] of radiation dose to the maximum exposed members of the public due to radioactive liquid and gaseous effluents during the previous calendar year.
- List unplanned releases of liquid or gaseous radioactive material, from the site to unrestricted areas, that exceeded the effluent dose rate limitations as specified in VPAP-2103
- Provide any supplemental information NRC may need to estimate maximum potential annual radiation doses to the public from effluents
- Explain any failures to correct inoperable radioactive gaseous effluent monitoring instrumentation within the limits specified in VPAP-2103

(continued)

1. Where the Regulatory Guide refers to Technical Specification limits, use the values in VPAP-2103.
2. A single submittal may be made for both units; material common to both units shall be combined. For units with separate radwaste systems the submittal shall specify the releases of radioactive material from each unit.
3. The assessment method shall be as specified in VPAP-2103.

Step 6.10.3.a (continued)

- If samples were unavailable for the Radiological Environmental Monitoring Program, as specified in VPAP-2103, identify the cause of the unavailability of samples and identify the new locations for obtaining replacement samples. If not implemented during the reporting period in a revision to VPAP-2103, provide a copy of the figures and tables that reflect the new locations
- If new land use census locations are added to comply with VPAP-2103, identify the new locations and, if not reflected in an enclosed submittal of VPAP-2103, provide a copy of the figures and tables that reflect the new locations
- Describe any major changes to radioactive liquid, gaseous, or solid waste treatment systems, initiated by Dominion and reviewed by FSRC during the reporting period. Include[1]:
  - •• A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59
  - •• Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information
  - •• A detailed description of the equipment, components, and processes involved, and interfaces with other Station systems (continued)
  - •• An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents or in solid waste quantity that differs from those previously predicted in the license
  - •• An evaluation of the change which shows the expected maximum exposures to an individual in the unrestricted area that differ from those previously estimated in the license application and amendments thereto
  - •• A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to the changes
  - •• An estimate of plant operating personnel exposure due to the change
  - •• Documentation of FSRC review and acceptance
- Include a copy of VPAP-2103 if revised during the reporting period (as specified in Step 6.23.12)

b. Licensing (Station) shall review the report.

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1. Alternatively, this information may be provided in the UFSAR.

c. The Director Nuclear Station Safety and Licensing shall approve the report.

d. By May 1 (March 1-Surry ISFSI and May 1-North Anna ISFSI), Licensing (Station) shall submit the report for the previous year to NRC, to the NRC Regional Office, and for the ISFSI, to the NRC Director, Office of Nuclear Material Safety and Safeguards. **[SPS ISFSI TS App. C, 1.4.1 & NAPS ISFSI TS 5.5.2]**

6.10.4 **10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors**

a. Nuclear Analysis and Fuel shall prepare a Report of Emergency Core Cooling System (ECCS) Evaluation Changes Pursuant to the Requirements of 10 CFR 50.46 for each calendar year. For each change or error discovered in an acceptable evaluation model that affects the peak clad temperature calculation, the report shall document the nature of the change or error and its estimated effect on the limiting ECCS analysis. **[10 CFR 50.46(a)(3)(ii)]**

   Note that if the change or error is significant, this shall be reported within 30 days and include a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with the 50.46 requirements. A significant change or error is one that results in a calculated peak fuel cladding temperature different by more than 50 degrees F from the temperature calculated for the limiting transient using the last acceptable model or is a culmination of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 degrees F. **[10 CFR 50.46(a)(3)(ii)]**

b. The Site Vice President shall review the report.

c. The Director NL&OS shall approve the submittal.

d. By June 30, Nuclear Licensing and Operations Support shall submit the report for the previous year to the NRC.

6.10.5 **10 CFR 50.54, Conditions of Licenses**

a. If the quality assurance program description in the UFSAR requires prior NRC approval, Nuclear Licensing and Operations Support shall prepare a submittal that includes all pages affected by that change and a forwarding letter. **[10 CFR 50.54(a)(4)]**

1. The forwarding letter shall identify the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy 10 CFR 50, Appendix B and associated commitments accepted by NRC.
2. The forwarding letter is not required to provide the basis for changes that correct spelling, punctuation or editorial items.
3. FSRC, the Site Vice President, Nuclear Oversight, and Nuclear Licensing and Operations Support shall review the submittal.
4. The Senior Vice President Nuclear Operations shall approve the submittal.

b. If the Security Plan is revised without prior NRC approval, as permitted by 10 CFR 50.54(p)(1):

1. Security shall prepare a report that describes the change(s). See also Step 6.14.12.
2. The Director Nuclear Security and Emergency Preparedness shall review and approve the report.
3. The Senior Vice President Nuclear shall approve the submittal.
4. Within two months after the change is implemented, Nuclear Licensing and Operations Support shall submit the report to NRC. **[10 CFR 50.54(p)(2)]**

c. If the Emergency Plan is revised without prior NRC approval, as permitted by 10 CFR 50.54(q) or **(ISFSI)** 10 CFR 72.44(f):

1. The Director Nuclear Protection Services and Emergency Preparedness shall initiate submittal of the revised Plan.
2. Within 30 days following the assigned effective date, Nuclear Licensing and Operations Support shall submit the revised Plan to NRC and to the NRC Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards. **[10 CFR 50.54(q) & 10 CFR 72.44(f)]**

d. Each year, Corporate Risk Management shall prepare a report of the current levels of Station property damage insurance or financial security that Dominion maintains and the sources of this insurance or financial security.

1. Nuclear Licensing and Operations Support shall review the report.
2. The Director NL&OS shall approve the report.
3. By April 1, Nuclear Licensing and Operations Support shall send the report for the preceding year to NRC. **[10 CFR 50.54(w)(3)]**

e. If action is taken as permitted by 10 CFR 50.54(x), see Step 6.3.3.a.

f. Not later than five years before the reactor operating license expires, Nuclear Licensing and Operations Support shall prepare and submit to NRC for review and approval a report that describes the program for managing and funding management for irradiated fuel after the license expires, until title is passed to the Secretary of Energy. **[10 CFR 50.54(bb)]**

g. If a voluntary or involuntary petition is filed under any chapter of Title 11 of the United States Code by or against: **[10 CFR 50.54(cc) & 10 CFR 72.44(b)(6)(i)]**

- Dominion
- Dominion Resources
- An affiliate of Dominion

Nuclear Licensing and Operations Support shall prepare and submit a report to the NRC Regional Administrator immediately following the filing. The report shall indicate:

- The bankruptcy court in which the petition was filed
- The date the petition was filed

6.10.6 10 CFR 50.59, Changes, Tests, and Experiments

a. For each calendar year, Licensing (Station) shall prepare a report to NRC that contains a brief description of changes, tests, and experiments for which Regulatory Evaluations were prepared, and a summary of each Regulatory Evaluation.

b. The Director Nuclear Station Safety and Licensing shall approve the report.

c. By March 31 of the following year, Licensing (Station) shall submit the report to NRC.

6.10.7 **10 CFR 50.61, Fracture Toughness Requirements for PTS Protection**

Whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values of RT_{PTS} (see also Step 6.10.14):

a. NAF shall prepare a revised assessment to reflect the changes.

b. Nuclear Licensing and Operations Support shall review the revised assessment.

c. FSRC shall approve the revised assessment.

d. Nuclear Licensing and Operations Support shall submit the revised assessment to NRC.

6.10.8 **10 CFR 50.71(b), Financial Report**

a. Upon issuance of each Dominion Annual Report (including certified financial statements), Accounting Research and Control shall immediately send a copy to the Senior Vice President Nuclear Operations.

b. Nuclear Licensing and Operations Support shall submit the report to NRC and to the NRC Office of Nuclear Material Safety and Safeguards. **[10 CFR 50.71(b) & 10 CFR 72.80(b)]**

6.10.9 **10 CFR 50.71(e), Final Safety Analysis Report Updating**

a. Annually or 6 months after each refueling outage, provided the interval between successive updates does not exceed 24 months, Nuclear Engineering (ISI/DBD/UFSAR) shall prepare a report to the NRC that includes Updated Final Safety Analysis Report pages that have been updated as delineated in 10 CFR 50.71(e).

b. An NBU Officer of the Company, as designated in LI-AA-200, NRC Licensing Correspondence, shall approve the submittal.

c. By October 15 of every year, NL&OS shall submit the report to the NRC.

6.10.10 **10 CFR 50.72, Immediate Notification Requirements**

See Subsection 6.3.

6.10.11 **10 CFR 50.73, Licensee Event Report System**

NOTE: Attachment 5, 10 CFR 50.73 Reportability Guidelines, provides examples and more detailed guidance for interpreting the requirements of parts of this Step.

a. Specified events shall be reported with an LER regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

b. Dominion shall report:

1. Completion of any plant shutdown required by the Technical Specifications **[10 CFR 50.73(a)(2)(i)(A)].**

2. Initiation of plant shutdown (reduction of power or temperature) required by Technical Specifications because a safety limit is exceeded, an automatic safety system does not function as required, or a limiting condition for operation is not met. **[10 CFR 50.36(d)(1)(i)(A), (d)(1)(ii)(A), (d)(2) & (d)(7)]**

3. An operation or condition prohibited by Technical Specifications except when **[10 CFR 50.73(a)(2)(i)(B)]**:
   - The Technical Specification is administrative in nature;
   - 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
   - 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

4. A deviation from the Technical Specifications authorized by 10 CFR 50.54(x). **[10 CFR 50.73(a)(2)(i)(C)]**

5. Any event or condition that resulted in the condition of the plant, including its principal safety barriers, being seriously degraded, or that resulted in the plant being in an unanalyzed condition that significantly degraded plant safety. **[10 CFR 50.73(a)(2)(ii)]**

6. A natural phenomenon or other external condition that posed an actual threat to the safety of the Station or significantly hampered Station personnel in the performance of duties necessary for the safe operation of the Station.
[10 CFR 50.73(a)(2)(iii)]

7. Any event or condition that resulted in manual or automatic actuation of any of the following systems, except when:
   - The actuation resulted from and was part of a preplanned sequence during testing or reactor operation

     or

   - The actuation was invalid (see Subsection 4.2) **and**:
     - •• Occurred while the system was properly removed from service

       or

     - •• Occurred after the safety function had already been completed

   [10 CFR 50.73(a)(2)(iv)(A)]

   - Reactor Protection System (RPS)

   - General containment isolation signals affecting containment isolation valves in more than one system or multiple Main Steam Isolation Valves (MSIVs)

   - Emergency core cooling systems (ECCS) including: HHSI and LHSI systems.

   - Auxiliary feedwater system.

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

   - Emergency Diesel Generators (EDGs)

NOTE: Events in Step 6.10.11.b.8. may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

8. An event or condition that could have prevented fulfillment of the safety function of structures or systems needed to **[10 CFR 50.73(a)(2)(v) and (vi)]**:
   - Shut down the reactor and maintain it in a safe shutdown condition
   - Remove residual heat
   - Control the release of radioactive material, or
   - Mitigate the consequences of an accident

9. An 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 **[10 CFR 50.73(a)(2)(vii)]**:
   - Shut down the reactor and maintain it in a safe shutdown condition, or
   - Remove residual heat, or
   - Control the release of radioactive material, or
   - Mitigate the consequences of an accident

NOTE: LERs submitted in accordance with Step 6.10.11.b.10. or Step 6.10.11.b.11. also fulfill the effluent release reporting requirements of Step 6.6.4.a.4., subject to the content requirements of Step 6.6.4.b.

10. An airborne radioactivity release that, when averaged over a period of one hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration of the limits specified in 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1. **[10 CFR 50.73(a)(2)(viii)(A)]**

11. A liquid effluent release that, when averaged over a period of one hour, exceeded 20 times the applicable concentrations specified in 10 CFR 20.1001-20.2401, Appendix B, 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. **[10 CFR 50.73(a)(2)(viii)(B)]**

NOTE: Events in Step 6.10.11.b.12. may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, component failures need not be reported if the event results from a shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or normal and expected wear or degradation.

12. 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: [10 CFR 50.73(a)(2)(ix)]
    - Shut down the reactor and maintain it in a safe shutdown condition;
    - Remove residual heat;
    - Control the release of radioactive material; or
    - Mitigate the consequences of an accident

13. An event that posed an actual threat to the safety of the plant or significantly hampered Station personnel in the performance of duties necessary for the safe operation of the Station including: **[10 CFR 50.73(a)(2)(x)]**
    - Fires
    - Toxic gas releases
    - Radioactive releases

c. Within 60 days after the discovery of the event, Licensing (Station) shall prepare, obtain FSRC and Site Vice President approval of, and submit an LER to NRC for any event specified in Step 6.10.11.b.

1. LER preparation shall comply with the requirements of 10 CFR 50.73(b), (c), (d), and (e). **[Commitments 3.2.5 and 3.2.6]**

2. If the end of the 60-day period falls on a holiday or weekend, the LER may be mailed on the first business day following the end of the 60 days.

3. If an LER is used to submit a report for an issue with an earlier time limit for reporting, the earlier limit shall apply.

d. Site Engineering (and other Station and corporate organizations as appropriate) shall support LER preparation as requested by Licensing (Station).

e. The EPIX Coordinator shall determine whether component failures or malfunctions should be shown as EPIX reportable on an LER. **[Commitment 3.2.3]**

f. If an LER is submitted pursuant to 10 CFR 20.2203, at the same time the LER is submitted to NRC, Licensing (Station) shall send a copy of the LER to the individual involved.

g. If it becomes appropriate or necessary to withdraw an LER, Licensing (Station) shall prepare an explanatory letter, in a format similar to that used to submit the LER but without any special LER notations. Review, approval, and submittal for the letter shall be the same as for the LER being withdrawn.

6.10.12 **10 CFR 50.74, Change in Licensed Operator Status**

a. The Manager Nuclear Training shall prepare a letter of notification for Licensing (Station) if an NRC-licensed operator:

- Ceases to be employed by Dominion **[Commitment 3.2.11]**
- Is permanently reassigned from a position for which Dominion has certified the need for such a license **[10 CFR 55.31(a)(3)]**
- Develops a physical condition that may permanently and adversely affect performance of assigned licensed duties or may cause operational errors. Notification is not required for temporary disabilities provided the licensee is administratively prevented from performing licensed duties during the period of the licensee’s disability. **[NUREG 1021, 10 CFR 55.25; 10 CFR 55.33(a)(1)]**
- Develops a physical condition that requires a conditional license **[10 CFR 55.23]**

b. The Manager Nuclear Operations with the assistance of the Manager Nuclear Training (if deemed appropriate) shall prepare a letter of notification for Licensing (Station) if an NRC-licensed operator:

- Develops a mental condition that may that may permanently and adversely affect performance of assigned licensed duties or may cause operational errors. Notification is not required for temporary disabilities provided the licensee is administratively prevented from performing licensed duties during the period of the licensee's disability. **[NUREG 1021, 10 CFR 55.25; 10 CFR 55.33(a)(1)]**

NOTE: The following confirmed positive chemical test notification may be superseded by a related NRC request for information.

- Has a confirmed positive chemical test as a result of Fitness for Duty testing
- Is convicted of a felony **[10 CFR 55.33]** (See also Steps 6.3.4.a.4., 6.8.1, and 6.27.2.a.)

c. The Site Vice President shall review the letter of notification for Step 6.10.12.a.

d. The Site Vice President shall review the letter of notification for Step 6.10.12.b.

e. The Site Vice President shall approve the letter of notification.

f. Within 30 days after the event, Licensing (Station) shall submit notification to NRC.

6.10.13 **10 CFR 50.75, Reporting and Record Keeping for Decommissioning Planning**

NOTE: Additional decommissioning reporting to the State Corporation Commission and FERC are contained in Step 6.27.1.

NOTE: Treasury is responsible for establishing and maintaining the Decommissioning Trust Fund and any required change to the certification of financial assurance.

a. Once every two years a report shall be submitted to the NRC providing a status of the decommissioning fund for each nuclear unit in accordance with 10 CFR 50.75(f)(1). Treasury shall provide NL&OS an annual Decommissioning Trust Fund Status Report, along with a description of any changes to the method of providing financial assurance, in order to develop the status report.

1. The Site Vice President shall review the submittal.

2. The Director NL&OS shall approve the submittal.
3. By March 31 every other year (odd years), NL&OS shall submit the report to the NRC.

b. NL&OS shall update the Site Specific Cost Studies every four years to be available as input to the Decommissioning Trust Fund Status Report. The studies may be used by the Company for determining the appropriate level of revenue collection necessary to assure financial ability to decommission the facilities. The decommissioning alternatives considered by Dominion shall be in accordance with those options allowed by 10 CFR 50.82. The report should be produced in accordance with the AIF/NESP “Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates” **(Reference 3.1.93)** and Regulatory Guide 1.159, “Assuring the Availability of Funds for Decommissioning Nuclear Reactors”, dated August 1990 **(Reference 3.1.94)**. The report shall contain as a minimum:

1. Cost breakdown between radiological and non-radiological cleanup.
2. ISFSI related costs.
3. Major activities need to be broken down (e.g., labor, waste burial, equipment, engineering).
4. An estimated schedule shall be provided for each evaluated method.
5. A cost comparison between the most recent recommended method and the current recommended method detailing the differences in cost.

c. At least five years prior to the projected end of operation, a preliminary decommissioning cost estimate shall be submitted to the NRC in accordance with 10 CFR 50.75(f)(3).

d. At least two years prior to the projected end of operation, a preliminary decommissioning plan shall be submitted to the NRC in accordance with 10 CFR 50.75(f)(4).

e. When permanently ceasing operation, reports shall be submitted to the NRC as required per 10 CFR 50.82.

6.10.14 **10 CFR 50, Appendix G, Fracture Toughness Requirements**

At least three years before the predicted date that fracture toughness levels will no longer satisfy the requirements of 10 CFR 50, Appendix G, Section IV.A.1.c. (see also Step 6.10.7):

a. NAF shall prepare the proposed programs to satisfy the requirements of 10 CFR 50, Appendix G, Sections IV.A.1.a and IV.A.1.b.

b. FSRC and Nuclear Licensing and Operations Support shall review the proposed programs.

c. The Senior Vice President Nuclear shall approve the proposed programs.

d. Nuclear Licensing and Operations Support shall submit the proposed programs to NRC for review and approval. **[10 CFR 50, App. G, Section IV]**

6.10.15 **10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program**

After specimens are withdrawn from capsules:

a. NAF shall coordinate preparation of a summary technical report that includes the data required by ASTM E 185, as specified in 10 CFR 50, App. H, Section III.B.1, and the results of all fracture toughness tests conducted on the surveillance capsule materials in the irradiated and unirradiated conditions.

b. Within one year after surveillance capsule removal, Nuclear Licensing and Operations Support shall submit the report to NRC. If a Technical Specification change is required, whether pressure-temperature limits or operation procedures required to meet the limits, the expected submittal date for the change shall be included with the report. **[10 CFR 50, App. H, Section IV]**

6.10.16 **10 CFR 50, Appendix I, Numerical Guides for Effluent ALARA**

a. If the limits specified in VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry) , are exceeded for:

- The quantity of radioactive material actually released to unrestricted areas during any calendar quarter
- The calculated air dose from radioactive noble gases in gaseous effluents
- The calculated dose from the release of I-131, tritium, and radionuclides in particulate form, with half-lives greater than eight days, in gaseous effluents
- Reporting radioactivity concentrations in environmental samples, when averaged over any calendar quarter

1. Radiological Protection shall prepare a report that:
   - Identifies the causes for exceeding the specified limits
   - Describes corrective actions taken to reduce releases
   - Describes actions taken or proposed to ensure subsequent releases will not exceed the specified limits
2. Nuclear Licensing and Operations Support shall review the report.
3. The Senior Vice President Nuclear Operations shall approve the report.
4. Within 30 days after the release, Nuclear Licensing and Operations Support shall submit the report to NRC. **[10 CFR 50, App. I, Section IV.A.3]**

b. If radioactive liquid or gaseous waste is discharged, without treatment, beyond the limits specified in VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry) :

1. Radiological Protection shall prepare a report that:
   - Explains why radwaste exceeding specified limits was being discharged without treatment
   - Identifies any nonfunctional equipment or sub-system, and the reason for nonfunctionality
   - Describes action taken to restore nonfunctional equipment to functional status
   - Summarizes action to prevent recurrence
2. FSRC and Licensing (Station) shall review the report.
3. The Site Vice President shall approve the report.

4. Within 30 days after the end of the quarter, the Site Vice President shall submit the report to NRC.

6.10.17 **10 CFR 50, Appendix J, Primary Reactor Containment Leakage Testing**

a. A post outage report shall be prepared by Nuclear Engineering presenting the results of the previous cycle's Type B and Type C tests, and Type A, if performed during that outage.

b. The technical contents of the report will be in accordance with NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J", and endorsed by Regulatory Guide 1.163, Performance-Based Containment Leak Test Program dated September 1995.

c. The report shall also show that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable.

d. The reports shall be retained for internal and external review.

6.11 10 CFR 55, Operator's Licenses

6.11.1 **10 CFR 55.25, Incapacitation Because of Disability or Illness**

See Step 6.10.12.

6.11.2 **10 CFR 55.46, Simulator Certification**

a. Performance testing shall be conducted throughout the life of the simulation facility in a manner sufficient to ensure that paragraphs 55.46(c)(2)(ii), as applicable, and 55.46(d)(3) are met. The results of performance tests must be retained for four years after the completion of each performance test or until superseded by updated test results;

b. Correct modeling and hardware discrepancies and discrepancies identified from scenario validation and from performance testing;

c. Make results of any uncorrected performance test failures that may exist at the time of the operating test or requalification program inspection available for NRC review, prior to or concurrent with preparations for each operating test or requalification program inspection **[10 CFR 55.46(d)]**

6.11.3 **10 CFR 55.53(g), Felony Conviction**

See Step 6.10.12. (See also Steps 6.3.4.a.4., 6.8.1, and 6.27.2.a.)

6.12 10 CFR 70, Domestic Licensing of Special Nuclear Material

6.12.1 10 CFR 70.7(e), Form NRC-3

See Step 6.5.1.

6.12.2 10 CFR 70.52, Report of Accidental Criticality

See Step 6.3.3.c. **[10 CFR 70.52(a)]**

6.13 10 CFR 71, Radioactive Material Packaging and Transportation

6.13.1 10 CFR 71.5, Accident Reports

See Steps 6.3.2.g. and 6.21.2. **[10 CFR 71.5(a)(1)(iv)]**

6.13.2 10 CFR 71.17, First Use of NRC Approved Package

If a package, pre-approved by the general license provisions of 10 CFR 71.17, is to be used for the first time by Dominion:

a. Radiological Protection shall prepare a written notification that includes:
   - The name of the licensee
   - The license number
   - The package identification number specified in the NRC package pre-approval

b. If the package is for low-level waste, Nuclear Licensing and Operations Support shall review the notification.

c. If the package is for spent nuclear fuel, NAF and Nuclear Licensing and Operations Support shall review the notification.

d. The Senior Vice President Nuclear shall approve the notification.

e. Before first use of the package, Nuclear Licensing and Operations Support shall submit the notification to the Director, Office of Nuclear Material Safety and Safeguards. **[10 CFR 71.17(c)(3)]**

6.13.3 10 CFR 71.95, Packaging Defects [Commitment 3.2.18]

a. IF there is an instance where there is significant reduction during use in the effectiveness of any authorized packaging for low-level waste, OR

b. There are defects with safety significance in the packaging after first use, OR

c. There are instances which the conditions of approval in the certificate of compliance were not observed in making a shipment, THEN :

1. Radiological Protection shall prepare a report that includes the information specified in 10 CFR 71.95(c)(2).
2. Licensing (Station) shall review the report.
3. The Site Vice President shall approve the notification.
4. Within 60 days, Licensing (Station) shall submit the report to the Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards. **[10 CFR 71.95]**

d. IF there is an instance where there is significant reduction during use in the effectiveness of any authorized packaging for spent nuclear fuel, OR

e. There are defects with safety significance in the packaging after first use, OR

f. There are instances which the conditions of approval in the certificate of compliance were not observed in making a shipment, THEN :

1. Radiological Protection shall prepare a report that includes the information specified in 10 CFR 71.95(c)(2).
2. Licensing (Station) shall review the report.
3. The Site Vice President shall approve the notification.
4. Within 60 days, Licensing (Station) shall submit the report to the Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards. **[10 CFR 71.95]**

6.14 10 CFR 72, Independent Storage of Spent Fuel

6.14.1 10 CFR 72.10(e)(1), Form NRC-3

See Step 6.5.1.

6.14.2 10 CFR 72.44, License Condition Reports

a. Voluntary or involuntary bankruptcy, see Step 6.10.5.g. **[10 CFR 72.44(b)(6)(i)]**

b. Effluent release report—see Step 6.10.3.a. **[10 CFR 72.44(d)(3), SPS ISFSI TS App. C, 1.4.1, & NAPS TS 5.6.3]**

c. Changes to the physical security plan without prior NRC approval—see Step 6.14.12. **[10 CFR 72.44(e) & ISFSI TS App. B, 1.1.1]**

d. Changes to the Emergency Plan without prior NRC approval—see Step 6.10.5.c.

6.14.3 10 CFR 72.48(d)(2), Report of Changes, Tests, and Experiments (ISFSI)

a. Each calendar year, Licensing (Station) shall prepare a report to NRC that contains a brief description of changes, tests, and experiments that did not require NRC approval for which Regulatory Evaluations were prepared including a summary of each Regulatory Evaluation. If none were prepared, a letter shall be submitted stating that no Regulatory Evaluations were required to be reported. **[10 CFR 72.48(d)(2)]**

b. The Director Nuclear Station Safety and Licensing shall approve the report.

c. By March 1 (Surry) and March 31 (North Anna) of the following year, Licensing (Station) shall submit the report to the NRC Region II Office and a copy to Office of Nuclear Material Safety and Safeguards. **[10 CFR 72.48(d)(2)]**

6.14.4 10 CFR 72.48(d)(6)(ii), Notice of Spent Fuel Cask Design Change

A copy of the record for any changes to the design of a spent fuel cask must be provided to the applicable cask certificate holder (cask vendor) within 60 days of implementing the change.

6.14.5 10 CFR 72.70 Safety Analysis Report Updating

a. Every 24 months from the date of the issuance of the license, Nuclear Engineering (ISI/DBD/UFSAR) shall prepare a report to NRC that includes any ISFSI Final Safety Analysis Report pages that have been updated, as delineated in 10 CFR 72.70. If none were updated, a letter shall be submitted stating that no ISFSI FSAR updates were required to be reported.

b. An NBU Officer of the Company, as designated in LI-AA-200, NRC Licensing Correspondence, shall approve the submittal.

c. By June 30 every other year (even numbered years), NL&OS shall submit the report to the NRC.

6.14.6 **10 CFR 72.74, Notifications of Accidental Criticality or Loss of SNM**

a. Accidental criticality—see Step 6.3.3.c.

b. Loss of special nuclear material—see Steps 6.3.3.d. and 6.3.3.e.

6.14.7 **10 CFR 72.75, ISFSI Specific Events and Conditions**

a. ISFSI Immediate, Four-Hour, Eight Hour, and Twenty-Four Hour Notifications shall be made in accordance with Steps 6.3.1.b. and 6.3.1.c. and shall include, if available at time of notification: **[10 CFR 72.75(e)(3)]**

1. The caller's name and call back telephone number
2. A description of the event, including time and date
3. The exact location of the event
4. The quantities, and chemical and physical form of the spent fuel or HLW involved
5. Any personnel radiation exposure data

b. Emergency Notifications (e.g., Declaration of an emergency)—see Steps 6.3.2 and 6.14.7.f. **[10 CFR 72.75(a)]**

c. Non-Emergency Notifications (Four-Hour Notifications) are required for the following ISFSI events or conditions involving spent fuel or High Level Waste (HLW):

1. An action taken in an emergency that departs from a license condition, technical specification, or certificate of compliance when the action is immediately needed to protect the public health and safety and no licensed action that provides adequate or equivalent protection is immediately apparent—see Step 6.3.4 and 6.14.7.f. **[10 CFR 72.75(b)(1)]**

2. 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. **[10 CFR 72.75(b)(2)]**

3. An event that requires unplanned medical treatment of a radioactively contaminated individual at an offsite medical facility—see Steps 6.3.4 and 6.14.7.f. **[10 CFR 72.75(c)(3)]**

d. Non-Emergency Notifications (Eight Hour Notifications) are required for the following ISFSI events involving spent fuel or High Level Waste:

1. A defect in any spent fuel storage structure, system, or component which is important to safety—see Steps 6.3.4 and 6.14.7.f. **[10 CFR 72.75(c)(1)**

2. A significant reduction in the effectiveness of any spent fuel storage confinement system during use—see Steps 6.3.4 and 6.14.7.f. **[10 CFR 72.75(c)(2)]**

e. Non-Emergency Notifications (Twenty-Four Hour Notifications) are required for the following ISFSI events involving spent fuel or High Level Waste:

1. An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive material that could exceed regulatory limits or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases)—see Steps 6.3.4 and 6.14.7.f. **[10 CFR 72.75(d)(1)(i)]**

2. An event in which safety equipment is disabled or fails to function as designed when:—see Step 6.3.6 **[10 CFR 72.75(d)(1)]**

   - The equipment is required to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposure to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident,

   and

   - No redundant equipment was available and operable to perform the required safety function

3. A violation of the functional and operating limits regarding fuel to be stored at the ISFSI in accordance with NAPS ISFSI TS Section 2.2. **[North Anna]**

4. A violation of the functional and operating limits regarding fuel to be stored at the ISFSI in accordance with NUHOMS-HD TS Section 2.2.

f. Sixty-Day Written Reports following ISFSI specific events or conditions in Steps 6.14.7.c., 6.14.7.d., and 6.14.7.e. and above occur must be sent to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy sent to the NRC Region II Office. Reports must include the information specified in 10 CFR 72.75(g).

6.14.8 **10 CFR 72.76, Material Status Report**
See Step 6.16.2.

6.14.9 **10 CFR 72.78, Nuclear Material Transfer Reports**
See Step 6.16.3.

6.14.10 **10 CFR 72.80(a), Other Records and Reports**
See Step 6.7.2.

6.14.11 **10 CFR 72.80(b), Annual Financial Reports**
See Step 6.10.8.

6.14.12 **10 CFR 72.186, Physical Security and Safeguards Contingency Plans**
If a Station physical security plan, training and qualification plan, or safeguards contingency plan is revised without prior NRC approval, as permitted by 10 CFR 72.186(b):

a. Security shall prepare a report that describes the change(s).

b. The Director Nuclear Security and Emergency Preparedness shall review and approve the report.

c. The Vice President Nuclear Operations shall approve the submittal.

d. Within two months after the change is implemented, Nuclear Licensing and Operations Support shall submit the report to the Director, NRC Region II Office/Administrator with a copy to the Office of Nuclear Material Safety and Safeguards. **[10 CFR 72.44(e)] and [10 CFR 72.186(b)]**

6.14.13 **10 CFR 72.212(b)(1), Reports on Use of ISFSI Under a General License**

a. Notification of the NRC prior to first storage of spent fuel under a general license:

1. At least 120 days prior to storage of spent fuel, Supervisor Nuclear Spent Fuel shall notify Supervisor Licensing (Station) of the date planned for first storage of spent fuel under a general license.

2. At least 90 days prior to first storage of spent fuel under this general license, the Supervisor Licensing (Station) shall submit a report, approved by the Site Vice President, notifying the Nuclear Regulatory Commission. **[10 CFR 72.212(b)(1)(i)]**

b. Registration of the use of each cask with the NRC after using that cask to store spent fuel:

1. Within 7 days after using a cask to store spent fuel, Supervisor Nuclear Spent Fuel shall notify Supervisor Licensing (Station) of the date the cask was used to store spent fuel.

2. No later than 30 days after using that cask to store spent fuel the Supervisor Licensing (Station) shall submit a report, approved by the Site Vice President, to register use of each cask with the Nuclear Regulatory Commission.

   [10 CFR 72.212(b)(1)(ii)]

6.15 10 CFR 73, Physical Protection of Plants and Materials

6.15.1 **10 CFR 73.37, Irradiated Fuel Shipment**

a. Before shipment of spent fuel within or through a state, Nuclear Analysis and Fuel shall prepare written notification to the governor or governor's designee. The notification shall include:

- The name, address and telephone number of the shipper, carrier, and receiver
- A description of the shipment as specified by the Department of Transportation in 49 CFR 172.202 and 172.203(d)
- A listing of the routes to be used within the state
- A statement that the information described in Step 6.15.1.b. is required by NRC regulations to be protected in accordance with the requirements of 10 CFR 73.21

b. The following information shall be provided in a separate enclosure to the written notification required by Step 6.15.1.a.:
   - The estimated date and time of departure from the point of origin of the shipment
   - The estimated date and time of entry into the governor's state
   - For a single shipment whose schedule is not related to the schedule of any subsequent shipment, statement that schedule information must be protected in accordance with the provision of 10 CFR 73.21 until at least 10 days after the shipment has entered or originated within the state
   - For a shipment in a series of shipments whose schedules are related, a statement that schedule information must be protected in accordance with the provisions of 10 CFR 73.21 until 10 days after the last shipment in the series has entered or originated within the state and an estimate of the date on which the last shipment in the series will enter or originate within the state

c. The Senior Vice President Nuclear shall approve the notification.

d. At least seven days (determined by postmark) before transport of a shipment within or through a state, NAF shall mail the notification to the governor or governor's designee. **[10 CFR 73.37(f)(1)]**

e. If any schedule changes by more than six hours from that supplied at Step 6.15.1.d., NAF shall notify, by telephone, the office of the governor or the governor's designee. **[10 CFR 73.37(f)(4)]**

6.15.2 **10 CFR 73.67, Shipment of SNM of Low Strategic Significance**

a. Before shipment of SNM of low strategic significance, NAF shall notify the receiver of the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier, and transport identification. **[10 CFR 73.67(g)(1)(i)]**

b. Upon receipt of a shipment of SNM of low strategic significance, NAF shall notify the shipper as specified by Step 6.16.3. **[10 CFR 73.67(g)(2)(ii)]**

6.15.3 **10 CFR 73.71, Safeguard Event Reporting**

a. **Loss of SNM Shipment**

1. See Step 6.3.3.d. **[10 CFR 73.71(a)(1)]**

NOTE: Events that are reportable in accordance with 10 CFR 50.73 need not be duplicated by Step 6.15.3.a.2.

2. Within 60 days after an event that is reportable in accordance with Step 6.3.3.d., a report shall be submitted as an LER attachment, as specified in Step 6.10.11.c. A copy of the LER shall be sent to the NRC Regional Office with a copy to the Director, Division of Nuclear Security and Incident Response. The report shall include sufficient information for NRC analysis and evaluation. **[10 CFR 73.71(a)(4)]**
3. If significant supplemental information becomes available after an initial one-hour notification or LER submittal, see Step 6.3.3.d. Licensing (Station) shall prepare a supplemental LER. **[10 CFR 73.71(a)(5)]**
4. Errors discovered in a report shall be corrected in a revised report with the revisions indicated. Revised reports shall completely replace prior reports. They shall not consist only of revised parts or a supplement. **[10 CFR 73.71(a)(5)]**

b. **Other Safeguards Events**

1. See Step 6.3.3.e. and 6.3.3.h. **[10 CFR 73.71(b)(1)]**

NOTE: Events that are reportable in accordance with 10 CFR 50.73 need not be duplicated by Step 6.15.3.b.2.

2. Within 60 days after an event reportable in accordance with Step 6.3.3.e. and 6.3.3.h., an LER shall be submitted for the event as specified in Step 6.10.11.c. **[10 CFR 73.71(d)]**

6.15.4 **10 CFR 73.72, Irradiated Fuel Shipments**

a. For each shipment of irradiated fuel, NAF shall prepare a written notification that includes **[10 CFR 73.72(a)(3)]**:

- The name, address, and telephone number of the shipper, receiver, and carrier
- The physical form, quantity, type of reactor, and original enrichment
- A listing of the mode of shipment, transfer point, and route to be used
- The estimated time and date that shipment will begin
- The estimated time and date of arrival of the shipment at the destination

b. Nuclear Licensing and Operations Support shall submit the notification to the NRC Director, Division of Nuclear Security and Incident Response in sufficient time that it will be received at least 10 days before transport begins. **[10 CFR 73.72(a)(2)]**

c. At least 2 days before transport begins, Nuclear Licensing and Operations Support shall notify the NRC Headquarters Operations Center. See Step 6.1.1.a. **[10 CFR 73.72(a)(4)]**

d. Nuclear Licensing and Operations Support shall notify the NRC Headquarters Operations Center of any shipment itinerary changes greater than $\pm$ 6 hours. See Step 6.1.1.a.

6.15.5 **10 CFR 73, Appendix G, Safeguards Events**

Events to be reported within one hour; see Step 6.3.3.e.

6.16 10 CFR 74, Material Control and Accounting of Special Nuclear Material

6.16.1 **10 CFR 74.11, Loss or Theft or Attempted Theft of SNM**

See Step 6.3.3.e.

6.16.2 **10 CFR 74.13, Material Status Report**

a. Nuclear Analysis and Fuel (NAF) shall prepare a material status report in computer readable format within 60 calendar days of the beginning of the physical inventory. **[10 CFR 74.13(a)].**

b. The report shall provide information concerning special nuclear material received, produced, possessed, transferred, consumed, disposed of, or lost during the report period.

c. NAF shall also prepare in computer readable format a statement the composition of the ending inventory. The report shall be submitted as an attachment to the material status report.

d. The Director NAF or Director NLOS shall approve the report.

e. The computer readable reports shall be prepared and submitted in accordance with instructions NUREG/BR-0007 and NMMSS Report D-24, Personal Computer Data Input for NRC Licensees.

6.16.3 **10 CFR 74.15, Nuclear Material Transfer**

a. **Shipment of SNM**

For each SNM shipment of 1 gram or more of contained uranium-235, uranium-233, or plutonium:

1. NAF shall prepare in computer readable format, Nuclear Material Transaction Report (Shipper), in accordance with the printed instructions from NUREG/BR-0006 and NMMSS Report D-24.
2. The Director NAF or Director NLOS shall approve the report.
3. Promptly, NAF shall submit the report to the DOE and a copy to the receiver.

b. **Receipt of SNM (Domestic Source)**

For each SNM receipt of 1 gram or more of contained uranium-235, uranium-233, or plutonium, received:

1. NAF shall prepare in computer readable format, Nuclear Material Transaction Report (Receiver), in accordance with the printed instructions from NUREG/BR-0006 and NMMSS Report D-24.
2. The Director NAF or Director NLOS, shall approve the report.
3. Within 10 days after receipt, unload, and verification, NAF shall submit the report in accordance with the printed instructions.

6.17 10 CFR 140, Financial Protection and Indemnity Agreements

6.17.1 **10 CFR 140.6, Reports**

In the event of bodily injury or property damage arising out of or associated with possession or use of radioactive material at the Station or transportation of radioactive material, or in the event such a claim is made:

a. Corporate Risk Management shall prepare written notice to identify, as reasonably obtainable, the time, place, and circumstances or nature of the event.

b. Nuclear Licensing and Operations Support shall review the notice.

c. The Senior Vice President Nuclear Operations shall approve the notice.

d. As promptly as practical, Nuclear Licensing and Operations Support shall submit the notice to the Director, Nuclear Reactor Regulation, or Director, Nuclear Material Safety and Safeguards. **[10 CFR 140.6(a)]**

6.17.2 **10 CFR 140.15, Proof of Financial Protection**

In the event of a material change in the proof of financial protection or other financial information filed with NRC to comply with 10 CFR 140.15:

a. Corporate Risk Management shall prepare a written notice to describe the change.

b. Promptly, Nuclear Licensing and Operations Support shall submit the notice to the Director, Nuclear Material Safety and Safeguards. **[10 CFR 140.15(e)]**

6.17.3 **10 CFR 140.17, Policy Renewal or Replacement**

If liability insurance policies obtained to provide all or part of the financial protection required by 10 CFR 140 are to expire, require renewal, or be replaced by another form of protection:

a. Corporate Risk Management shall prepare written notice that indicates renewal of the policy or shall obtain other proof of financial protection as required by Step 6.17.2.

b. Nuclear Licensing and Operations Support shall review the notice.

c. The Senior Vice President Nuclear Operations shall approve the notice.

d. At least 30 days before policy termination, Nuclear Licensing and Operations Support shall submit the notice to the Director, Nuclear Reactor Regulation, or Director, Nuclear Material Safety and Safeguards. **[10 CFR 140.17(b)]**

6.17.4 **10 CFR 140.21, Guarantees of Deferred Premium Payment**

Annually, for each licensed reactor:

a. Corporate Accounting shall prepare a report that establishes that Dominion maintains one of the following types of guarantee of payment of deferred premiums in an amount of $15 million:
   - Surety bond
   - Letter of credit
   - Revolving credit and term loan arrangements
   - Maintenance of escrow deposits of government securities
   - Annual, certified financial statement showing that either that a cash flow (i.e., cash available to a company after all operating expenses, taxes, interest charges, and dividends have been paid) can be generated and would be available for payment of retrospective premiums within three months after submission of the statement, or a cash reserve or a combination of cash flow and cash reserve
   - Another type of guarantee approved by NRC

b. Nuclear Licensing and Operations Support shall review the report.

c. The Director NL&OS shall approve the report.

d. Before April 1 of each year, Nuclear Licensing and Operations Support shall submit the report to the Director, Nuclear Reactor Regulation, or Director, Nuclear Material Safety and Safeguards.

6.18 18 CFR 12, Water Power Project Safety (Lake Anna Dam)

6.18.1 18 CFR 12.10(a), Conditions Affecting the Safety of a Project

NOTE: "Condition that affects the safety of" is defined at Subsection 4.8 (the Main Dam Daily Inspection form identifies a number of such conditions).

If a condition is identified that affects the safety of the dam or associated features, but does not require entry to the Emergency Action Plan for Lake Anna Dam:

a. See Step 6.3.2.i.

b. Licensing (Station) shall prepare a report that contains any information the FERC Regional Engineer directs, including:
   - The causes of the condition
   - A description of any unusual occurrences or operating or operating circumstances preceding the condition
   - An account of any measure taken to prevent worsening of the condition
   - A detailed description of any damage to project works and the status of any repair
   - A detailed description of any personal injuries
   - A detailed description of the nature and extent of any private property damages
   - Any other relevant information requested by the FERC Regional Engineer

c. A description shall be enclosed (with the report) of any modification that is an emergency measure taken in response to a condition affecting the safety of the dam or its works. This description is required even if the modification does not otherwise require specific, prior, FERC approval. See also Step 6.18.8. **[18 CFR 12.11(a) & (b)]**

d. The Site Vice President shall review and approve the report.

e. Within the time specified by the FERC Regional Engineer, Licensing (Station) shall submit the report to the FERC Regional Office.

6.18.2 **18 CFR 12.10(b), Deaths or Serious Injuries At or Near the Dam**

If a serious accident or a death occurs at or immediately above or below Lake Anna Dam[1], or is alleged to be related to the existence or operation of the dam: **[18 CFR 12.10(b)]**

a. See Step 6.3.2.h.

b. Licensing (Station) shall prepare Attachment 2, FERC Public Safety Database Report, and a transmittal letter. Additional information, such as newspaper articles, maps, and law enforcement agency reports should be attached, and may be required. Any action to prevent recurrence shall be described in the transmittal letter.

c. The Manager Nuclear Operations shall review the letter and report.

d. The Site Vice President shall approve the letter and report, including notary verification as specified by 18 CFR 12.13, Verification form.

e. Within 72 hours after discovery, Licensing (Station) should submit the report to the FERC Regional Office. **[Commitments 3.2.13 and 3.2.15]**

6.18.3 **18 CFR 12.11, Modifications to the Dam or Its Works**

Except as noted at Step 6.18.1.c., if modifications to the dam or its works are planned:

a. Nuclear Engineering shall notify Electric Environmental Services and Licensing (Station) as specified in VPAP-0301, Design Change Process.

b. Nuclear Engineering shall prepare a report that describes the modifications.

c. The Site Vice President shall review and approve the report.

d. At least 60 days before work on the modification begins, Licensing (Station) shall submit the report to the FERC Regional Office.**[18 CFR 12.11(a) & (c)]**

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1. Incidents which involve other parts of the lake are excluded. **[18 CFR 12.10(b)(4)]**

6.18.4 **18 CFR 12.24, Review and Updating of Emergency Action Plans**

a. If Dominion discovers that a significant change has occurred in upstream or downstream circumstances that might affect water flows or the location or extent of the areas, persons, or property that might be harmed in an emergency that involves the dam:

1. The Director Nuclear Protection Services and Emergency Preparedness shall prepare a notification of the change in circumstances.
2. The Site Vice President shall approve the notification.
3. Licensing (Station) shall submit the notification to the FERC Regional Engineer and the State Water Control Board.

b. When a new independent consultant report is issued, as required by Step 6.18.7, Nuclear Emergency Preparedness shall review the report and update, as necessary, applicable sections of the North Anna Hydroelectric Project Emergency Action Plan by replacing the previous report in the Plan[1], and implementing any corresponding changes to the Plan.

c. Nuclear Emergency Preparedness shall prepare a letter[2] to submit the annual North Anna Hydroelectric Project Emergency Action Plan adequacy review. If the Plan has been revised, submit three copies of the revised Plan to the FERC.

1. The Site Vice President shall approve the letter.
2. By December 31 each year, Licensing (Station) shall submit the letter to the FERC Regional Engineer. **[NAHP EAP, App. B.3]**

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1. Independent Consultant reports are an integral part of the North Anna Hydroelectric Project Emergency Action Plan. When a new report is issued, applicable sections of the plan should be updated, as necessary, to reflect updated information.
2. The same letter may also submit the exercise summary and critique required by Step 6.18.5.

6.18.5 **18 CFR 12.25, Posting and Readiness**

a. The Station Coordinator Emergency Preparedness shall ensure that a copy of the current North Anna Hydroelectric Project Emergency Action Plan is posted at the spillway office, the hydroelectric building, and in the North Anna Control Room.

b. Nuclear Emergency Preparedness shall prepare a letter to submit the annual North Anna Hydroelectric Project Emergency Action Plan test exercise summary and critique.

c. The Site Vice President shall approve the letter.

d. Within 30 days after performing a test exercise, but no later than December 31 of each year, Licensing (Station) shall submit the letter to the FERC Regional Engineer. **[NAHP EAP V.E and App. B.2]**

6.18.6 **18 CFR 12.36, Emergency Corrective Measures**

If, during an inspection, an independent consultant discovers any condition for which emergency corrective measures are advisable:

a. The independent consultant shall notify Nuclear Engineering.

b. Nuclear Engineering shall notify the Shift Manager. See Step 6.3.2.i.

6.18.7 **18 CFR 12.37-39, Independent Consultant Reports**

Nuclear Engineering shall obtain an independent consultant report for each five-year inspection required by 18 CFR 12.37(c). The contract for the independent consultant shall include the notification requirement of Step 6.18.6.a. Nuclear Engineering shall send one copy of the report to Nuclear Emergency Preparedness and three copies to Licensing (Station). Licensing (Station) shall submit three copies of the Independent Consultant Report to the FERC Regional Engineer as required by 18 CFR 12.37(a). See Step 6.18.4.b.

6.18.8 **18 CFR 12.42, Warning and Safety Devices**

Except as noted at Step 6.18.1.c., if it is planned to remove a safety device from service:

a. The Shift Manager shall notify Licensing (Station).

b. Licensing (Station) shall prepare a notification letter to the FERC Regional Engineer that includes:
   - The reason for removing the device from service
   - The proposed date of removal
   - The schedule for return to service
   - Any mitigating actions to be taken

c. The Site Vice President shall review and approve the letter.

d. At least 10 days before the device is removed from service, Licensing (Station) shall submit the letter to the FERC Regional Engineer. **[Commitment 3.2.14]**

e. When the device has been returned to service, the Shift Manager shall notify Licensing (Station).

f. Licensing (Station) shall prepare a notification letter to the FERC Regional Engineer.

g. The Site Vice President shall approve and verify the statement, as specified by 18 CFR 12.13, Verification form.

h. Within 10 days after the device is returned to service, Licensing (Station) shall submit the letter to the FERC Regional Engineer. **[Commitment 3.2.14]**

6.18.9 **18 CFR 12.44(b), Spillway Gate Test**

If each spillway gate is not operated on a test basis during the periodic FERC inspection:

a. Licensing (Station) shall prepare a statement that each spillway has been operated at least once during the twelve months preceding the inspection.

b. The Site Vice President shall approve and verify the statement, as specified by 18 CFR 12.13, Verification form.

c. By December 31, Licensing (Station) shall submit the statement to the FERC Regional Engineer.

6.18.10 **18 CFR 12.44(c), Emergency Diesel Load Test**

a. Licensing (Station) shall prepare a statement that describes the intervals at which the emergency diesel generator was load tested during the year preceding the annual FERC inspection.

b. The Site Vice President shall approve and verify the statement, as specified by 18 CFR 12.13, Verification form.

c. By December 31, Licensing (Station) shall submit the statement to the FERC Regional Engineer.

6.19 29 CFR 1900, Occupational Safety and Health Administration

6.19.1 **29 CFR 1904, Reporting Occupational Injuries and Illnesses**

a. **29 CFR 1904.32, Annual Summary Posting**

1. Each year, Nuclear Site Safety (Station) shall prepare an OSHA form No. 300 - Log of Work-Related Injuries and Illnesses, that contains an annual summary of occupational injuries and illnesses for the Station during the previous calendar year. The form shall include:
    - The calendar year covered by the report
    - “Dominion,” address, and Station name
    - Supervisor Nuclear Site Safety (Station)

2. Nuclear Site Safety (Station) shall post the form from February 1 to April 30 on the turbine building entry bulletin board **(North Anna)** “Safety” bulletin board in the service building hallway **(Surry)**.

b. **29 CFR 1904.39, Reporting Fatalities or Multiple Hospitalization Accidents**
See Step 6.3.5.c.

6.19.2 **29 CFR 1910.120, Hazardous Waste Operations and Emergency Response**

NOTE: Lists of hazardous materials and associated reportable quantities are updated frequently. Consult with the Environmental Compliance Coordinator if you have any doubt about the classification of a material or its reportability threshold.

See Step 6.3.2.e.

6.20 40 CFR, Protection of Environment

6.20.1 40 CFR 61, National Emissions Standards for Hazardous Air Pollutants - Asbestos

See Step 6.27.3.b.

6.20.2 40 CFR 82, Protection of Stratospheric Ozone

a. For commercial refrigeration units that contain a charge of more than 50 pounds of refrigerant, leaks greater than 35% per year must be repaired within thirty days of the date of discovery, or within 30-days of a failed follow-up verification test, except when a retrofit/retirement plan is developed within thirty days and actions under that plan are completed within one year from the plan's date.

b. For comfort-cooling equipment with a charge of more than 50 pounds, leaks greater than 15% per year must also be repaired within thirty days of the date of discovery, or within 30-days of a failed follow-up verification test, except when a retrofit/retirement plan is developed within thirty days and actions under that plan are completed within one year from the plan's date.

c. Additional time (beyond the 30-day time period) is allowed to conduct leak repairs if the necessary repair parts are unavailable or if other applicable federal, state, or local regulations make a repair within 30 impossible. EPA must be notified per 40 CFR 82.166(n).

d. If repairs cannot be completed within 30-days of discovery or within 30-days of a failed follow-up verification test, the EPA must be notified in accordance with 40 CFR 82.166(n). The report shall be sent to the address listed in 40 CFR 82.160.

6.20.3 40 CFR 110, Discharge of Oil

See Step 6.3.2.d.

6.20.4 **40 CFR 112, Oil Pollution Prevention**

If more than 1000 gallons of oil are released into or upon navigable waters or adjoining shorelines in a single event, or if there are two release events to navigable waters within any 12 month period, Electric Environmental Services shall submit a package to the EPA Regional Administrator that includes:

- The name of the facility
- Name of the owner or operator
- Location of the facility
- Date and year of initial facility operation
- The maximum storage capacity of the facility
- A description of the facility, including maps, flow diagrams, and topographical maps
- A complete copy of the Plan and any amendments
- The cause of the release, including a failure analysis of system or subsystem in which the failure occurred
- Corrective actions or countermeasures taken, including and adequate description of equipment repairs or replacements
- Additional preventive measures taken or contemplated to minimize the possibility of recurrence
- Any other information the Regional Administrator may request

6.20.5 **40 CFR 117, Hazardous Substance Reportable Quantities**

See Step 6.3.2.e.

6.20.6 **40 CFR 190, Environmental Radiation Protection Standards**

See Step 6.6.4.a.4.

6.20.7 **40 CFR 262, Standards Applicable to Generators of Hazardous Waste**

a. **Biennial Report**

1. Electric Environmental Services shall prepare EPA Form 8700-13A to cover the previous year, including:
   - The EPA identification number, name, and address of the generator
   - The calendar year covered by the report
   - The EPA identification number, name, and address for each off-site treatment, storage, or disposal facility to which the waste was shipped during the year
   - The name and EPA identification number of each transporter used during the reporting year
   - A description, EPA hazardous waste number (from 40 CFR 261, Subpart C or D), DOT hazard class, and quantity of each hazardous waste shipped off site, listed by EPA identification number of each such off-site facility to which a waste was shipped
   - A description of the efforts undertaken during the year to reduce the volume and toxicity of waste generated
   - A description of the changes in volume and toxicity of waste actually achieved during the year in comparison to previous years to the extent such information is available for years before 1984
2. By March 1 of each even numbered year, Electric Environmental Services shall submit the report to the EPA Regional Administrator. **[40 CFR 262.41]**

b. **Exception Reporting**

1. Hazardous waste shipments require a shipment form (see 40 CFR 262). If a copy of the form, signed by the disposal site operator, is not returned to the shipper within 35 days, the Environmental Compliance Coordinator shall notify Electric Environmental Services.
2. If the signed form still has not been returned, by the 45th day after shipment, Electric Environmental Services shall prepare and submit an Exception Report to the EPA Regional Administrator that includes **[40 CFR 262.42]**:
   - A legible copy of the unacknowledged manifest
   - A cover letter that explains efforts to locate the hazardous waste and the results of those efforts

6.20.8 **40 CFR 302, Reportable Quantities and Notification**

See Step 6.3.2.e.

6.20.9 **40 CFR 355.30, Extremely Hazardous Substances**

If the Station ever has an extremely hazardous substance in an amount greater than its threshold planning quantity, as established by Appendix A or B of 40 CFR 355.30, the Station will be subject to the requirements of 40 CFR 355. If this occurs, the Environmental Compliance Coordinator shall notify the Local Emergency Planning Coordinator within 60 days and shall initiate appropriate change requests to this procedure to implement the corresponding reporting requirements.

6.20.10 **40 CFR 370, Hazardous Chemical Reporting: Community Right-to-Know**

a. **Material Safety Data Sheet (MSDS) Reporting**

If significant new information is discovered concerning a hazardous substance for which an MSDS was submitted previously or a new substance exceeds the minimum threshold level for reporting as established by 40 CFR 370.20, within three months, the Environmental Compliance Coordinator shall submit a new or revised MSDS for that substance to the State Department of Environmental Quality (Waste), the Local Emergency Planning Coordinator, and all applicable fire departments. **[40 CFR 370.21(c)]**

b. **Inventory Reporting**

1. Each year, the Environmental Compliance Coordinator shall prepare a Tier I information form (40 CFR 370.41) on hazardous chemicals present at the Station above the threshold levels established in 40 CFR 370.20(b).

2. By March 1 of each year, the Environmental Compliance Coordinator shall submit the form for the previous year to the State Department of Environmental Quality (Waste), the Local Emergency Planning Coordinator, and applicable fire departments. **[40 CFR 370.25(a)]**

6.20.11 **40 CFR 761, Polychlorinated Biphenyls (PCBs)**

See Step 6.3.2.e.

6.21 49 CFR 171, Hazardous Materials Transport Incidents

6.21.1 If an incident occurs during transport (including loading, unloading, and temporary storage) of hazardous materials, see Step 6.3.2.g. **[49 CFR 171.15]**

6.21.2 If:

- The Department of Transportation (DOT) is notified as specified by Step 6.3.2.g.
- There is an unintentional release of hazardous materials from a package (including a tank), or
- Any quantity of hazardous waste has been discharged during transportation:

a. The Environmental Compliance Coordinator shall prepare a report on DOT Form F 5800.1.[1]

b. Nuclear Licensing and Operations Support shall review the report.

c. The Senior Vice President Nuclear Operations shall approve the report.

d. Within 15 days, Nuclear Licensing and Operations Support shall submit the report to DOT. **[49 CFR 171.16]**

6.21.3 If a hazardous substance is discharged (accidentally or intentionally) in a reportable quantity from one package, or transport vehicle if not packaged, into or upon the navigable waters or adjoining shorelines, see Step 6.3.2.e.

6.22 Federal Environmental & Wildlife Protection Acts

6.22.1 **Clean Air Act**

See Step 6.22.3.

6.22.2 **Clean Water Act**

See Step 6.22.3.

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1. If the report pertains to a hazardous waste discharge, a copy of the hazardous waste manifest for the waste shall be attached to the report and an estimate of the quantity of the waste removed from the scene, the name and address of the facility to which it was taken, and the manner of disposition of any unremoved waste must be entered in Part H of the form. **[49 CFR 171.16(a)]**

6.22.3 **Comprehensive Environmental Responsibility, Compensation and Liability Act[1] (CERCLA)**

NOTE: CERCLA notification requirements do not apply to releases that are entirely on-site.

a. See Step 6.3.2.e.

b. If notifications are made as specified in Step 6.3.2.e.:

1. Electric Environmental Services shall prepare a report (as notices or additional information become available) that updates information provided in the immediate notification and that includes additional information with respect to:
   - Actions taken to respond to and to contain the release
   - Any known or anticipated acute or chronic health risks associated with the release
   - Where appropriate, advice regarding medical attention necessary for exposed individuals
2. The Site Vice President shall review the report.
3. The Director Electric Environmental Services shall approve the report.
4. As soon as practicable, Electric Environmental Services shall submit the report to all agencies notified in accordance with Step 6.3.2.e., except the National Response Center. **[CERCLA Sec. 304(c)]**

6.22.4 **Migratory Bird Treaty Act**

If an osprey nest is disturbed or a raptor (bird of prey) is injured or killed by electrocution, notify the Environmental Compliance Coordinator. The Environmental Compliance Coordinator shall collect the information necessary to complete a Raptor (Bird of Prey) Incident Report (form no. 721841), and notify Electric Environmental Services at one of the telephone numbers provided on the form.

6.22.5 **Superfund Amendments and Reauthorization Act of 1986[2] (SARA)**

Includes Emergency Planning and Community Right-To-Know Act of 1986[3]. See Steps 6.3.2.e., 6.20.7, and 6.20.10.

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1. 42 U.S.C. §§9601-57 (1982)
2. Pub. L. 99-499, 100 Stat. 1613 (1986)
3. Pub. L. 99-499.

6.23 North Anna Units 1 & 2 License and Technical Specifications/Topical Report

NOTE: North Anna Independent Spent Fuel Storage Installation License and Technical Specifications reporting requirements are included with the requirements for 10 CFR 50.36a at Step 6.10.3 and 10 CFR 72 at Subsection 6.14.

6.23.1 **Special Report Review**

FSRC shall review all special reports submitted to NRC.

6.23.2 **Reportable Events**

a. For all NRC-reportable events, NRC shall be notified as specified in Step 6.10.10 and an LER submitted as specified in Step 6.10.11.

b. FSRC shall review each event reported by an LER and send the results to the Senior Vice President Nuclear Operations and the MSRC. **[Topical Report]**

6.23.3 **Safety Limit Violations**

If a technical specification safety limit is violated, see Steps 6.10.11 and 6.3.2.a.5.

6.23.4 **Inservice Inspection Reports**

After each refueling outage:

a. Nuclear Engineering shall prepare Owner's Report for Repairs or Replacements, Form NIS-2 (ASME Section XI) in accordance with the Dominion Inservice Inspection Manual.

b. Nuclear Engineering shall prepare Owner's Activity Report, Form OAR-1 (ASME Section XI) in accordance with the Dominion Inservice Inspection Manual.

c. The Site Vice President, Director NSS&L, or Director NLOS shall approve the reports.

d. Licensing (Station or Corporate) shall submit the OAR-1 to the NRC. **[ASME IWA-6000]**

6.23.5 **Changes Related to Radioactivity in Effluents**

If it is planned to remove or change significantly the normal operation of equipment that controls the amount of radioactivity in effluents (regardless of whether the change affects the amount of radioactivity in the effluents):

a. Licensing (Station) shall prepare a report that describes the planned change.

b. Nuclear Licensing and Operations Support shall review the report.

c. The Senior Vice President Nuclear Operations shall approve the report.

d. Before the change is implemented, Nuclear Licensing and Operations Support shall submit the report to NRC. **[Unit 1 License, 2.C.(3)(b); Unit 2 License, 2.C.(3)(a)]**

6.23.6 **Violations of Requirements of the License**

a. See Step 6.3.6.d.

b. If NRC is notified as specified in Step 6.3.6.d.:

1. Licensing (Station) shall prepare a follow-up report.
2. The Senior Vice President Nuclear Operations shall approve the report.
3. Within 14 days after discovery of the violation, Nuclear Licensing and Operations Support shall submit the report to the NRC Regional Office. **[Unit 2 License, 2.C(3)(a)]**

6.23.7 **Steam Generator Tube Inspection Reports**

a. **Steam Generator Tube ISI**

If performance criterion is exceeded and results require prompt notification to the Commission pursuant to Section 50.72 to 10 CFR Part 50, an LER shall be submitted pursuant to Section 50.73 to 10 CFR Part 50. The LER should include a root cause evaluation identifying the performance criteria exceeded and an operational assessment establishing the bases for the next operating cycle.

b. **Tube Inspection Report** **[TS 5.6.7]**

Engineering Programs - ISI/IST/Materials (Station) shall prepare a report for submittal by Licensing (Station) to the NRC within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall be approved by the Site Vice President and shall include:

1. The scope of inspections performed on each SG.
2. Active degradation mechanisms found.
3. Nondestructive examination techniques utilized for each degradation mechanism.

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications.
5. Number of tubes plugged during the inspection outage for each active degradation mechanism.
6. Total number and percentage of tubes plugged to date.
7. The results of condition monitoring, including the results of tubes pulled and the in-situ testing.
8. The effective plugging percentage for all plugging in each SG.

6.23.8 **Core Operating Limits Report**

a. NAF shall prepare a report for each refueling that provides the information specified in Technical Specification 5.6.5 (Surry TS 6.2.C.)

NOTE: Any information needed to support N(Z) and/or the Axial Flux Difference limits will be by NRC request and need not be included in this report.

b. FSRC shall review the report.
c. The Director NL&OS or Director NSS&L shall approve the report.
d. When issued, Nuclear Licensing and Operations Support shall submit the report to NRC and copies to the NRC Regional Office and the NRC Resident Inspector. **[TS 5.6.5]**

6.23.9 **Annual Radiological Environmental Operating Report**

a. Radiological Protection shall prepare a draft Radiological Environmental Operating Report[1] for each calendar year. The report shall be consistent with the objectives outlined in VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry) , and 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

1. A single submittal is acceptable for both units.

b. The report shall provi3de summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program during the year, including:
   - A summary description of the Radiological Environmental Monitoring Program
   - Analysis results of radiological environmental samples and of all environmental radiation measurements, taken during the period pursuant to the locations specified in the tables and figures of the ODCM as well as summarized and tabulated results of these analyses and measurements commensurate with the format in the ODCM. If some results are not available for inclusion with the report, the missing information shall be identified, and their unavailability explained. Missing information shall be submitted in a supplemental report as soon as possible **[TS 5.6.2]**
   - Comparisons (as appropriate) with preoperational studies, operational controls, and previous environmental surveillance reports
   - At least two legible maps that include sampling locations, keyed to a table that gives distances and directions from the centerline of one reactor. One map shall include locations near the site boundary; the second shall include more distant locations
   - Land use census results
   - Descriptions of radionuclide levels—not due to plant effluents—that would otherwise have required a special report as specified in Step 6.10.16
   - A discussion of deviations from the sampling schedule as specified by VPAP-2103N
   - A discussion of analyses in which the lower limit of detection (LLD) as specified by VPAP-2103N was not achievable
   - If Interlaboratory Comparison Program analyses were not performed as required by VPAP-2103N, a description of corrective actions to prevent recurrence
   - An assessment of the observed impacts of Station operation on the environment

c. If the Radiological Environmental Monitoring Program was not conducted as specified in VPAP-2103N, the report shall state the reasons, describe actual or planned corrective action, and actions to prevent a recurrence.

d. Radiological Protection shall forward the draft to Licensing (Station). Licensing (Station) shall complete preparation of the report with assistance from others as required.

e. The Manager Radiological Protection and Licensing (Station) shall review the report.

f. The Director Nuclear Station Safety and Licensing shall approve the report.

g. Before May 1 of the following year, Licensing (Station) shall submit the report to the NRC. [TS 5.6.2]

6.23.10 **Annual Radiological Effluent Release Report**

See Step 6.10.3. [TS 5.6.3]

6.23.11 **Special Reports**

a. Licensing (Station) shall prepare the special reports within the time period specified for each report pursuant to the requirements of the applicable specification.

b. FSRC shall review the reports.

c. The Site Vice President shall approve the reports.

d. On the schedule specified in the Technical Specifications, Site Vice President shall submit the reports to NRC.

6.23.12 **Offsite Dose Calculation Manual (ODCM)**

If VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry), is revised, a complete, legible copy shall be submitted with the Radiological Effluent Release Report for the year in which the revision is implemented. See Step 6.10.3. [TS 5.6.3]

6.23.13 **Post Accident Monitoring (PAM) Report**

a. When required by Condition B of LCO 3.3.3, Licensing (Station) shall prepare PAM Report. [TS 5.6.6]

b. The report shall outline the cause of inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

c. FSRC shall review the report.

d. The Site Vice President shall approve the report.

e. The report shall be submitted to the NRC within the following 14 days when the report is required by Condition B of LCO 3.3.3.

6.23.14 **Reactor Pressure Vessel (RPV) Head Related Inspection Results**

a. RPV head and penetration inspections, on a time period related to effective degradation years, are required by Revised NRC Order EA-03-009 Paragraph IV.C. For each of these inspections, a report detailing the inspection results is required to the NRC.

b. For any boron deposits discovered on the surface of the RPV head or related insulation, regardless of the source of the deposit, Revised NRC Order EA-03-009 Paragraph IV.D requires inspections of the affected RPV head surfaces and penetrations. If a leak or boron deposit is found a report detailing the results to the NRC is required.

c. For the above reports to the NRC, within 45 days after the plant is returned to operations the inspection results will be provided to the Supervisor Licensing (Station) by the site Boric Acid Corrosion Control (BACC) Coordinator and by the Supervisor ISI/IST/Materials Engineering (Station) for nonvisual NDE inspections.

d. For the above reports to the NRC, within 60 days after the plant is returned to operations, the Supervisor Licensing (Station) shall submit a report, approved by the Site Vice President, detailing the inspection results to the NRC, as required by Revised NRC Order EA-03-009 Paragraph IV.E.

6.23.15 **Technical Specification (TS) Bases Changes**

a. Changes to the TS Bases implemented without prior NRC approval, including TS Bases revisions transmitted to the NRC for information as part of a TS change request, shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). **[TS 5.5.13.d]**

b. Licensing (Station) shall prepare the transmittal.

c. The Director NSS&L shall approve the transmittal.

d. Consistent with Step 6.10.9.c., by October 15 of every year, Licensing (Station) shall transmit the TS Bases changes to the NRC.

6.24 Surry License and Technical Specifications

NOTE: Surry Independent Spent Fuel Storage Installation License and Technical Specifications reporting requirements are included with the requirements for 10 CFR 50.36a at Step 6.10.3 and 10 CFR 72 at Subsection 6.14.

6.24.1 Special Report Review

See Step 6.23.1.

6.24.2 Reportable Events

a. See Step 6.23.2.a. **[TS 6.2.A.1]**

b. See Step 6.23.2.b.

c. See Step 6.10.10. **[TS 6.2.B]**

6.24.3 Safety Limit Violations

See Step 6.23.3.

6.24.4 Startup Report

a. NAF shall prepare a summary report of plant startup and power escalation testing after:
   - Receiving a license amendment that involves a planned power level increase
   - Installing a different fuel design
   - Installing fuel from a different manufacturer
   - Making plant modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant

b. The report shall describe:
   - Each of the tests identified in the UFSAR
   - The measured values of the operating conditions or characteristics obtained during the test program and compare these values with design predictions and specifications
   - Any corrective actions required to obtain satisfactory operation
   - Any additional specific details requested in license conditions based on other commitments

c. FSRC shall approve the report.

d. Nuclear Licensing and Operations Support shall submit the report to NRC **[TS 6.6.A.1]** and copies to the MSRC and the NRC Regional Office **[TS 6.6.A.1]** within 90 days after the earlier of:
  - Startup test program completion
  - Resumption of commercial power operation

6.24.5 **Inservice Inspection Reports**

See Step 6.23.4. **[ASME IWA-6000]**

Required reports shall include the information required by the 1998 Edition, with 2000 Addenda of the ASME Section XI, Appendix IV, Article IWV-7000.

6.24.6 **Annual Reports**

NOTE: This report can be submitted in conjunction with the Report of Individual Monitoring—see Step 6.6.7.

Each year Radiological Protection shall provide a report input for the previous calendar year that includes the results of specific activity analyses in which the primary coolant exceeded the limits of Technical Specification 3.1.D.4. The input shall include the information specified in 3.1.D.4. **[TS 6.6.A.2.b]**

a. Radiological Protection shall forward the input to Licensing (Station).

b. Licensing (Station) shall complete preparation of the report with assistance from others as necessary.

c. Licensing (Station) shall review the report.

d. The Director Nuclear Station Safety and Licensing shall approve the report.

e. By April 30 each year, Licensing (Station) shall submit the report for the previous year to the NRC Regional Office.

6.24.7 **Core Operating Limits Report**

See Step 6.23.8. **[TS 6.2.C.]**

6.24.8 **Annual Radiological Environmental Operating Report**

a. Radiological Protection shall prepare a draft Radiological Environmental Operating Report[1] for each calendar year. The report shall be consistent with the objectives outlined in VPAP-2103N, Offsite Dose Calculation Manual (North Anna), VPAP-2103S, Offsite Dose Calculation Manual (Surry) , and 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

b. The report shall provide summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program during the year, including:

- A summary description of the Radiological Environmental Monitoring Program
- Analysis results of radiological environmental samples and of environmental radiation measurements, summarized and tabulated in the format of the table in the Radiological Assessment Branch Technical Position: An Acceptable Radiological Environmental Monitoring Program. If some results are not available, the missing information shall be identified, and their unavailability explained. Missing information shall be submitted in a supplemental report as soon as practicable
- Comparisons (as appropriate) with preoperational studies, operational controls, and previous environmental surveillance reports
- At least two legible maps that include sampling locations, keyed to a table that gives distances and directions from the centerline of one reactor. One map shall include locations near the site boundary; the second shall include more distant locations
- Land use census results
- Descriptions of radionuclide levels—not due to plant effluents—that would otherwise have required a special report as specified in Step 6.10.16
- A discussion of deviations from the sampling schedule as specified by VPAP-2103
- A discussion of analyses in which the lower limit of detection (LLD) as specified by VPAP-2103 was not achievable
- If Interlaboratory Comparison Program analyses were not performed as required by VPAP-2103, a description of corrective actions to prevent recurrence
- An assessment of the observed impacts of Station operation on the environment

1. A single submittal is acceptable for both units.

c. If the Radiological Environmental Monitoring Program was not conducted as specified in VPAP-2103, the report shall state the reasons, describe actual or planned corrective action, and actions to prevent a recurrence.

d. Radiological Protection shall forward the draft to Licensing (Station). Licensing (Station) shall complete preparation of the report with assistance from others as required.

e. The Manager Radiological Protection and Licensing (Station) shall review the report.

f. The Director Nuclear Station Safety and Licensing shall approve the report.

g. Before May 1 of the following year, Licensing (Station) shall submit the report to the NRC. **[TS 6.6.B.2]**

6.24.9 **Annual Radiological Effluent Release Report**

See Step 6.10.3. **[TS 6.6.B.3]**

6.24.10 **Offsite Dose Calculation Manual (ODCM)**

See Step 6.23.12. **[TS 6.8.B.3]**

6.24.11 **Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems**

See Step 6.10.3.

6.24.12 **Containment Leak Rate Test**

See Step 6.10.17.

6.24.13 **Special Reports**

a. If the Reactor Vessel Overpressure Mitigating System is used to mitigate an RCS pressure transient:

1. Licensing (Station) shall prepare a special report that describes the circumstances initiating the transient, the effect of the PORVs or the administrative controls on the transient, and any corrective action necessary to prevent recurrence.

2. FSRC shall review the report.

3. The Site Vice President shall approve the report.

4. Within 30 days, Licensing (Station) shall submit the report to the NRC Regional Office. [TS 6.6.C]

b. If less than the minimum number of explosive gas monitoring instrument channels are operable for 30 days:

1. Licensing (Station) shall prepare a special report that explains why the inoperability was not corrected in a timely manner.
2. FSRC shall review the report.
3. The Site Vice President shall approve the report.
4. Within 30 days, Licensing (Station) shall submit the report to the NRC Regional Office. [TS 3.7.D.2]

c. If the concentration of oxygen in the waste gas holdup system is greater than two percent by volume, and is not restored to less than or equal to two percent by volume within 48 hours:

1. Licensing (Station) shall prepare a special report that describes the cause for the waste gas decay tank exceeding two percent limit, the reason the oxygen concentration could not be restored within limits, and the actions taken and time required to restore the oxygen concentration to within limits.
2. FSRC shall review the report.
3. The Site Vice President shall approve the report.
4. Within 30 days, Licensing (Station) shall submit the report to the NRC Regional Office. [TS 3.11.C]

d. Except for physics and rod exercise testing, if quadrant to average power tilt exceeds two percent for 24 hours and the design hot channel factors for rated power are not exceeded:

1. Licensing (Station) shall prepare a special report that describes the cause of the discrepancy.
2. FSRC shall review the report.
3. The Site Vice President shall approve the report.

4. Within 30 days, Licensing (Station) shall submit the report to the NRC Regional Office. [TS 3.12.B.7.]

6.24.14 **Steam Generator Tube Inspection Report** [TS 6.6.A.3]

Engineering Programs - ISI/IST/Materials (Station) shall prepare a report for submittal by Licensing (Station) to the NRC within 180 days after Tavg exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, Steam Generator (SG) Program. The report shall be approved by the Site Vice President or Vice President Nuclear Engineering and shall include:

a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications.

e. Number of tubes plugged during the inspection outage for each active degradation mechanism.

f. Total number and percentage of tubes plugged to date.

g. The results of condition monitoring, including the results of tubes pulled and the in-situ testing.

h. The effective plugging percentage for all plugging in each SG.

6.24.15 **Reactor Pressure Vessel (RPV) Head Related Inspection Results**

See Step 6.23.14.

6.24.16 **Technical Specification (TS) Bases Changes**

See Step 6.23.15. [TS 6.4.J.4]

6.24.17 **Accident Monitoring Instrumentation Report**

a. With one required accident monitoring instrumentation channel inoperable, the inoperable channel shall be restored to operable status within 30 days or:

1. Licensing (Station) shall prepare a report that outlines the cause of inoperability and the plans and schedule for restoring the inoperable channel to operable status.

2. FSRC shall review the report.
3. The Site Vice President shall approve the report.
4. Licensing (Station) shall submit the report to the NRC within the next 14 days. **[TS 3.7.E.1]**

b. With two required accident monitoring instrumentation channels inoperable, an inoperable channel(s) shall be restored to operable status within 7 days or the preplanned alternate method of monitoring the appropriate function shall be initiated and:

1. Licensing (Station) shall prepare a report that outlines the preplanned alternate method of monitoring the function, the cause of inoperability, and the plans and schedule for restoring the inoperable channel to operable status.
2. FSRC shall review the report.
3. The Site Vice President shall approve the report.
4. Licensing (Station) shall submit the report to the NRC within the next 14 days. **[TS 3.7.E.2]**

6.24.18 **Reactivity Anomalies**

a. If the difference between the monthly observed and predicted steady-state boron concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and a LER shall be submitted to the Nuclear Regulatory Commission as specified in Step 6.10.11. **[TS 4.10]**

b. If the hot channel factors identified in Technical Specification 3.12 exceed their limits during periods of POWER OPERATION at greater than 10% of RATED POWER, an evaluation as to the cause of the anomaly shall be made and a LER shall be submitted to the Nuclear Regulatory Commission as specified in Step 6.10.11. **[TS 4.10]**

6.25 Technical Requirements Manual (TRM)

6.25.1 Fire Protection

If any fire protection or Appendix R functionality requirements in the TRM are not satisfied, the requirements of 10 CFR 50.72 and 10 CFR 50.73 shall be reviewed to determine if a reportable condition exists. If notification is required, refer to Steps 6.10.10 and 6.10.11 for immediate notification requirements and licensee event report system, respectively.

6.26 Environmental Protection Plan (North Anna)

6.26.1 Reporting Related to the VPDES Permit

a. Nuclear Licensing and Operations Support shall submit copies to NRC of VPDES permit violation reports excluding Discharge Monitoring Reports (DMR) at the same time they are submitted to the State Department of Environmental Quality (Water). See also Step 6.27.3.n. [EPP 5.4.2]

b. In the event:

- Of a change or addition to the VPDES permit
- A permit or certification appeal is stayed (entirely or in part)
- Dominion submits proposed changes to the effective VPDES permit to the State Department of Environmental Quality (Water)

1. The Environmental Compliance Coordinator shall prepare a notification of the event that includes a description of the event or situation and relevant supporting documentation.
2. Nuclear Licensing and Operations Support and Electric Environmental Services shall review the notification.
3. The Senior Vice President Nuclear Operations shall approve the notification.
4. Within 30 days, Nuclear Licensing and Operations Support shall submit the notification to NRC. [EPP 3.2]

6.26.2 Unusual or Important Environmental Events

a. If there is an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation, see Step 6.3.6.c.

b. If a notification is made as specified in Step 6.3.6.c., Electric Environmental Services shall prepare a report that:

- Describes, analyzes, and evaluates the event, including extent and magnitude of the impact and plant operating characteristics
- Describes the probable cause of the event
- Indicates the action taken to correct the reported event
- Indicates the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems
- Indicates the agencies notified and their preliminary responses

c. The Environmental Compliance Coordinator shall review the report.

d. The Site Vice President shall approve the report.

e. Within 30 days of an occurrence of an event reported in accordance with Step 6.3.6.c., Licensing (Station) shall submit the report to NRC. **[EPP 4.1 & 5.4.2]**

6.26.3 **Environmental Operating Report**

a. Each year, Environmental Compliance Coordinator shall prepare an environmental operating report for the previous year.

1. The report shall provide summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of the Environmental Protection Plan (EPP) for the report period.

2. The report shall include:
   - A comparison with operational controls, as appropriate, and previous nonradiological environmental monitoring reports
   - An assessment of the observed impacts of Station operation on the environment

3. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, the report shall include a detailed analysis of the data and proposed corrective action.

4. The report shall also include:
   - A list of EPP noncompliances and the corrective actions taken
   - A list of all changes in Station design or operation
   - A list of tests and experiments, done in accordance with EPP Subsection 3.1, that involved a potentially significant unreviewed environmental issue
   - A list of nonroutine reports submitted in accordance with Step 6.26.2.a.
5. If some results are not available by the report due date, the report shall note and explain missing results. Missing data shall be submitted as soon as possible in a supplementary report.

b. Electric Environmental Services shall review the report.

c. The Site Vice President shall approve the report.

d. Before May 1, Licensing (Station) shall review and submit the report to NRC.

6.27 State and Local Agency Regulations and Permits

NOTE: The NRC must be notified if another government agency is notified. See Step 6.3.4.a.4.

6.27.1 State Corporation Commission

NOTE: Power Supply provides estimated replacement power cost information to Regulation Services. Regulation Services submits the information to the State Corporation Commission.

a. **Unplanned Outages**

1. The Senior Vice President Nuclear Operations shall notify, by telephone, the State Corporation Commission Staff of unplanned outages. Notification should be within 48 hours. **[Commitment 3.2.10]**

2. For each unplanned outage, the Senior Vice President Nuclear Operations shall collect the following information:
   - Chronological sequence of events leading to the outage (a summary if a unit tripped)
   - Description of each major work item performed during the outage
   - Identification of root cause of any equipment failure or response that led to the outage, including any related company or industry experience with similar failures or responses
   - Corrective steps (if any) to avoid further or similar events
   - Outage duration and costs (for the total outage and for each major work item)

3. The Senior Vice President Nuclear Operations shall report the unplanned outage information to the State Corporation Commission by the time set by the Commission. **[Commitment 3.2.10]**

b. **Planned Outages**

1. For each planned refueling or maintenance outage, Outage & Planning shall collect and forward the following information to the Senior Vice President Nuclear Operations:
   - Planned schedule of events and duration of refueling/maintenance outage
   - Description of work performed during the outage
   - Explanation of differences between actual and planned work and outage duration
   - Duration of and costs associated with the outage

2. The Senior Vice President Nuclear Operations shall report the outage information to the State Corporation Commission by the time set by the Commission.

c. **Decommissioning Fund Status Report**

1. Site-specific decommissioning cost studies for each power station shall be revised every four years and transmitted to the NC PUC, Virginia SCC, and FERC **[Commitment 3.2.20]**. The site-specific cost estimate updates are coordinated by NL&OS and transmitted to the PUC, SCC, and FERC by the Treasury Department. See Step 6.10.13 for details as to the content and scope requirements of the site-specific cost estimates.

2. An annual update to the Decommissioning Trust Fund Status Report shall be submitted to the NC PUC, Virginia SCC, and FERC. The Status Report is to be prepared by the Treasury Department and submitted to Nuclear Licensing for review prior to transmittal by Treasury. A bi-annual update of the status of the decommissioning trust fund is submitted to the NRC by NL&OS. See Step 6.10.13.

6.27.2 **State Department of Emergency Management**

a. **Station Situations**

NOTE: These conditions may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure if the event may be of media significance.

1. Immediately, and in no case later than one hour, the Shift Manager shall notify the Manager Nuclear Operations, a Director, or the Site Vice President of any event that may be of media significance. Some examples (not all inclusive) of events to be evaluated for media significance are:
   - A reactor trip (while the reactor is critical) with unusual circumstances
   - An unplanned radioactive release that is reportable in accordance with 10 CFR 50.72
   - Contamination of multiple persons from the same release at the same time. For example, contamination resulting from a radioactive material release from a piping system, a release outside of a Radiological Control Area, or other circumstances that may be of media interest.
   - A radiological overexposure that is reportable in accordance with 10 CFR 20
   - Transport of a person injured during work-related activities by ambulance to an off-site medical facility
   - Special circumstances that may be of concern to nearby residents **[Commitment 3.2.8]**

2. Within one hour, the Manager Nuclear Operations or Operations Manager On Call shall notify the Site Vice President or a Director of a potentially media significant event.

3. The Site Vice President, a Director, Manager Nuclear Operations, or Shift Manager shall notify the Senior Vice President Nuclear Operations, and the Nuclear Public Affairs Director (or Corporate Security on weekends to contact the Public Affairs Duty Officer) of a potentially media significant event.

4. Should the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or the Public Affairs Duty Officer on weekends) determine that either Dominion or DEM intends to issue a news release, the Nuclear Public Affairs Director or Corporate News Services on weekdays if NPA Director cannot be reached (or the Public Affairs Duty Officer on weekends) shall inform the Operations Shift Manager. The Operations Shift Manager shall make a four-hour report to NRC in accordance with Step 6.3.4.a.4. **[Commitment 3.2.16]**

b. **Emergency Plan Activation Reports**
See Steps 6.3.5.b. and 6.3.7.

c. **EWS Availability Report**

1. Each year during January, Nuclear Emergency Preparedness shall prepare a report[1] of Early Warning System availability for the previous year that includes:
   - Preventive maintenance (growl) test results
   - A description of failures due to spurious activations
   - Corrective maintenance (full cycle) test results
   - A summary of miscellaneous outages
   - A performance summary

2. By January 31, Nuclear Emergency Preparedness shall submit the report to DEM. **(Reference 3.1.86)**

6.27.3 **State Department of Environmental Quality (DEQ)**

a. **Smoke**
See Step 6.3.4.b.

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1. This report supplies information to support the Annual Letter of Certification the State is required to submit to FEMA in accordance with FEMA Guidance Memorandum PR-1, Policy on NUREG-0654/FEMA-REP-1 and 44 CFR 350 Periodic Requirements, October 1, 1985.

b. **Asbestos Notification and Reporting Requirements**

NOTE: Insulation and certain other materials at the Station may contain asbestos. If potential asbestos-containing materials have not been identified as asbestos-free, then asbestos should be assumed to be present. The proper procedures should be followed when handling, removing, and disposing asbestos-containing materials. (see the Corporate Policy and Procedures Manual)

Asbestos removals are done by contractors who are required to have the necessary training and understanding of the regulatory requirements. However, Station personnel should have knowledge of the removal process in order to ensure that the contractors follow proper handling and disposal procedures.

1. The asbestos removal contractor is responsible for notifying the Virginia Department of Labor and Industry (DLI), if the removal amount is projected to be equal to or greater than the threshold amount of 10 linear feet or 10 square feet. **This written notification must be made at least 20 days prior to any of the asbestos-containing material being disturbed.**
2. The asbestos removal contractor is responsible for notifying EPA if the removal amount is projected to be equal to or greater than the threshold amount of 260 linear feet, 160 square feet, or 35 cubic feet of regulated asbestos-containing material. **This written notification must be made to EPA at least 10 working days prior to any of the asbestos-containing material being disturbed.**
3. A copy of the letter to EPA and the asbestos notification form(s) must be maintained on-site and available for review during annual inspections. A copy of this information must also be forwarded to the Environmental Compliance Coordinator Department (EES).
4. If the amount of asbestos to be removed changes by at least twenty percent, a re-notification to the regulatory agencies would be required **10 working days before asbestos removal work begins**. If the start date is revised to a later date, re-notification is required as soon **as possible before the original start date**. The ESS must be notified promptly of any such changes.

5. At times asbestos must be removed immediately to make emergency equipment repairs or for safety purposes. A trained on-site Station representative must be present before any asbestos can be removed. The emergency renovations should be reported to the EP&C as soon as possible to allow enough time to notify the regulatory agencies. The EP&C should be contacted for assistance in identifying an emergency removal.

c. **Waste Shipment Record Requirements**

1. Prior to any shipment of asbestos a waste shipment form must be completed and forwarded to the required individuals.

2. A signed copy of the waste shipment form must be received at the Station where the asbestos was removed **no later than 35 days after the shipment was accepted for transport**.
   - If the signed copy is not received by the originating facility within the **35-day** time frame, the Station shall initiate a search to determine the status of the shipment. The EES should be notified.
   - If the copy is not received by the originating Station within a **45-day** time frame, the EES should be notified no later than noon of the 45th day. The EES will make the required notification to the regulatory agencies.

d. **Radiological Contaminated Asbestos Waste (RCAW)**
The Environmental Protection Agency - NESHAPS Coordinator for Region III approved the Dominion procedure for tracking and disposing of low level RCAW. In order to maintain compliance with NESHAPS regulations the following EPA approved procedure must be utilized.

1. The Radiological Material Control Departments at each nuclear station will begin tracking the RCAW when it leaves the Station.

2. If the signed waste shipment record (WSR) is not returned to the Station by the waste disposal site operator **within 35 days**, the Station will determine the status of the shipment and notify Dominion EP&C. If the WSR is not received **45 days after shipment**, the Station will contact the EES and written notification will be made to the Environmental Protection Agency (EPA).

3. If the shipment is sent to an interim processor for volume reduction, and, following processing, the RCAW is to be transported to the ultimate disposal site in partial shipments, the interim processor will make a sufficient number of copies of the original WSR to accompany each partial shipment to the ultimate disposal site. Each WSR copy will be given an appropriate identification number (e.g., WSR Nos. 1A, 1B, 1C, etc.) and must include information describing the portion that this partial shipment represents of the original shipment. If the volume of the waste has been altered, this information must be described in the copy sent with the partial shipment. If the original shipment of RCAW is combined with other shipments, the original WSR will accompany the entire consolidated shipment.

4. The original WSR will be sent with the final shipment of the processed RCAW. The WSR will indicate that the shipment is closed and also indicate the identification numbers and amounts of the partial shipments sent earlier. The disposal site operator will forward the WSR copies and signed original to the Station. The Station must then notify the EES so that the EPA can be notified that the outstanding WSR has been received. The Station will retain the WSRs for at least two years.

5. Copies of the above written notifications will be sent to the Nuclear Regulatory Commission concurrent with transmission of the original notifications to EPA.

e. **Pollution Control Equipment Malfunction**

1. If any pollution control equipment malfunction results in excess opacity for more than one hour, the Environmental Compliance Coordinator shall notify the Department of Environmental Quality (DEQ) as soon as practical but no later than four daytime business hours after discovery.

2. After notification is made to the DEQ, within four daytime business hours of the occurrence, the Environmental Compliance Coordinator shall notify EES of the release including all the pertinent information obtained regarding the release.

f. **Auxiliary Boiler Operation (Surry)**

1. If either of the auxiliary boilers is to be in continuous operation for more than three hours, the boiler operator shall notify EES.

2. Electric Environmental Services shall notify DEQ (Air) regional inspector, if the inspector has requested an opportunity to observe the boiler in operation.

g. **Hazardous Material Releases**

If a hazardous material is released from the Station that poses an immediate or imminent threat to public health, see Step 6.3.2.e. See also Step 6.20.9.

[Code of VA § 10.1-1429]

h. **Bypassing**

1. If an unplanned bypass occurs, see Step 6.3.6.f.
2. If the need for a bypass is known in advance, Electric Environmental Services shall notify DEQ at least ten days before the bypass. **[VPDES Permit]**

i. **Discharge Monitoring Report**

1. If required by DEQ, the Environmental Compliance Coordinator shall prepare a report that describes general operational data for the month.
2. By the tenth of each month, the Environmental Compliance Coordinator shall submit a monitoring report (consisting of completed, original DMR forms and any other information required from time to time by DEQ) to DEQ.

   [VPDES Permit]

j. **Changes in Discharge or Management of Pollutants**

In the event of:

- Any new introduction of pollutants into treatment works or pollutant management activities which represents a significant increase in the discharge or management of pollutants which may interfere with, pass through, or otherwise be incompatible with such works or activities, if the Station was discharging or has the potential to discharge pollutants to State waters
- Any substantial change, whether permanent or temporary, in the volume or character of pollutants being introduced into treatment works, pollutant management activities, or discharge that was introducing pollutants into treatment works at the time the VPDES permit was issued
- Any reason to believe that an activity has occurred or will occur which would result in the discharge on a routine or frequent basis of any toxic pollutant which is not limited in the permit, if that discharge will exceed the highest of the following notification levels:
    - One hundred micrograms per liter
    - Two hundred micrograms per liter for acrolein and acrylonitrile
    - Five hundred micrograms per liter for 2, 4-dinitrophenol and for 1-methyl-4, 6-dinitrophenol
    - One milligram per liter for antimony
    - Five times the maximum concentration value reported for the pollutant in the permit application
    - The level established in accordance with regulation under 307(a) of the Act and accepted by the Board
- Any activity has occurred or will occur which would result in any discharge on a nonroutine or infrequent basis of a toxic pollutant which is not limited in the permit, if that discharge will exceed the highest of the following notification levels:
    - Five hundred micrograms per liter
    - One milligram per liter for antimony
    - Ten times the maximum concentration value reported for that pollutant in the permit application
    - The level established by the Board

1. Electric Environmental Services shall prepare a report that includes information on:
   - The characteristics and quantity of pollutants involved
   - Any anticipated impact of such change in the quantity and characteristics of the pollutants
   - Any additional information that may be required by DEQ
2. The Environmental Compliance Coordinator, the Director Nuclear Station Safety and Licensing, and the Plant Manager (Nuclear) shall review the report.
3. The Authorized Signatory shall approve the report.
4. Promptly, Electric Environmental Services shall submit the report to DEQ. **[VPDES Permit]**

k. **Groundwater Pumpage and Use Report (Surry)**

1. The Environmental Compliance Coordinator shall prepare a Groundwater Pumpage and Use Report each quarter.
2. The Environmental Compliance Coordinator shall submit the report to DEQ. **(Reference 3.1.11)**
3. Special condition 11.4 under the Groundwater Withdrawal Permit requires notice in writing to DEQ within 30 days of a major emergency that requires water withdrawals up to their pump capacity.

l. **Oil Discharge Contingency Plan (ODCP) Changes**

If significant changes occur, including:

- A change of licensee for the Station
- A substantial increase in the maximum oil storage capacity at the Station
- Decreased availability of private personnel or equipment necessary to remove, to the maximum extent practicable, the worst case release and to mitigate or prevent a substantial threat of such a release
- A change in type of product stored or handled at the Station for which an Material Safety Data Sheet (MSDS) has not been submitted

Within 30 days, Electric Environmental Services shall prepare and submit an amendment or revision to the ODCP to the DEQ Office of Spill Response and Remediation. **(Reference 3.1.10)**

m. **Operator Requirements**

If the Station does not employ or contract at least one operator who holds a current wastewater license appropriate for the facility, or there are grounds to expect that this situation will develop:

1. Electric Environmental Services shall prepare a report that provides reasons for noncompliance and a prompt schedule for achieving compliance.
2. The Environmental Compliance Coordinator, and the Director Nuclear Station Safety and Licensing, and the Plant Manager (Nuclear) shall review the report.
3. The Authorized Signatory shall approve the report.
4. Electric Environmental Services shall submit the report to DEQ.

[VPDES Permit]

n. **VPDES Permit Noncompliance**

When any requirement of the VPDES permit is not met:

1. If the noncompliance may adversely affect State waters or may endanger public health[1], see Step 6.3.2.f.
2. If the noncompliance is an unpermitted, unusual, or extraordinary discharge[2] that enters or could be expected to enter State waters, see Step 6.3.6.e.
3. If the noncompliance is an unplanned bypass, see Step 6.3.6.f.

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1. Applicable regulations use, but do not define, the terms “adversely affect” and “endanger public health.” These terms must be interpreted on a case-by-case basis by individuals with aquatic ecology expertise and thorough familiarity with current regulatory agency reporting and enforcement policy. Such individuals will also determine how soon a specific event must be reported to avoid enforcement (i.e., within minutes of an event, or some longer time within the not-to-exceed 24-hour limit established by the VPDES Permit).
2. Unusual or extraordinary discharge includes, but is not limited to: a) unplanned bypasses, b) upsets, c) spillage of materials resulting directly or indirectly from processing operations or pollutant management activities, d) breakdown of processing or accessory equipment, e) failure of or taking out of service, sewage or industrial waste treatment facilities, auxiliary facilities, or pollutant management activities, or f) flooding or other acts of nature. [VPDES Permit]

4. Electric Environmental Services shall prepare a letter report that includes:
   - A description and cause of noncompliance
   - The period of noncompliance, including exact dates and times or the anticipated time when the noncompliance will cease
   - Actions taken or to be taken to reduce, eliminate, and prevent recurrence
5. The Director Electric Environmental Services shall approve the report.
6. As directed by DEQ, Electric Environmental Services shall submit the report to DEQ within five days or as an attachment to the discharge monitoring report. See Step 6.27.3.i. **[VPDES Permit]**
7. Electric Environmental Services shall send a copy of the report to Nuclear Licensing and Operations Support. (See Step 6.26.1. **(North Anna)**)
8. Nuclear Licensing and Operations Support shall submit copies of VPDES permit violation reports excluding DMRs to NRC and to the NRC Regional Office at the same time they are submitted to DEQ. **(Surry) [Commitment 3.2.4]**

If Dominion submits proposed VPDES permit changes to DEQ:

9. **GO TO** Step 6.26.1.b. **(North Anna)**
10. Electric Environmental Services shall coordinate the DEQ submittal with Nuclear Licensing and Operations Support.
11. At the same time the proposed changes are submitted to DEQ, Nuclear Licensing and Operations Support shall submit a copy to NRC and to the NRC Regional Office. **(Surry) [Commitment 3.2.4]**

o. **VPDES Permit Changes**

If the VPDES permit is changed:

1. **GO TO** Step 6.26.1.b. **(North Anna)**
2. Electric Environmental Services shall notify Nuclear Licensing and Operations Support when DEQ approves a VPDES permit change.
3. Within 30 days after DEQ approves a VPDES permit change, Nuclear Licensing and Operations Support shall notify NRC and the NRC Regional Office. **(Surry) [Commitment 3.2.4]**

p. **Pump and Haul Activities (Surry)**

1. Electric Environmental Services shall prepare a report for all pump and haul activities that involve removal of tank-bottom waters from the bulk storage tanks. The report shall include:
   •The name of the responsible haul contractor
   •The date and time the haul occurred
   •The final destination and disposition of the waste
   •The quantity of waste hauled
2. By the tenth of the following month, Electric Environmental Services shall submit the report to DEQ. **[VPDES Permit]**

q. **Temperature Monitoring Program (North Anna)**

1. Electric Environmental Services shall prepare a Temperature Monitoring Program report annually.
2. By March 31 of each year, Electric Environmental Services shall submit the data to DEQ. **[VPDES Permit]**

r. **Water Withdrawals**

1. Each year, the Environmental Compliance Coordinator shall prepare a Water Withdrawals Report.
2. By January 31, the Environmental Compliance Coordinator shall submit the report for the prior year to DEQ. **(Reference 3.1.11)**

s. **Underground Oil Storage Tanks (UST)**

1. The Environmental Compliance Coordinator Department must be notified if any USTs are added, removed, modified, or closed.
2. The Environmental Compliance Coordinator Department must notify the State within thirty days of new or existing tank installations and any changes in tank usage.
3. The Environmental Compliance Coordinator Department must also notify the State within thirty days prior to closure of any UST.

t. **Above Ground Oil Storage Tanks (AST)**

1. The Environmental Compliance Coordinator Department must be notified if any ASTs are added, removed, modified, or closed.
2. The Environmental Compliance Coordinator Department must notify the State within thirty days of new or existing tank installations and any changes in tank usage.

6.27.4 **State Department of Health**

a. **Operation Report Meter Readings (North Anna)**

1. The Environmental Compliance Coordinator shall prepare an Operation Report Meter Readings report each month.
2. By the tenth of the following month, the Environmental Compliance Coordinator shall submit the report to the State Department of Health.
3. Additional reporting requirements for bacteriological and chemical analysis of drinking water are met by following the stations drinking water permits and by the direction of the State Department of Health.

b. **Sewage Treatment Plant Operation Report (Surry)**

By the fifteenth of the following month, the Environmental Compliance Coordinator shall submit the Sewage Treatment Plant Operation report to the State Department of Health.

c. **Waterworks Operation (Surry)**

1. By the fifteenth of the following month, the Environmental Compliance Coordinator shall submit the Waterworks Operation report to the State Department of Health.
2. Reporting requirements are met for bacteriological and chemical analysis of drinking water by following the bacteriological monitoring plan and by direction of the State Department of Health.

6.27.5 **State Department of Labor and Industry**

a. See Step 6.27.3.b. (Asbestos Notification and Reporting Requirements).

b. **Voluntary Protection Program (VPP) [Commitment 3.2.29]**

1. Nuclear Site Safety shall prepare an annual self-assessment.
2. By the fifteenth of February, Nuclear Site Safety shall submit the report to the State Department of Labor and Industry.

6.28 Nuclear Insurance

6.28.1 Evaluation Reports

If Nuclear Electric Insurance Limited (NEIL) or American Nuclear Insurers (ANI) provides Dominion with an inspection report that contains compliance recommendations:

a. The Supervisor Nuclear Site Safety (Station) shall prepare a notification letter to the appropriate insurer to convey the Dominion response and shall ensure the letter is submitted in accordance with current insurance policy provisions.

b. The Supervisor Nuclear Site Safety (Station) shall ensure status reports are submitted to the appropriate insurer, in accordance with current insurance policy provisions, until recommendations are closed or withdrawn.

6.28.2 Adverse Conditions

a. Immediately upon notification of a potential adverse condition (see Subsection 4.4), the Supervisor Nuclear Site Safety (Station) shall confer with the Director Corporate Risk Management to determine whether the condition is reportable in accordance with current insurance policy requirements.

b. If the condition is determined to be reportable, promptly, the Supervisor Nuclear Site Safety (Station) shall notify NEIL, by telephone or facsimile, of the adverse condition or loss.

c. Promptly, if significant additional information related to an adverse condition is obtained, Supervisor Nuclear Site Safety (Station) shall notify NEIL, by telephone or facsimile.

d. The appropriate Station department shall prepare a report to document the adverse condition.

e. The Site Vice President shall approve the report and forward it to the Supervisor Nuclear Site Safety (Station).

f. Within 30 days after discovery, the Supervisor Nuclear Site Safety (Station) shall submit the report to NEIL.

6.28.3 **Incidents**

NOTE: NEIL encourages related events be reported, even though they are not incidents as defined in Subsection 4.23.

a. If an event is potentially an incident (see Subsection 4.23) or involves off-site transport of radioactive materials, the Supervisor Nuclear Site Safety (Station) shall notify the Director Corporate Risk Management.

b. The Supervisor Nuclear Site Safety (Station) shall prepare an event report.

c. The Supervisor Nuclear Site Safety (Station) shall confer with the Director Corporate Risk Management to determine whether the event is reportable in accordance with current insurance policy requirements.

d. If the event is determined to be reportable to NEIL, or a determination is made to submit an information report, as soon as practicable, but within 15 business days after the event, the Supervisor Nuclear Site Safety (Station) shall submit the report to NEIL.

e. If the event involves off-site transport or release of radioactive material, Corporate Risk Management shall determine whether the event is reportable to ANI. If determined to be reportable:

1. Promptly, Corporate Risk Management shall notify ANI by telephone.
2. Corporate Risk Management shall prepare a confirmatory letter to document the notification and shall ensure the letter is submitted promptly to ANI.

6.28.4 **Fire System Impairment**

If a fire system is prevented from performing its intended function, for whatever reason, and the duration of the impairment is expected to exceed 48 hours:

a. As soon as practicable the Supervisor Nuclear Site Safety shall notify NEIL of the impairment by telephone or fax.

b. As soon as practicable, by telephone, letter, or fax, the Supervisor Nuclear Site Safety shall notify NEIL that the impairment has been corrected.

6.28.5 **INPO Ratings and Membership**

a. **Suspension or Downgrading**

1. Within three days after receipt of notice that Dominion membership in INPO has been suspended or cancelled, or INPO has placed the Station in Category 5, the Site Vice President, an Director or the Director NL&OS shall notify the Director Corporate Risk Management.
2. Within five days after receipt of notice, Corporate Risk Management shall notify NEIL.

b. **Upgrading**

1. Within 30 days after receipt of notice that INPO has placed the Station in Category 1, the Site Vice President, a Director, or the Director NL&OS shall notify the Director Corporate Risk Management.
2. Corporate Risk Management shall include this information as part of its annual endorsement submittals to NEIL.

6.28.6 **License Status**

Upon receipt of notice that the NRC license to operate has been revoked or suspended, or that NRC has issued a shutdown order:

a. The Site Vice President, an Director or the Director NL&OS shall notify the Director Corporate Risk Management.

b. Promptly, Corporate Risk Management shall notify NEIL and ANI.

6.29 Discretionary Reports

Submittal letters for special reports required by this subsection shall contain the same information as LER submittal letters. Submittal letters shall omit the headings for LERs.

6.29.1 **Station Blackout Alternate AC Source**

If the Station Blackout Alternate AC Source is out of service for 14 consecutive days, the requirements of 10 CFR 50.72 and 10 CFR 50.73 will be reviewed to determine if a reportable condition exists. If notification is required, refer to Steps 6.10.10 and 6.10.11 for immediate notification requirements and licensee event report system, respectively.

6.29.2 **AMSAC**

If the ATWS Mitigation System Actuation Circuit (AMSAC) is out of service (see Attachment 7, ATWS Mitigation System Actuation Circuitry (AMSAC) Functionality) for 30 consecutive days, the requirements of 10 CFR 50.72 and 10 CFR 50.73 will be reviewed to determine if reportable condition exists. If notification is required, refer to Steps 6.10.10 and 6.10.11 for immediate notification requirements and licensee event report system, respectively.

6.29.3 **Regulatory Guide 1.97 Variables**

If instruments (not otherwise specified in Technical Specifications for post accident monitoring), that are required to measure Type A, B, or C variables specified in Regulatory Guide 1.97[1], are out of service for 30 consecutive days, the requirements of 10 CFR 50.72 and 10 CFR 50.73 will be reviewed to determine if reportable condition exists. If notification is required, refer to Steps 6.10.10 and 6.10.11 for immediate notification requirements and licensee event report system, respectively.

6.29.4 **Seismic Monitoring System**

If the Seismic Monitoring System is out of service for 30 consecutive days, the requirements of 10 CFR 50.72 and 10 CFR 50.73 will be reviewed to determine if reportable condition exists. If notification is required, refer to Steps 6.10.10 and 6.10.11 for immediate notification requirements and licensee event report system, respectively.

6.29.5 **Mishaps Involving Low Level Waste (LLW) Forms**

a. If an LLW form event is identified as reportable in accordance with VPAP-2104, Radioactive Waste Process Control Program (PCP), Radiological Protection shall prepare a special report that describes, as a minimum:
   - The mishap
   - The cause of the mishap
   - Immediate corrective actions or compensatory measures
   - Corrective actions to prevent recurrence

b. FSRC shall review the report.

1. North Anna TRM Section TR 3.3.9, Virginia Power Technical Report No. PE-0013, North Anna Power Station Response to Regulatory Guide 1.97, and Virginia Power Technical Report No. PE-0014, Surry Power Station Response to Regulatory Guide 1.97, identify applicable RG 1.97 instruments and variable types.

c. The Site Vice President shall approve the report.

d. Within the next 30 days, Licensing (Station) shall submit the report to the NRC Division of Low-Level Waste Management and Decommissioning and a copy to the South Carolina Department of Health and Environmental Control. **[Commitment 3.2.7]**

6.29.6 **Otherwise Unreportable Items of Some Safety Merit**

If a situation arises that has some safety merit, but is not required to be reported by any other part of this procedure (see also Step 6.10.1), Licensing (Station), as directed by the Site Vice President, shall write a discretionary LER. Refer to Step 6.10.11 for licensee event report system.

6.29.7 **Directive on Shift Manager's Responsibilities**

Each year, the Senior Vice President Nuclear shall reissue a Directive on Shift Managers' Responsibilities stating that the Shift Manager (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function and shall be the only individual that may direct the licensed activities of licensed operators. The directive shall be posted on bulletin boards for regulatory required documents.

6.30 Consolidated Data Entry (CDE) Reporting System

Data needed to populate INPO's - CDE Database shall be collected and reported in accordance with LI-AA-500, NRC/INPO/WANO Performance Indicator and MOR Reporting.

6.31 Groundwater Protection Voluntary Communication Notification and Reports

NOTE: VPAP-2103N Offsite Dose Calculation Manual (North Anna), and VPAP-2103S Offsite Dose Calculation Manual (Surry) contain the guidance as to whether or not a sample result or a spill or leak meets the NEI Industry Initiative on Groundwater Protection.

6.31.1 Notification Protocol

a. The Shift Manager will notify the Supervisor Licensing (Station) that the Abnormal Procedure for accidental, unplanned or uncontrolled radioactive liquid release was entered and voluntary communications may be required in accordance with the NEI Industry Initiative on Groundwater Protection for a spill, leak or groundwater sample results (either onsite or offsite)

NOTE: Notification to the following individuals will need to be made in a timely manner to allow required notifications to local/state/federal stakeholders.

b. The Supervisor Licensing (Station) or designee will contact the following Dominion personnel:

1. Site Vice President
2. Director, Electrical Environmental Services
3. Director, Nuclear Licensing and Operations Support (NL&OS)
4. Director, Nuclear Protection Services & EP

NOTE: Notifications to the local/state/federal stakeholders are to be made by the close of the next business day.

NOTE: When establishing communications with the State/Local officials, the GPI Notification matrix, maintained by Nuclear Public Affairs, is followed.

NOTE: When communicating to the State/Local officials, be clear and precise on quantifying the actual release information as it applies to the appropriate regulatory criteria.

c. The following individuals will contact the local/state/federal stakeholders:

1. Site Vice President or designee will contact the County Administrator.

2. Director, Nuclear Protection Services & EP or designee will contact the Virginia Department of Health and the Virginia Department of Emergency Management.
3. Director, Electrical Environmental Services or designee will contact the Virginia Department of Environmental Quality.
4. Director NL&OS or designee will contact the NRC Region II Branch Chief, the NRC Project Managers, NEI, and ANI.
5. Supervisor Licensing (Station) or designee will contact the NRC Senior Resident Inspector.

6.31.2 **Reports**

a. A written 30-day NRC report is required for all sample results (either onsite or offsite) that exceed the REMP/ODCM reporting criterion and could potentially reach the groundwater that is or could be in the future used as a source of drinking water.

b. It is not expected that a written 30-day report will be generated each time a subsequent sample(s) from the same "plume" identifies concentrations greater than the REMP/ODCM criterion.

c. Licensing (Station) shall prepare a notification letter to the NRC that includes:
   - Description of the event
   - Corrective measures taken
   - Actions to prevent recurrence
   - Any environmental or public health and safety consequences

d. The Site Vice President shall review and approve the report.

e. A copy will be provided to local and state officials.

7.0 RECORDS

7.1 The following individual and packaged documents and copies of any related correspondence completed as a result of implementing or performing this procedure are records. They shall be transmitted to Records Management in accordance with RM-AA-101, Record Creation, Transmittal, and Retrieval. Before transmittal, the sender shall assure that:

- Each record is packaged when applicable
- QA program requirements have been fulfilled for Quality Assurance records
- Each record is legible, completely filled out, and adequately identifiable to the item or activity involved
- Each record is stamped, initialed, signed, or otherwise authenticated and dated, as required by this procedure

7.1.1 **Individual Records**

- All reports required by this procedure, including transmittal letters, except as specifically excluded by Subsection 7.2
- Supporting logs for notifications required by this procedure

7.1.2 **Record Packages**

None

7.2 The following documents completed as a result of implementing this procedure are **not** Quality Assurance records and are not required to be transmitted to Records Management.

- Documentation of item postings and removal per Step 6.5.1
- Fitness for duty reports—not submitted to NRC, including information specifically prepared for inclusion in or as a basis for these reports; these documents shall be retained by the Supervisor Management Information and Planning for a minimum of three years
- Documentation of oil releases to the ground up to 25 gallons, in lieu of reporting, shall be retained by the Environmental Compliance Coordinator for a minimum of five years

Dominion

Oil or Hazardous Substance Release Report

VPAP-2802 - Attachment 1 **Page 1 of 2**

This report is supplemental to Plant Issue (Deviation) Number ____________________ Date

1.* Shift Manager on Duty (Name)

2.* Release Reported By (Name) * Primary Phone Number

3.* Date Release Occurred *Time Release Occurred

4.* Type, Origin, and Location of Release (Exactly)

5.* Estimated Amount (Exact Quantity if Known)

6.* Material Discharged (for oil spills use the codes found in the SPCC Oil Spill Report form):

7.* Reason
☐ Personnel Error ☐ Other (explain) ____________________
☐ Equipment ____________________

8.* Status of Release
☐ Continuing ☐ Other (explain) ____________________
☐ Stopped ____________________

9. Dominion Personnel Notified

	Time (2400 Hours)	Date	By (Signature)	Name of Individual Notified (Print)
Station				
Electric Environmental Services				

10. Agency Notifications

	Time (2400 Hours)	Date	By (Signature)	Name of Individual Notified (Print)
Va. DEQ				
NaRC				
VA ERC				
LEPC				

11.* Did release reach navigable waters or ground? ☐ Yes ☐ No
12.* If No, is there a potential for release? ☐ Yes ☐ No

13.* Was release contained? ☐ Yes ☐ No

Key: LEPC-Local Emergency Planning Coordinator; NaRC-National Response Center; ERC-Emergency Response Center; DEQ-Department of Environmental Quality; SPCC-Spill Prevention, Control, and Countermeasures

*** - Information is required to be reported to the agencies listed in block 10.**

Oil or Hazardous Substance Release Report

VPAP-2802 - Attachment 1 **Page 2 of 2**

14.* Clean Up Procedure (Explain if Known)

15.* Statement of Shift Manager Concerning How Release Occurred

16. Send a copy of this report within 24 hours to the Environmental Compliance Coordinator and to Electric Environmental Services.

Copy sent by ______________________________ (name required).

17. Upon completion, send telecopy of this report to the Vice President Nuclear Operations.

Telecopy initiated by ______________________________ (name required).

Completed By (Name)	Date
16. Send original to Manager Nuclear Operations	
Reviewed By Manager Nuclear Operations (Signature)	Date

*** - Information is required to be reported to the agencies listed in block 10.**

ATTACHMENT 2

(Page 1 of 2)

FERC Public Safety Database Report - Instructions

1. **Project Number - State:** As shown.
2. **Name of Project & Name of Development:** As shown.
3. **Licensee or Exemptee:** As shown.
4. **River or Stream:** As shown.
5. **Date of Incident & Time of Incident:** Self-explanatory, however, enter unknown, if applicable.
6. **Licensee Report Dates:** Self-explanatory.
7. **Description of Incident:** Enter a brief, clear, description of the incident, including who was involved, and where and when the incident occurred. FERC considers this the most important entry.
8. **Location of Incident:** Select the one category that best describes the location of the incident. If no listed category is appropriate, describe as “Other” but do not duplicate the material in “Description of Incident.”
9. **Number of people involved in incident:** Self-explanatory.
10. **Type of Activity:** Select one category that best describes the incident.
11. **Result from Incident:** Indicate if incident was a drowning or electrocution. Leave blank if neither.
12. **Preparer Signature and Phone:** Self-explanatory.

ATTACHMENT 2

(Page 2 of 2)

FERC Public Safety Database Report

Atlanta Regional Office
Federal Energy Regulatory Commission
Public Safety Database

1. Project Number-State: 6335-VA
2. Name of Project: North Anna Hydroelectric Project
   Name of Development: —
3. Licensee or Exemptee: Virginia Electric and Power Company
4. River or Stream: North Anna River
5. Date of Incident __/ __/ __/ Time of Incident ________________________________
6. Licensee Report Dates: Verbal ___/ ___/ ___ Written ___/ ___/ ___
7. Description of Incident (20 words or less): ____________________________
   ____
   ____
   ____
   ____
8. Location of Incident (check one): Reservoir or Upstream ____; Tailrace ____; Intake ____; Downstream ____; Canal ____; Conduit ____; Penstock ____; Powerhouse ____; Substation ____; Spillway or Dam ____; Project Land ____; Other (describe): _________
   ____
   ____
9. Number of people involved in incident ____; Injured ____; Fatalities ____; No injury or fatality ____
10. Type of Activity (check one): Boating ____; Injured ____; Auto Vehicle ____; Fall ____; Inspection/Maintenance ____; Bank Fishing ____; Boat Fishing ____; Suicide ____; Natural Causes ____; Homicide ____; Construction ____; Unknown ____; Other ______
    ____
11. Result from Incident: Drowning ____; Electrocution ____

- -

This Form Prepared by: ____ __________
(Name) (Date)

____
(Title)

Phone No. (___) ___-____________

ATTACHMENT 3

(Page 1 of 4)

Emergency Response Unavailability

Systems and facilities that are part of the Station emergency response capability are **unavailable** if:

a. **Safety Parameter Display System (SPDS) (part of the PCS)**

The following define SPDS nonfunctionality conditions. Their existence for more than eight hours is considered a major loss of accident assessment (see 6.3.5.a.6.). Engineering judgement may be needed to assess the significance of losing certain equipment. (see also VPAP-2602, Safety Parameter Display System (SPDS)(Surry), VPAP-2606, Safety Parameter Display System (SPDS) (North Anna):

- System time is not updating
- Unit's mode is invalid or incorrect and cannot be corrected
- No Technical Support Center (TSC) CRT/keyboard is available
- Neither Control Room Unit CRT/keyboard is available for either Unit
- Both Local Emergency Operations Facility (LEOF) and Central Emergency Operations Facility (CEOF) SPDS data links are unavailable
- Any top-level SPDS bar is failed and the failure is not due to the associated field equipment being nonfunctional (i.e., failure is not computer related)

ATTACHMENT 3

(Page 2 of 4)

Emergency Response Unavailability

b. **Emergency Response Facilities (ERFs)**

NOTE: ERFs may be degraded, but still considered functional when the PCS is nonfunctional in one or more facilities if alternative methods to acquire and distribute plant data are available. Reportability of PCS loss is addressed in the PCS section of this attachment.

NOTE: ERFs may be degraded, but still considered functional when radiological monitoring or ventilation systems are nonfunctional when an ERF is activated in an emergency that does not present an immediate habitability problem. The determination would be made by the appropriate facility manager during facility activation.

1. **TSC**
   - Voice (direct or indirect) communication is unavailable between the TSC and any of the following:
     - •• Control Room
     - •• LEOF
     - •• Operational Support Center (OSC)
     - •• NRC Operations Center
     - •• State Emergency Operations Center
     - •• Local Emergency Operations Centers
   - Electrical service is unavailable for more than one hour
2. **LEOF**
   - Voice (direct or indirect) communication is unavailable between the LEOF and any of the following:
     - •• TSC
     - •• NRC Operations Center
     - •• State Emergency Operations Center
     - •• Local Emergency Operations Centers
   - Electrical service is unavailable for more than one hour

ATTACHMENT 3

(Page 3 of 4)

Emergency Response Unavailability

b. **Emergency Response Facilities (ERFs) (continued)**

3. **CEOF**

- Voice communication (direct or indirect) is unavailable between the CEOF and any of the following:
  - •• TSC
  - •• NRC Operations Center
  - •• State Emergency Operations Center
  - •• Local Emergency Operations Centers
- Electrical service is unavailable for more than one hour

c. **Emergency Communications**

NOTE: If unavailability of emergency communications also constitutes unavailability of the TSC, LEOF, CEOF, or the OSC, only a single notification is required.

NOTE: If NRC Emergency Telecommunications System (ETS) functionality is provided using licensee corporate communications systems, the NRC Operations Center should be informed through any means available of any communication failures which render ETS communication functions unavailable. This does not apply to minor interruptions in portions of the site or corporate telecommunications systems. It is intended to apply to serious conditions during which the telecommunications system can no longer fulfil the communications requirements of the Emergency Plan or provide ETS functionality. **[Commitment 3.2.22]**

- The Emergency Notification System (NRC) not available
- No means exists to contact the State and risk jurisdictions (e.g., INSTA phone, EOC ringdown, commercial communications)

d. **Early Warning System**

- There is a **total** inability to actuate the system
- More than 25 percent of all sirens are unavailable
- The capability to alert a large segment of the population does not exist

ATTACHMENT 3

(Page 4 of 4)

Emergency Response Unavailability

e. **Plant Monitors**

Fewer than the minimum number of channels are operable per Technical Specification:

- (Table 3.3.3-1) for longer than 48 hours **(North Anna)**
- (Table 3.7-6) and the applicable LCOs of Technical Specification 3.7.E have been exceeded and the 12-hour action statements to Hot Shutdown have been entered **(Surry)**

Dominion

Significant Fitness for Duty Violation or Programmatic Failure/ Drug or Alcohol Testing Errors NRC 24 Hour Notification

VPAP-2802 - Attachment 4 **Page 1 of 1**

Instructions

1. Fitness for Duty Administrator shall complete Section A and submit this form to station management.
2. Station management shall notify the NRC Operations Center within 24 hours of Event being reported.
3. Station Operations or Licensing shall complete Section B and return this form to the Fitness for Duty Administrator.
4. Fitness for Duty Administrator shall file original and distribute copies as specified on the bottom of this form.

Section A Event Information

1. Event	2. Event Time [ ] A.M. [ ] P.M.

3. Event Classification

A [ ] Sale, use or possession of illegal drugs or consumption or presence of alcohol within the Protected Area.

B [ ] Acts by a supervisor, FFD Program personnel, or licensed operator involving sale, use, or possession of a controlled substance, use of alcohol within the protected area, determination of unfitness for scheduled work due to consumption of alcohol.

C [ ] Any intentional act that casts doubt on the integrity of the FFD program.

D [ ] Any programmatic failure, degradation, or discovered vulnerability of the FFD program that may permit undetected drug or alcohol use or abuse by individuals within the protected area.

E [ ] Drug and Alcohol Testing Errors.

4. Event Description

5. Personnel Involved in Event

Names	**Job Descriptions**

6. Report Prepared By (Signature)	Date	Time of Report [ ] A.M. [ ] P.M.
7. Report Approved By (Signature)	Date	Time Approved [ ] A.M. [ ] P.M.

Section B Notification Summary

8. Name of Individual in NRC Contacted	Date	Time NRC Contacted [ ] A.M. [ ] P.M.

9. Distribution: Fitness For Duty Program Manager; Site Vice President; Fitness For Duty Administrator for Participant File; Director Nuclear Licensing and Operations Support

ATTACHMENT 5

(Page 1 of 4)

10 CFR 50.73 Reportability Guidelines

This Attachment provides guidance to clarify situations that may be reportable in accordance with the requirements of 10 CFR 50.73(a).

a. **10 CFR 50.73(a)(2)(i) - Shutdowns, Technical Specification Violations, 10CFR50.54(x)**

1. "Shutdown" as used in this paragraph is the time when Technical Specifications require the unit to be in the **first** LCO-required shutdown condition (e.g., Mode 3, hot standby). If a condition is corrected before the time limit for shutdown (i.e., before completion of the shutdown), the event need not be reported. A condition (e.g., a degraded mode allowed by Technical Specifications) that exists longer than permitted by Technical Specifications, discovered after the Technical Specification time limit, **is** reportable even if rectified immediately after its discovery.

2. Although failure to meet administrative requirements of Technical Specifications is a violation, an LER is not required if the violation is administrative only and does not result in operation prohibited by the Technical Specifications. Failure to obtain FSRC approval for non-intent procedure changes within 14 days (a violation of Technical Specifications) or an organizational structure change that has not yet been approved as a Technical Specifications revision are examples of situations that do not require LERs in accordance with 10 CFR 50.73(a)(2)(i)(B).

a. **10 CFR 50.73(a)(2)(ii) - Unanalyzed Conditions**

3. Engineering judgment and experience may be necessary to determine whether a condition is unanalyzed. A minor variation in individual parameters or problems involving single pieces of equipment are excluded (e.g., at any time, one or more safety-related components may be out of service due to testing, maintenance, or a fault that has not yet been repaired). Any trivial single failure or minor error in performing surveillance tests could produce a situation in which two or more, often unrelated, safety-related components are out of service. Technically, this is an unanalyzed condition. However, these events should be reported only if they involve functionally related components or if they significantly compromise unit safety. Small voids in systems designed to remove heat from the reactor core—that have been previously shown through analysis to be not safety significant—are **not** reportable.

ATTACHMENT 5

(Page 2 of 4)

10 CFR 50.73 Reportability Guidelines

a. **10 CFR 50.73(a)(2)(ii) - Unanalyzed Conditions (continued)**

Accumulation of voids that could inhibit the ability to remove heat adequately from the reactor core, particularly under natural circulation conditions, may constitute an unanalyzed condition that **is** reportable.

4. Situations that **are** reportable include:
   - Fuel cladding failures in the reactor or in the storage pool that exceed expected values, that are unique or widespread, or that resulted from unexpected factors and would involve a release of significant quantities of fission products
   - Cracks and breaks in piping, the reactor vessel, or major components in the primary coolant circuit that have safety relevance (steam generators, reactor coolant pumps, valves)
   - Significant welding or material defects in the primary coolant system
   - Serious temperature or pressure transients (e.g., transients that violate Technical Specifications)
   - Loss of relief or safety valve operability during test or operation (so the number of operable valves is less than required by Technical Specifications)
   - Loss of containment function or integrity (e.g., containment leakage rates exceeding authorized limits, loss of containment isolation valve function during tests or operation, loss of main steam isolation valve function during test or operation, or loss of containment cooling capability)

b. **10 CFR 50.73(a)(2)(iii) - External Threats**

5. 10 CFR 50.73(a)(2)(iii) applies only to acts of nature and external hazards (e.g., railroad tank car explosion). Acts of sabotage are addressed by 10 CFR 73.71.
6. A minor brush fire in a remote area of the site that is quickly controlled by fire fighting personnel is **not** reportable. A major forest fire, large-scale flood, or major earthquake that presents a clear threat to the Station **is** reportable. Industrial accidents near the Station that create a Station safety concern **are** reportable.

ATTACHMENT 5

(Page 3 of 4)

10 CFR 50.73 Reportability Guidelines

c. **10 CFR 50.73(a)(2)(iv) - Actuations**

7. “Actuation” of multi-channel systems occurs when enough channels are actuated to cause activation of the system. Single channel actuations, whether caused by failures or otherwise, are **not** reportable if they do not complete the minimum actuation logic.
8. If planned procedure calls for a manual reactor trip, but conditions develop during the shutdown that require an automatic trip, the trip **is** reportable.
9. A preplanned sequence that implies a procedure step indicates that a specific actuation will be generated and the control room personnel are aware of its specific signal generation **before** its occurrence or indication in the control room. (See note at Step 6.3.4.a.3.)

d. **10 CFR 50.73(a)(2)(v) and (vi) - Events That Could Have Prevented Fulfillment of a Safety Function**

10. A potentially serious human error that could have prevented fulfillment of a safety function **is** reportable even if recovery factors resulted in the error being corrected (e.g., an individual who improperly operates or maintains a component could have made the same error at functionally redundant components). The actions must affect or involve components in more than one train or channel of a safety system, and the result of the actions must be undesirable from the perspective of protecting the health and safety of the public. The components need not be functionally redundant.
11. Engineering judgment is necessary to determine whether a failure or operator action that disabled one train of a safety system could have, but did not, affect a redundant train within that ESF system. If a redundant train could have been affected, the event **is** reportable.
12. A component that fails by an apparently random mechanism **may be** reportable. A failure **is** reportable if it constitutes a condition for which there is reasonable doubt that a functionally redundant train or channel would remain operational until it completed its safety function or is repaired (e.g., if a pump in one train of an ESF system fails because of improper lubrication, and engineering judgment is that there is a reasonable belief that a functionally redundant pump in another train was also improperly lubricated and would have also failed before it completed its safety function, then the actual failure **is** reportable and the potential failure of the functionally redundant pump must be discussed in the LER).

ATTACHMENT 5

(Page 4 of 4)

10 CFR 50.73 Reportability Guidelines

d. **10 CFR 50.73(a)(2)(v) and (vi) - Events That Could Have Prevented Fulfillment of a Safety Function (continued)**

13. Failure of two or more trains in safety systems that include three or more trains **is** reportable if the functional capability of the overall system was jeopardized.

14. A lost or degraded non-safety service (e.g., heating, ventilation, cooling) or input (e.g., compressed air) **is** reportable if proper fulfillment of a safety function is not or cannot be assured. Failures that affect input or services to systems with no safety function are **not** reportable.

e. **10 CFR 50.73(a)(2)(vii) - Failure of Independent Portions of Multiple Trains/Channels**

15. An event or failure **is** reportable if it results in or involves failure of independent portions of more than one train or channel in the same or different systems that have a safety function (e.g., a cause or condition causes components in Train A and B of a single system to become inoperable, even if additional trains were still available).

16. If part of a system is removed from service to perform maintenance, and the Technical Specifications permit the resulting configuration, and the system or component is returned to service within the time limit specified in the Technical Specifications, the action is **not** reportable. A condition, identified while a train or component is out of service, that could have prevented the whole system from performing its intended function **is** reportable.

f. **10 CFR 50.73(a)(2)(viii) - Effluent Releases**

Reports made to the NRC in accordance with Paragraph (viii) also meet the effluent release reporting requirements of 10 CFR 20.2203 (a)(3).

g. **10 CFR 50.73(a)(2)(x) - Internal Threats**

In-plant releases **are** reportable if they require evacuation of rooms or buildings that contain systems important to safety and, as a result, the ability of operators to perform necessary safety functions is significantly hampered. Precautionary evacuations of rooms and buildings subsequently determined to have been not required are **not** reportable.

ATTACHMENT 6

(Page 1 of 1)

Example DEM Summary Report

SURRY POWER STATION

SUMMARY REPORT
Emergency Classification: Notification of Unusual Event
Initiated: September 15, 1999, 2200 hours
Terminated: September 16, 1999, 1420 hours

EVENT
Hurricane force winds from Hurricane Floyd projected to be onsite within 12 hours.

DESCRIPTION OF EVENT
A Notification of Unusual Event was declared at 2200 hours on September 15, 1999 in accordance with Emergency Plan Implementing Procedure (EPIP) 1.01, Emergency Manager Controlling Procedure; Attachment 1, Emergency Action Level Table, Tab L, Condition 8 (Hurricane force winds projected onsite within 12 hours). Notification to the Virginia Department of Emergency Management and local governments was performed in accordance with EPIP-2.01, Notification of State and Local Governments. The initial notification was started at 2206 hours. Follow-up reports were transmitted at intervals as agreed upon with the State.

Surry Power Station implemented its Hurricane Response Plan at 0900 hours on September 15, 1999. Prior to that time, actions had been implemented to prepare for forecasted high winds as directed by station procedures. Extensive reviews were performed to make sure that loose material that could be affected by high winds had been either removed or secured. Also, workers ensured that emergency power supplies were ready, if needed. Additional station personnel were directed to augment the staff for operations and recovery. The Corporate Hurricane Response Center was activated in support of the response plan. At 1906 hours on September 15, 1999, shutdown of Surry Unit 2 was initiated because utility Weather Center forecasters projected hurricane force winds would be experienced onsite on Thursday, September 16, 1999. Surry Unit 1 shutdown was initiated at 2040 hours on Wednesday, September 15, 1999.

At 2300 hours September 15, 1999, the revised forecast predicted that Hurricane Floyd would take a more easterly track and hurricane force winds were not projected onsite. Power reduction on both Units was stopped and reactor power was stabilized at reduced levels. It was decided to remain in the emergency classification and at reduced power until such time as the station was clear of the storm even though the original conditions for the emergency did not exist at that time. The storm continued to veer away from Surry throughout the night and next day. Hurricane force winds were never experienced on site.

The event was terminated at 1420 hours on September 16 and termination notifications were transmitted to the Virginia Department of Emergency Management and local jurisdictions at 1426 hours. The Hurricane Response Plan was exited at 1530 hours on September 16, 1999.

ATTACHMENT 7

(Page 1 of 1)

ATWS Mitigation System Actuation Circuitry (AMSAC) Functionality

1. **Surry [Commitments 3.2.24 and 3.2.25]**

   AMSAC cannot perform its intended function and should be considered nonfunctional when:

   - The Control Room “NORMAL - BYPASS” switch is not functional
   - Bypassed for maintenance (“NORMAL - BYPASS” switch is placed in “BYPASS”)
   - Either turbine load input signal to AMSAC out of service
   - More than one programmable logic controller is out of service
   - More than one rotary output isolator is lost
   - The Loop B or Loop C power supply is out of service **and** both Loop A power supplies are out of service
   - The Loop B and Loop C power supplies are out of service
   - Black Battery and/or AMSAC inverter are out of service
   - Powered from the alternate power source

   NOTE: One Loop A power supply can keep Loop A in service with the other Loop A supply out of service. AMSAC will be functional with a 1 out of 2 signal to trip the unit even if the Loop B or Loop C power supply is out of service.

   NOTE: The SER (an attachment to **Reference 3.1.90**) specifies that AMSAC must function upon loss of offsite power. When the Black Battery or the AMSAC inverter is removed from service for maintenance or testing, AMSAC can be powered by using an alternate source (i.e., turbine building MCC). While powered from the MCC, AMSAC will not function as designed if there is a loss of offsite power.

2. **North Anna [Commitment 3.2.19 and 3.2.25]**

   AMSAC cannot perform its intended function and should be considered nonfunctional when:

   - The Control Room “NORMAL - BYPASS” switch is not functional
   - Bypassed for maintenance (“NORMAL - BYPASS” switch is placed in “BYPASS”)
   - Either turbine load input signal to AMSAC out of service
   - More than one programmable logic controller is out of service
   - More than one safety related output isolation relay is not functional
   - TSC UPS and/or TSC Battery is placed out of service

   NOTE: The SER (an attachment to **Reference 3.1.90**) specifies that AMSAC must function upon a loss of offsite power.

Pearson NCS Test Sheet 100/W

Form No. 95677

Reorder Form No. 95677
1-800-367-6627
Fax 1-507-451-4513
valuebridge.ncspearson.com

KEY ID Ⓐ Ⓑ Ⓒ Ⓓ

SCORING & PRINTING OPTIONS: ◯ RESCORE ◯ MULTIPLE ANSWER SCORING ◯ CORRECT ANSWER ◯ MARK X ◯ TOTAL ONLY — MARK ONLY ONE

FEED IN THIS DIRECTION

Q	Ans	Q	Ans	Q	Ans	Q	Ans
1	A	26	C	51	D	76	D
2	B	27	C	52	D	77	A
3	C	28	C	53	A	78	B
4	C	29	B	54	D	79	B
5	D	30	A	55	A	80	A
6	B	31	A	56	C	81	C
7	B	32	C	57	B	82	A
8	B	33	B	58	B	83	C
9	A	34	D	59	B	84	B
10	D	35	B	60	B	85	C
11	B	36	B	61	A	86	B
12	C	37	D	62	C	87	C
13	D	38	B	63	C	88	B
14	C	39	B	64	D	89	B
15	D	40	C	65	B	90	A
16	C	41	C	66	A	91	B
17	D	42	B	67	A	92	C
18	A	43	B	68	D	93	D
19	D	44	A	69	A	94	A
20	C	45	D	70	A	95	C
21	C	46	D	71	D	96	D
22	B	47	A	72	B	97	B
23	B	48	B	73	B	98	D
24	A	49	B	74	B	99	B
25	C	50	D	75	B	100	D

Ro SRo

ANSWER KEY INFO. — # OF KEYS — ITEM COUNT

PERFORMANCE ASSESSMENT — % OF TOTAL SCORE (00 = 100%) — POINTS EARNED — EQUAL PT VALUE

Bar Code

NUMBER CORRECT
PERCENT CORRECT
ROSTER NUMBER
SCORE
RESCORE

PEARSON NCS

COMBINED POINTS EARNED
COMBINED PERCENT CORRECT
LETTER GRADE
SCORE
RESCORE

100 ITEM

STUDENT ID NUMBER

NAME SURRY 2010-301
SUBJECT ANSWER KEY
PERIOD DATE

MARKING INSTRUCTIONS
Use a No. 2 Pencil
Fill oval completely
Erase cleanly