Draft Submittal

NORTH ANNA JUNE EXAM
50-338 & 50-33912004-301

JUNE 17 - 25,2004

1. Senior Reactor Operator Written Exam

011A2.10 001

Unit 1 is currently at 7% power following a ramp down from 100% power due to main transformer problems. The following plant conditions exist:

- 1-MS-PT-1446, Turbine First Stage Pressure, is out of service with all associated bistables in trip
- 1-RC-PT-1459, Pressurizer Level Transmitter, has Bailed high
- Pressurizer level control *is* in position 459/460.

In accordance with the applicable procedure the crew will place _________

A?' 1-CH-FCV-1122 in manual and increase demand or the reactor will eventually trip on high pressurizer level.
5. letdown back in service or the reactor will eventually trip on high pressurizer level.
C. 1-CH-FCV-1122 in manual and decrease demand or the pressurizer PORV's will eventually open.
D. letdown back in service or the pressurizer PORV's will eventually open.

A. This is the correct answer. Charging will go *to* minimum and actual level will decrease to a point that letdown will isolate. At that point pressurizer level will increase slowly reaching 92%. This is an at power trip blocked by P-7 below 10% power. Because bistables for PT 446 are thrown the unit will never see P-7 and the reactor will trip on high pressurizer level of 92%. Actions per 1-AP-3 are to place 1-CH-FCV-1122 in manual and increase demand.

5. This is incorrect. The channel failed high so letdown will not isolate if the actions of 1-AP-3 are taken. **If** not then letdown will isolate leading to a reactor trip.

C. This is incorrect. Charging will go to minimum but the reactor will trip even though power is below P-7 because the bistables for PT-446 are thrown. This **does** not enable P-7 which blocks the pressurizer hi Bevel trip. Examinee could choose this answer if they don't realize bistables for PT-446 prevent P-7 from blocking the pressurizer hi level trip.

D This is incorrect. The candidate may not realize that the reactor will trip based on the failure of turbine first stage pressure. This will make them choose between the answers involving a PORV opening. Letdown will have to isolate to do this but **will** only happen with no operator action.

Ability to (a) predict the impacts of the following on the Pressurizer Level Control system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Failure of PZR level instrument - high

(CFR: 41.5 I 43.5 145.3 I 45.13)

Modified bank question 2275

References:
Objectives 18656, 8841, and 11996 from study guide on Pressurizer Control and Protection
1-AP-3 Loss of Vital Instrumentation
1-MOP-55.81 Turbine First Stage Pressure Instrumentation.
NCRODP-74-NA pages 39-42.

Level(RO/SRO):	SRO	**Tier:**	2
Group:	2	**Importance Rating:**	3.4/3.6
Type(Bank/Mod/New):	MOD	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

The following plant conditions exist.

The unit operating at 100% power

- PRESS LEVEL CHANNEL DEFEAT switch in the 459/460 position

Pressurizer level channel 1-RC-LT-1459 has just failed low

Select the response that correctly describes the unit's response to this failure

A. Letdown isolates, backup heaters de-energize, charging flow increases, actual tevel increases. and the reactor trips on high pressurizer level.

B. Letdown isolates, backup heaters turn on, charging flow decreases, actual level decreases, and the reactor trips on low pressurizer pressure.

C. Letdown temperature increases, backup heaters turn on when actual level increases above program, charging flow decreases, and the reactor trips on high pressurizer level.

D. Letdown temperature increases, backup heaters de-energize when actual level decreases below program, charging flow decreases, and the reactor trips on low pressurizer pressure.

Answer: A

Topic 2.2.11: PRZR Level Control System Failures 10656

2.2.1la. Objective

Explain how the Pressurizer Level Control System would respond to a level channel failure

2.2.11b. Content

NA-DW-5655D33, SH 11 may be used to explain.

1. The position of the PRZR level channel defeat switch determines how an individual PRZR level channel will affect the output of LC-459 and LC-460.
   1.1. If the failed channel was not selected, then there will be no affect on PRZR level.

2. Assuming a selected channel to LC-459 failed high:
   2.1. Charging flow is reduced to a minimum (approx. 25 gpm)
   2.2. Annunciator B-G6, PZR HIGH LEVEL - BU HTRS ON alarms
   2.3. PRZR decreases until a letdown isolation from LC-460 occurs.(closes 14608 and HCV-1200A, B, and C at 15% PRZR level)
   2.4. Annunciator B-G7, PZR LO LEV HTRS OFF - LETDWN ISOL alarms
   2.5. PRZR level increases until a high level trip occurs.

3. Assuming a selected channel to LC-459 failed low:
   3.1. Letdown isolates (closes 1460A and HCV-1200A, B, and C at 15% PRZR level)
   3.2. Annunciator B-G7, PZR LO LEV HTRS OFF - LETDWN ISOL alarms.
   3.3. Charging flow increases to maximum.
   3.4. PRZR level increases until a high level trip occurs.

4. Assuming a selected channel to L6-460 failed high:

4.1. Annunciator B-G8, PRZ HI LEVEL (69.5%) alarms

5. Assuming a selected channel to bC-460 failed low:

5.1. Letdown isolates (closes 1460B and HCV-1200A, B, and C at 15% PRZR level)

5.2. Annunciator B-G7, PZR LO LEV HTRS OFF - LETDWN ISOL alarms.

5.3. Pressurizer level slowly increases due to letdown isolation.

5.4. Charging flow is reduced to minimum due to increasing PRZR level.

5.5. PRZR level increases until a high level trip occurs.

2.3: Pressurizer Level Protection

Topic 2.3.1: PRZR Level Protection
8841

2.3.1a. Objective

List the following information associated with pressurizer level protection.

- High-level reactor trip setpoint
- Conditions that will cause the PRESSURIZER HIGH LEVEL CH I-II-III annunciator to alarm
- Interlock that will block the high-level reactor trip

2.3.1b. Content

1. High-level reactor trip setpoint is 92% level (2/3 channels).

2. PRESSURIZER HIGH LEVEL CH I-II-III annunciator alarm setpoint is 92% level (1/3 channels)

3. The high PRZR level reactor trip is blocked when the unit is < P-7 setpoint (2/2 first stage pressure < 10% and 3/4 PRNIS < 10%)

CONTROL SYSTEM FAILURE ANALYSIS

(NOTE: **ALL** EFFECTS ASSUME NO OPERATOR ACTION IS TAKEN.)

PRESSURIZER PRESSURE CONTROL SYSTEM

1444	
HIGH	EFFECT: PCV1455C and both spray valves go full open and all heaters will turn off. Pressure drops rapidly until the P-11 interlock (2/3 < 2000 psig) closes the PORV. Pressure continues to drop due to both sprays being open and the reactor trips on a low-pressure signal that is rate-compensated. A low pressure SI will also occur (1780 psig).
LOW	EFFECT: All heaters on full. Spray valves stay shut. Pressure slowly rises until PCV-1456 opens. Pressure cycles around BQRV setpoint (–2335 psig). Note that PCV-1455C is disabled.

1445	
HIGH	EFFECT: PCV-1456 opens and pressure drops rapidly. PORV will be shut by P-11 interlock (2000 psig). Pressure will cycle around interlock setpoint. Mote that the could overshoot and cause a plant trip ~~ low pressure.
LOW	EFFECT: None Note that PCV-1456 is disabled.

PRESSURIZER LEVEL CONTROL SYSTEM

Main Control Channel (Channel section 459 or 461)	
HIGH	EFFECT: Backup heaters turn on and the charging flow is reduced to minimum. Pressurizer level will decrease. At 45% level. letdown isolation occurs, heaters turn off, and the other control channel will generate an alarm. with no letdown. pressurizer level will then rise until a trip occurs on high level (92%).
LOW	EFFECT: Letdown isolation, heaters off, charging flow increases to maximum. Reactor trip will occur on high PRZR level (92%).

Secondary Control Channel (460 or 461)	
HIGH	EFFECT: None. High level alarm will be generated (69.5%).
LOW	EFFECT: Letdown isolation occurs and the heaters turn off. Pressurizer level will rise slowly causing charging flow to be reduced to minimum (–25 gpm). Reactor will eventually trip on high **PRZR** level (92%).

T_{ave} Circuit Failure	
HIGH	EFFECT: Level reference limited to 100% programmed level. If level is less that program, it will be brought to 100% program level (64.5%).
LOW	EFFECT: Charging flow is reduced until level decreases to 0%-programmed level (28.4%).

pressure signal which is used in calculating the overtemperature AT setpoint.

6. Input to comparator PC-1455E. If pressure falls to 1780 psig. the camparator sends an output to the train A and B low pressure safety injection circuits in the RPS. If two of the three channels provide the signal, safety injection is initiated, and a reactor trip occurs. The comparator annunciates the PRESSURIZER LOW-LOW PRESSURE SI CH I-II-III alarm (window 1B-E4). This is a common alarm for all three channels and has reflash capability.

Comparator PC-1455A provides one output not provided by PC-1456A or PC-1457A. Comparator PC-1455A annunciates the PRESSURIZER HIGH PRESSURE alarm (window 1B-E7) at 2310 psig. Another difference between the channels is that the other two channels use comparators PC-14568 and PC-1457D to accomplish the same result as PC-1455E.

The input to low pressurs reactor trip comparator (PC-1455C) is modified by a rate compensation device (PM-1455A) which accounts for the rate at which plant pressure is approaching the comparator setpoint. If ?he pressure is falling rapidly, the rate compensator alters the signal so that the comparator trip occurs at a pressure higher than its actual setpoint. This anticipatory feature could cause the low pressure reactor trip to occur at a pressure above 1870 psig.

Pressurizer Level Control Subsystem

The level control and level protection subsystems use the same three level transmitters (LT-1459, -1460, -1461) for input. The level protection subsystem is discussed later in this section. Each level transmitter provides a data log input to the computer, as well as the following MCB indication:

1. LT-1459 to LI-1459A
2. LT-1460 to LI-1460
3. LT-1461 to LI-1461

Additionally LT-1459 provides level indication at the Auxiliary Shutdown Panel on LI-1459B and C.

The three transmitters are used to provide input to two control channels (LC-453 and LC-460), with a control switch determining which transmitter inputs to which channel (see Figure 74-8-NA). This arrangement provides a redundant capability to the control subsystem.

The control subsystem inputs, from the level transmitters (see Figure 74-8-NA), are controlled by the PRESSURIZER LEVEL CHANNEL DEFEAT handswitch (1/LM-1459) on the benchboard. The three position (1461-1460, 1459-1460, 1459-1461) handswitch allows the operator to align the level transmitter output to specific control channels. The first number of a handswitch position indicates which transminer is aligned to channel LC-1459; the second number indicates which transmitter is aligned to channel LC-1460. As evidenced by the handswitch positions. transmitter LT-1459 can only input to channel LC-1459, LT-1460 can only input to channel LC-1460, but LT-1461 can input to either channel. Level transmitter LT-1461 serves as a backup to both channels. The fallowing discussion assumes the handswitch is in the 1459-1460 position.

Level Control Channel LC-1459. The LC-1459 control channel uses an indication of actual pressurizer level from LT-1459 as well as a setpoint signal supplied from the $\Delta T/T_{avg}$ control subsystem. The setpoint signal establishes the programmed pressurizer level as a function of T_{avg}. At a T_{avg} = 547°F the programmed level signal setpoint is 25.8 percent. The programmed level setpoint increases linearly to a maximum setpoint of 64.5 percent for a T_{avg} = 580.8°F. Below a T_{avg} = 547°F, the programmed setpoint remains at 25.8 percent; if T_{avg} exceeds its maximum allowed value of 580.8°F, the programmed setpoint remains at 64.5 percent. The setpoint value is supplied to pen 1 of level recorder LR-1459 (LR-2459).

The output of transmitter LT-1459 and the programmed setpoint are compared in the following:

1. Comparator LC-1459D which energizes the backup heaters if actual level rises more than 5 percent above the programmed setpoint. This feature anticipates the pressure drop caused by the excess insurge of relatively cold loop coolant. The comparator also annunciates the PRESSURIZER HIGH LEVEL BACKUP HEATERS ON alarm (window 1B-G6).
2. Comparator LC-1459E which annunciates the PRESSURIZER LOW LEVEL alarm (window 1B-8) if actual level falls more than 5 percent below the programmed level.
3. Comparator LC-1459F, discussed later in this subsection.

Comparator LC-1459C compares the actual level signal to an internal setpoint which, when level falls to 15 percent, initiates the following:

1. Deenergizes all the heaters to ensure that they do not operate when they become uncovered due to the low liquid level in the pressurizer. The signal opens the control group breaker, and opens the backup groups main line contactors.
2. Shuts CVCS letdown isolation valves LCV-1460A, HCV-1200A, HCV-1200B, and HCV-1200C which secures letdown flow from the RC System in order to restore pressurizer level.
3. Annunciates the PRESSURIZER LOW LEVEL HEATERS OFF LETDOWN ISOLATION alarm (window 1B-G7).

bevel controller LC-1459F compares the actual level signal and the setpoint signal in order to produce a level demand signal. The controller is a PID-type controller with an associated M/A station (LC-1459G). The station is a standard M/A station except the setpoint dial is not functional. The operator can manually control the controller and produce a level demand signal. As with other controllers, the manual and automatic signals track each other for bumpless transfer.

The level demand signal is supplied to flow controller FC-1122A. This controller-M/A station operates exactly as the controller described above. The controller compares a flow signal from flow transducer FT-1122, in the normal charging path to the RC System, against the level error signal from LC-1459F. The resulting output signal is sent to control the position of charging flow control valve FCV-1122.

If pressurizer level is falling below the programmed setpoint, the level error grows larger. In this case, the larger signal tends to open FCV-1122 in order to increase the charging flow to the RC System and restore pressurizer level. The input from FT-1122 is used to provide feedback to the circuit to reduce the lag time associated with the Contrd. The feedback also ensures that the valve does not oscillate excessively or hunt about the setpoint. If pressurizer level is rising above the setpoint, the circuits acts to shut the valve and decrease charging flow to the RC System. In either case, the signal is applied to an U P converter which controls the operating air to the valve operator, thereby achieving the desired valve position.

The circuit can be manually controlled at the MCB using the manual raise and lower pushbuttons at the PRESSURIZER LEVEL CONTROL M/A station, or locally at the Auxiliary Shutdown Panel. A LOCAL/REMOTE handswitch at the panel operates contacts which allow passage of the control signal from either FC-1122A or from the manual station at the panel. In the LOCAL position manual controller HIC-1122 is used to adjust the position of FCV-1122 in order tc control pressurizer level. Level indication is provided by LI-1459B and C. If iocal control is taken the CHARGING LINE FLOW LOCAL OPERATION alarm (window 1C-B8) annunciates.

Flow transmitter FT-1122 provides indication of charging flow at the MCB (FI-1122A) and at the Auxiliary Shutdown Panel (FI-1122B). The transmitter also provides a data log input to the computer as well as annunciates the CHARGING PUMP TO REGENERATIVE HEAT EXCHANGER HIGH-LOW FLOW alarm (window 1C-C5).

Level recorder LR-1459 continuously records the level setpoint from the $\Delta T/T_{avg}$ control subsystem on pen 2. Pen 1 is used to record actual pressurizer level as measured by one of the level transmitters. The LEVEL RECORDER SELECTOR SWITCH (1/LR-459) on the vertical board allows the operator to select which transmitter inputs to the recorder.

Level Control Channel LC-1460. Control channel LC-1460 uses the input from LT-1460 (in this example) to provide input to a dual comparator (LC-1460C). The comparator is used for the fcllowing:

1. Deenergising all the heaters to ensure that they do not operate when they become uncovered due to the low water level in the pressurizer. The signal opens the control group breakers, and opens the backup

groups main line contactors. This action is redundant to the action initiated by LC-1459C.

2. Shutting CVCS letdown isolation valves LCV-1460B, HCV-1200A, HCV-1200B and HCV-1200C which secures letdown flow from the RC System in order to restore pressurizer level.
3. Annunciating the PRESSURIZER LOW LEVEL HEATERS OFF - LETDOWN ISOLATION alarm (window 1B-G7).
4. Annunciating the PRESSURIZER HIGH LEVEL alarm (window 1B-G8) at 69.5 percent.

The first three functions are redundant to those accomplished by LC-1459C and annunciate the same alarm window.

Pressurizer Level Protection Subsystem

The three level transmitters (LT-1459, -1460, and -1461) also provide input to the level protection subsystem. Each channel provides an input to a level comparator (e.g., LS-1459A) which provides a high pressurizer level trip signal to the RPS. if two of the three protection channels provide the trip signals. a reactor trip occurs (provided permissive interlock P-7 is satisfied). If one of the channels provides the signal the PRESSURIZER HIGH LEVEL CH I-II-III alarm annunciates (window 1B-F6). At one time the channels also provided input to a low level-low pressure safety injection actuation circuit. Design changes in reactor protection have replaced this feature with the Low-Low pressure safety injection signal previously discussed.

Dominion NORTH ANNA POWER STATION	PROCEDURE NO: 1-MOP-55.81
	REVISION NO: 6
PROCEDURE TYPE: MAINTENANCE OPERATING	UNIT NO: 1

PROCEDURE TITLE

TURBINE FIRST STAGE PRESSURE INSTRUMENTATION

				EOP AP		

REVISION **SUMMARY:**

Converted to FrameMaker using Tempalte Rev. *030.*

- Incorporated DCP 01-007, Phase *2* PCS Installation and P-250 Removal – Unit 1:
  - Added DCP **to** references as Step **2.3.5.**
  - Added conditional in Step 4.3, and added '(P-250) or actuates (Phase *2* PCS)' in **Steps** 5.2.12 and 5.3.12.
  - Added Phase 2 PCS Computer Point descriptions and designated original point descriptions for P-250 in substeps of Steps 5.2.12 and 5.3.12 for Phase 2 PCS.
- Made **ITS** changes permanent in Section 2.2 **and** in Steps 4.1, 4.3, **5.1.1.a,** 5.1.1.b, 5.1.2.a, **5.1.2.d Note,** and Cautions **before** Steps 5.2.3, 5.2.4, 5.3.3, and 5.3.4. Removed ITS from Review **Bar.**
- Deleted **Step** 2.3.4, 1-AP-3.11, Loss of Vital Instrumentation Turbine First Stage Pressure, **which** was replaced by this procedure. No need to reference a deleted procedure.
- Deleted VPAP-2802 Step Number in Step 4.2 since this step number could change and is not necessary to specify.

PROCEDURE **USED**:	☐ Entirely	☐ Partially	**Note: If used** partially, note reasons in remarks.
PROBLEMS ENCOUNTERED:	☐ **NO**	☐ **YES**	**Note:** If **YES, note** problems in remarks.

REMARKS: ______________________________

(Use **back** ***for*** additional remarks.)

SHIFT SUPERVISOR:	DATE:

TABLE OF CONTENTS

1.0 PURPOSE

To provide instructions for placing the Turbine First Stage Pressure instrument channels in TEST.

2.0 REFERENCES

2.1 Source Documents

2.1.1 DCP 88-01, Changed Unit I annunciators

2.2 Technical Specifications

2.2.1 Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Functions 18b and 18e

2.2.2 Tech Spec 3.3.2, Table 3.3.2-1, Engineered Safety Feature Actuation System, Functions 1f, 1g, 4d, and 4e

2.2.3 Tech Spec 3.0.3

2.2.4 VPAP-2802 (AMSAC)

2.3 Technical References

2.3.1 Westinghouse SSPS Tech Manual

2.3.2 Westinghouse Process Instrumentation Manual and Prints

2.3.3 Instrument Department PTs

2.3.4 1-AP-3, Loss of Vital Instrumentation

2.3.5 DCP 01-007, Phase 2 PCS Installation and P-250 Removal - Unit 1

2.4 Commitment Documents

2.4.1 CTS-02-97-2196-003 Revise procedures to include WAF'-2802 actions for AMSAC

2.4.2 CTS Assignment 02-99-1801-003, Tech Spec Change 290

3.0 INITIAL CONDITIONS

The applicable steps of 1-AP-3, Loss of Vital Instrumentation, have been completed.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Tech Spec 3.3.2, Engineered Safety Feature Actuation System, Condition "D":

With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

4.1.1 The inoperable channel is placed in the tripped condition within 72 hours.

4.1.2 The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 3.3.2, Engineered Safety Feature Actuation System, Condition "D".

4.2 VPAP-2802: IF the ATWS Mitigation System Actuation Circuit (AMSAC) is out of service for 30 consecutive days, THEN Licensing Dept. shall prepare a special report for review by SNSOC, approval by Site Vice President, and submittal to the NRC within the next 30 days. See VPAP-2802 Attachment 14 for operability criteria.

4.3 IF the indicated alarm and printout (printout only if P-250 not removed by DCP 01-007) is not actuated for a trip switch, THEN verify the correct switch was placed in TEST. IF the correct switch was in TEST and no actuations occurred, THEN the operability of both trains of the Solid State Protection System is questionable and the unit must be placed in the mode required by Tech Spec 3.0.3.

4.4 **Placing a Turbine First Stage Pressure Channel in TEST will cause the AMSAC System to become inoperable because the 2/2 coincidence cannot be obtained. A separate Action Statement will be needed for the AMSAC System.**

4.5 **IF the Unit is in Mode 4, 5, or 6, THEN a Turbine First Stage Pressure Instrument MAY be placed in Test as desired by the Shift Supervisor.**

.nit **Verif**

5.0 INSTRUCTIONS

5.1 P-13 Permissive Check

5.1.1 IF Reactor Power is greater than 10percent, THEN do the following:

a. Verify the following annunciator status in accordance with Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System **Instrumentation,** Functions 18b and 18e:

- **Annunciator P-F2,** P-13 PERM TURB PWR IMP PRESS < 10%, is NOT LIT
- **At** least one of the following annunciators are LIT:
  - L-F3, TURB PWR IMP PRESS INTLK >10% P-13 CHNL III
  - L-F4, TURB PWR IMP PRESS INTLK >10% P-13 CHNL IV

b. IF the conditions *of* Step 5.1.1.a cannot be met, THEN enter **Tech Spec** 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Condition "R".

c. IF the conditions of Step 5.1.1.a is met, THEN complete Section 5.2 or 5.3 as applicable within 72 hours.

5.1.2 IF Reactor Power is less than 10 percent, THEN do the following:

a. Verify the following annunciator status in accordance with Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Functions 18b and 18e:

- Annunciator P-F2, P-13 PERM TURB PWR IMP PRESS < 10%, is LIT
- BOTH of the following annunciators are NOT LIT:
  - L-F3, TURB PWR IMP PRESS INTLK >10% P-13 CHNL III
  - L-F4, TURB PWR IMP PRESS INTLK >10% P-13 CHNL IV

b. IF the conditions of Step 5.1.2.a cannot be met, THEN enter Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Condition "R".

NOTE: WHEN Fist Stage Pressure is placed in TEST with Reactor Power less than 10 percent, THEN P-E3 will change state and restore AT Power Trips.

NOTE Reducing Reactor Power to less than 5 Percent (Mode 2) will prevent entering Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Condition "R", when First Stage Pressure is placed in TEST. P-13 is Mode 1 applicable in accordance with Tech Spec 3.3.1, Table 3.3.1-1, Reactor Trip System Instrumentation, Functions 18b and 18e.

c. IF the conditions of Step 5.2.2.a is met AND Unit is in Mode 1 less than 10 percent, THEN reduce Reactor Power to less than 5 Percent (Mode 2), before continuing with this procedure.

d. WHEN Reactor Power is less than 5 percent, THEN complete Section 5.2 or 5.3 as applicable within 72 hours.

5.2 Placing Turbine First Stage Pressure Channel III (P-1446) in Test

_____ 5.2.1 Verify Initial Conditions are satisfied.

5.2.2 Review Precautions and Limitations.

CAUTION

IF power is greater than 10 percent AND one of the coincident channel annunciators is LIT, THEN the following apply:

- Placing this channel in TEST could trip the Unit.
- IF placing this channel in TEST would cause a unit trip, THEN comply with the following:
- The channel must NOT be placed in TEST.
- The unit must be placed in the mode required by Tech Spec 3.0.3.

5.2.3 IF in Mode 1 with power greater than 10 percent AND the At Power Trips are not Mocked, THEN verify coincident channels are not tripped by verifying the following annunciators are NOT LIT:

_____ • Panel "L" F-5, RC LOOP 1A LO-LO TAVG CHNL I

• Panel "L" F-7, RC LOOP 1C LO-LO TAVG CHNL III

_____ • Panel "N" H-4, STM LINE 1A LO PRESS CHNL II

_____ • Panel "N" H-7, STM LINE 1B LO PRESS CHNL III

_____ • Panel "N" H-8, STM LINE 1C LO PRESS CHNL IV

(Step 5.2.3 Continues on the Next Page)

(Step 5.2.3 Continued)

______ • Panel "N" E-8, STM LINE 1A HI FLOW CHNL IV

______ • Panel "N" F-8, STM LINE 1B HI FLOW CHNL IV

______ • Panel "N" G-8, STM LINE 1C HI FLOW CHNL IV

CAUTION

IF one of the annunciators listed below is LIT AND placing this channel in TEST will trip the unit, THEN the unit must be placed in the mode required by Tech Spec 3.0.3.

NOTE WHEN First Stage Pressure is placed in TEST, THEN ail At Power Trips are net blocked.

5.2.4 IF in Mode 1, 2, or 3 with power less than 10 percent AND the At Power Trips are blocked, THEN verify coincident channels are not tripped by verifying the following annunciators are NOT LIT:

- Panel "L" A-5, RC LOOP 1A LO FLOW CHNL I
- Panel "L" A-6, RC LOOP 1A LO FLOW CHNL II
- ______ Panel "L" A-7, RC LOOP 1A LO FLOW CHNL III
- Panel "L" A-8, RCP-SS BUS 1sA UV
- Panel "L" B-5, RC LOOP 1B LO FLOW CWNL I

(Step 5.2.4 Continues on the Next Page)

(Step 5.2.4 Continued)

______ • Panel "L" B-6, RC LOOP 1B LO FLOW CHNL II

______ • Panel "L" B-7, RC LOOP 1B LO FLOW CHNL III

______ • Panel "L" B-8, RCP-SS BUS 1B UV

______ • Panel "L" C-5, RC LOOP 1C LO FLOW CHNL I

______ • Panel "L" C-6, RC LOOP 1C LO FLOW CHNL II

______ • Panel "L" C-7, RC LOOP 1C LO FLOW CHNL III

______ • Panel "L" C-8, RCP-SS BUS 1C UV

______ • Panel "L" D-5, RCP 1A BKR OPEN CHNL I

______ • Panel "L" D-6, RCP 1B BKR OPEN CHNL II

______ • Panel "L" D-7, RCP 1C BKR OPEN CHNL III

______ • Panel "L" F-5, RC LOOP 1A LO-LO TAVG CHNL I

______ • Panel "L" F-6, RC LOOP 1B LO-LO TAVG CHNL II

(Step 5.2.4 Continues on the Next Page)

(Step 5.2.4 Continued)

- ______ **Panel "L" F-7, RC LOOP 1C LO-LO TAVG CHNL III**
- **Panel "L" F-8, RCP-SS BUS 1A UF**
- **Panel "L" G-5,** PRZR **HI LVL CHNL I**
- ______ **Panel "L" G-6, PRZR HI LVL CHNL II**
- ______ **Panel "L" G-7, PRZR HI LVL** CHNL **III**
- ______ **Panel "L" G-8, RCP-SS BUS 1B UF**
- ______ **Panel "L" H-8, RCP-SS BUS 1C UF**
- ______ **Panel** " M **H-5, PRZR LO PRESS CHNL I**
- Panel **"M" H-6, PRZR** LO **PRESS CHNL II**
- ______ **Panel** " M **H-7, PRZR LO PRESS CHNL III**
- ______ **Panel** "N" H-6, **STM LINE 1A LO PRESS CHNL 11**
- ______ **Panel "N'H-7, STM** LINE **1B LO PRESS** CHNL **III**
- ______ **Panel** "N" H-8, STM **LINE 1C LO PRESS CHNL IV**

______ 5.2.5 **Ensure that the selector switch listed below is in the position indicated, using the following guidelines as required**

- **IF the CRO must reposition a switch to a required position, THEN ensure the associated controller is in MANUAL before the switch is repositioned**
- **After the switch has been repositioned, THEN the associated controller may be placed in AUTO or remain in MANUAL as required**

Switch	Required Position	Associated Controller
FIRST STAGE PRESSURE: SIGNAL SELECTOR 1/1 PM-446	PQY-447	FC-1478, FC-1488, FC-1498, FC-1479, FC-1489, FC-1499, Rod Control

...

NOTE: The Steam Flow and Feedwater Flow selector switches could be placed to Channel IV for reliability. The Channel III Stem Flow and Feedwater Flow signals will *still* be functional with Channel III Turbine First Stage Pressure in TEST.

5.2.6 IF desired, THEN ensure that the selector switches listed below are in the positions indicated, using the following guidelines as required:

- IF the CRO must reposition a switch to a required position, THEN ensure the associated controller is in **MANUAL** before the switch is repositioned.
- After the switch has been repositioned, THEN the associated controller **may** be **placed** in AUTO or remain in MANUAL as required.

Switch	Required Position	Associated Controller
FD WTR FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-478B	FM-476A	FC-478F, 1-FW-FCV-1478
STEAM FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-478A	FM-475A	FC-478F, 1-FW-FCV-1478
FD WTR FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-488B	FM-486A	FC-488F, 1-FW-FCV-1488
STEAM FLOW SIGNAL SEE CONTROLLER AND RECORDER 1/1 FM-488A	FM-485A	FC-488F, 1-FW-FCV-1488
FD WTR FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-498B	FM-496A	FC-498F, 1-FW-FCV-1498
STEAM FLOW SIGNAL SEI. CONTROLLER AND RECORDER 1/1 FM-498A	FM-495A	FC-498F, 1-FW-FCV-1498

5.2.7 Place the **AMSAC** selector switch in BYPASS.

5.2.8 Record AMSAC into the Action Statement **Status Log.**

______ 5.2.9 Place the **Steam** Dumps into the STEAM **PRESSURE** mode.

______ 5.2.10 Using Attachment 1, go to the Instrument Rack Room and locate Channel III Protection Cabinet 3.

______ 5.2.11 Unlock and open Channel III Protection Cabinet 3 and verify that annunciator Panel "P" G-7, PCC CAB III VIOLATED **DOOR** OPEN, is LIT.

NOTE Attachment 2, Process Typical 2-Bay Cabinet, and Attachment 3, Process Typical 3-Bay Cabinet, will aid in identifying the correct **card and** Attachment 4, Process Typical Channel Test Card, will aid in identifying the correct switch on the card.

NOTE: **The** Channel Test **Status** light on the top **edge** of the Channel Test Card comes on **only** when the loop is properly in TEST with the master test switch in NORMAL. Attachment **4** will aid in identifying the Channel Test Status light.

5.2.12 Place the following Bistable (BS) switches in TEST and verify the associated annunciators are LIT and computer alarm prints out (P-250) or actuates (Phase 2 PCS):

a. C3-741, **BS-1** (1-MS-FTS-1474-1)

SV

- Panel "N" E-7, STM LINE 1A HI FLOW CHNL III

______

- Panel "F" G-2, MS LINE HI FLOW CH III-IV

- Computet Point F0406D, STM LINE 1 HI F 1 SI **PART RE** (P-250), OR STM GEN A HI STEAM F CH III (Phase 2 PCS)

SV

b. C3-746, RS-1 (1-MS-FTS-1484-1)

- Panel "N" F-7, STM LINE 1B MI FLOW CHNL III

- Computer Point F0426D, STM LINE 2 HI F 1 SI PA T RE (P-250), OR STM GEN B HI STEAM F CH III (Phase 2 PCS)

SV

c. C3-748, BS-1 (1-MS-FTS-1494-1)

- Panel "N" G-7, STM LINE 1C HI FLOW CHNI, III

- Computer Point F0446D, STM LINE 3 HI F 1 SI PART RE (P-250), OR STM GBN C HI STEAM F CH III (Phase 2 PCS)

SV

d. C3-736, BS-1 (1-MS-PTS-1446-A)

- Panel "L" F-3, TURB PWR IMP PRESS INTLK > 10%P-13 CHNL III

- Computer Point Y0001D, TB PWR 1RE TR PART PERM (P-250), OR IMPULSE P CH III <10%: PART P13 (Phase 2 PCS)

5.2.13 Close and lock Channel III Protection Cabinet 3 and verify that annunciator Panel "P" G-7, PCC CAB III VIOLATED DOOR OPEN, is NOT LIT.

5.2.14 Record the failed instrument channel in the Action Statement Status Log.

5.2.15 Notify the Instrument Department that Turbine First Stage Pressure Channel III (F-1446) failed and has been placed in trip.

Completed by: ____________ ____________ **Date:** ________

5.3 Placing Turbine First Stage Pressure Channel IV (P-1447) in Test

______ **5.3.1** Verify Initial Conditions are satisfied.

______ **5.3.2** Review Precautions and Limitations.

CAUTION

IF power is greater than 10 percent AND one of the coincident channel annunciators is LIT, THEN the following apply:

- Placing this channel in TEST could trip the unit.
- IF placing this channel in TEST would cause a *unit* trip, THEN comply with the following:
- The channel must NOT be placed in TEST.
- The unit must be placed in the mode required by Tech Spec 3.0.3.

5.3.3 IF in Mode 1 with power greater than 10 percent AND the At Power Trips are not blocked, THEN verify coincident channels are not tripped by verifying the following annunciators are NOT LIT:

- Panel "L" F-5, RC LOOP 1A LO-LO TAVG CHNL I
- Panel "L" F-6, RC LOOP 1B LO-LO TAVG CHNL II
- ______ Panel "L" F-7, RC LOOP 1C LO-LO TAVG CHNL III
- ______ Panel "N" H-6, STM LINE 1A LO PRESS CHNL II
- ______ Panel "N" H-9, STM LINE 1B LO PRESS CHNL III

(Step 5.3.3 Continues on the Next Page)

(Step 5.3.3 Continued)

______ • Panel "N" H-8, STM LINE 1C EO PRESS CHNL IV

______ • Panel "N" E-7, STM LINE 1A HI FLOW CHNL III

• Panel "N" F-7, STM LINE 1B HI FLOW CHNL III

______ • Panel " N G-7, STM LINE IC HI FLOW CHNL III

CAUTION

IF one of the annunciators listed below is LIT AND placing this channel in TEST will trip the unit, THEN the unit must be placed in the mode required by Tech Spec 3.0.3.

NOTE WHEN First Stage Pressure is placed in TRIP, THEN all At Power Trips are not blocked.

5.3.4 IF in Mode 1, 2, or 3 with power less than 10 percent AND the At Power Trips are blocked, THEN verify coincident channels are not tripped by verifying the following annunciators are NOT LIT

______ • Panel "E" A-5, RC LOOP 1A LO &LOWCHNL I

______ • Panel "L" A-6, RC LOOP 1A LO FLOW CHNL II

______ • Panel "L" A-7, RC M O P 1A LO FLOW CHNL III

______ • Panel "L" A-8, RCP-SS BUS 1A UV

(Step 5.3.4 Continues on the Next Page)

(Step 5.3.4 Continued)

_______ • Panel "L" B-5, RC LOOP 1B LO FLOW CHNL I

• Panel "L" B-6, RC LOOP 1B LO FLOW CHNL II

_______ • Panel "L" E-7, RC LOOP 1B LO FLOW CHIC III

_______ • Panel "L" B-8, RCP-SS BUS 1B UV

_______ • Panel "L" C-5, RC LOOP 1C LO FLOW CHNL I

_______ • Panel "L" C-6, RC LOOP 1C LO FLOW CHNL II

• Panel "L" C-7, RC LOOP 1C LO FLOW CHNL III

_______ • Panel "L" C-8, RCP-SS BUS 1C UV

_______ • Panel "L" D-5, RCP 1A RKR OPEN CHNL I

_______ • Panel "L" D-6. RCP 1B BKR OPEN CHNL II

_______ • Panel "L" D-7, RCP 1C BKR OPEN CHNL III

_______ • Panel "L" F-5, RC LOOP 1A LO-LO TAVG CHNL I

(Step 5.3.4 Continues on the Next Page)

(Step 5.3.4 Continued)

_______ • Panel "L" F-6, RC LOOP 1B LO-LO TAVG CHNL II

_______ • Panel "L" F-7, RC LOOP 1C LO-LO TAVG CHNL III

_______ • Panel "L" F-8, RCP-SS BUS 1A UF

_______ • Panel "L" G-5, PRZR HI LVL CHNL I

_______ • Panel "L" G-6, PRZR HI LVL CHNL II

_______ • Panel "L" G-7, PRZR HI LVL CHNL III

_______ • Panel "L" G-8, RCP-SS BUS 1B UF

_______ • Panel "L" H-8, RCP-SS BUS 1C UF

_______ • Panel "M" H-5, PRZR LO PRESS CHNL I

_______ • Panel " M H-6, PRZR LO PRESS CHNL II

_______ • Panel "M" N-7, PRZR LO PRESS CHNL III

_______ • Panel "N" H-6, STM LINE 1A LO PRESS CHNL II

(Step 5.3.4 Continues on the Next Page)

(Step 5.3.4 Continued)

______ • **Panel "N" H-7, STM LINE 1B LO PRESS CHNL III**

______ • **Panel " N H-8, STM LINE 1C LO PRESS CHNL IV**

5.3.5 **Ensure that the selector switch listed below is in the position indicated, using the following guidelines as required:**

- **IF the CRO must reposition a switch to a required position, THEN ensure the associated controller is in MANUAL before the switch is repositioned.**
- **After the switch has been repositioned, THEN the associated controller may be placed in AUTO or remain in MANUAL as required.**

Switch	Required Position	Associated Controller
FIRST STAGE PRESSURE SIGNAL SELECTOR 1/1 PM-446	PQY-446	FC-1478, FC-1488, FC-1498, FC-1479, FC-1489, FC-1499, Rod Control

NOTE: The Steam Flow and Feedwater Flow selector switches could be placed to Channel III for reliability. The Channel IV Steam Flow and Feedwater Flow signals will still be functional with Channel IV Turbine First Stage Pressure in TEST.

5.3.6 IF desired, THEN ensure that the selector switches listed below are in the positions indicated, using the following guidelines as required:

- IF the CRO must reposition a switch to a required position, THEN ensure the associated controller is in MANUAL before the switch is repositioned.
- After the switch has been repositioned, THEN the associated controller may be placed in AUTO or remain in MANUAL as required.

switch	Required Position	Associated Controller
FD WTR FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-478B	FM-477A	FC-478F, 1-FW-FCV-1478
STEAM FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-478A	FM-474A	FC-478F, 1-FW-FCV-1478
FD WTR FLOW SIGNAL, SEL CONTROLLER AND RECORDER 1/1 FM-488B	FM-487A	FC-488F, 1-FW-FCV-1488
STEAM FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-488A	FM-484A	FC-488F, 1-FW-FCV-1488
FD WTR FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-498B	FM-497A	FC-498F, 1-FW-FCV-1498
STEAM FLOW SIGNAL SEL CONTROLLER AND RECORDER 1/1 FM-498A	FM-494A	FC-498F, 1-FW-FCV-1498

5.3.7 Place the AMSAC selector switch in BYPASS.

5.3.8 Record AMSAC into the Action Statement Status Log.

____ 5.3.9 Place the Steam Dumps into the STEAM PRESSURE mode.

____ 5.3.10 Using Attachment 1, go to the Instrument Rack Room and locate Channel IV Protection Cabinet 4.

____ 5.3.11 Unlock and open Channel IV Protection Cabinet 4 and verify that annunciator Panel "P" G-8, PCC CAB IV VIOLATED DOOR OPEN, is LIT.

NOTE: Attachment 2, Process Typical 2-Bay Cabinet, and Attachment 3, Process Typical 3-Bay Cabinet, will aid in identifying the correct card and Attachment 4, Process Typical Channel Test Card will aid in identifying the correct switch on the card.

NOTE: The Channel Test Status Bight on the top edge of the Channel Test Card comes on only when the loop is properly in TEST with the master test switch in NORMAL. Attachment 4 will aid in identifying the Channel Test Status light.

5.3.12 Place the following Bistable (BS) switches in TEST and verify the associated annunciators are LIT and computer alarm prints out (P-250) OK actuates (Phase 2 PCS):

____ ____ SV a. C4-422, BS-1 (1-MS-FTS-1475-1)

____ • Panel " F G-2, MS LINE HI FLOW CH III-IV

____ • Panel "N" E-8, STM LINE 1A HI FLOW CHNL IV

____ • Computer Point F0407D, STM LINE 1 HI F 2 SI PART RE (P-250), OR STM GEN A HI STEAM F CH IV (Phase 2 PCS)

______ ______
SV

b. C4-427, BS-1 (1-MS-FTS-1485-1)

______ • Panel "F" G-2, MS LINE MI FLOW CH III-IV

______ • Panel "N" F-8, STM LINE 1B HI FLOW CHNL IV

______ • Computer Point F0427D, STM LINE 2 HI F 2 SI PART RE (P-250), OR STM GEN B HI STEAM F CH IV (Phase 2 PCS)

SV

c. C4-429, BS-1 (1-MS-FTS-1495-1)

______ • Panel "F" G-2, MS LINE HI FLOW CH III-IV

______ • Panel "N'G-8. STM LINE 1C HI FLOW CHNL IV

______ • Computer Point F0447D, STM LINE 3 HI F 2 SI PART RE (P-250), OR STM GEN C HI STEAM F CH IV (Phase 2 PCS)

______ ______
SV

d. C4-421, BS-1 (1-MS-PTS-1447E)

______ • Panel "L" F-4, TURB PWR IMP PRESS INTLK > 10% P-13 CHNL IV

______ • Panel "F" E-4, C-7 PERM STM DUMP ARMED FROM LOSS OF LOAD

______ • Computer Point Y0002D, TB PWR 2 RE TR PART PERM (P-250), OR IMPULSE P CH IV <10%: PART P13 (Phase 2 PCS)

______ 5.3.13 Close and lock Channel IV Protection Cabinet 4 and verify that annunciator Panel "P" G-8, PCC CAR IV VIOLATED DOOR OPEN, is NOT LIT.

_______ 5.3.14 Record the failed instrument channel in the Action Statement Status Log.

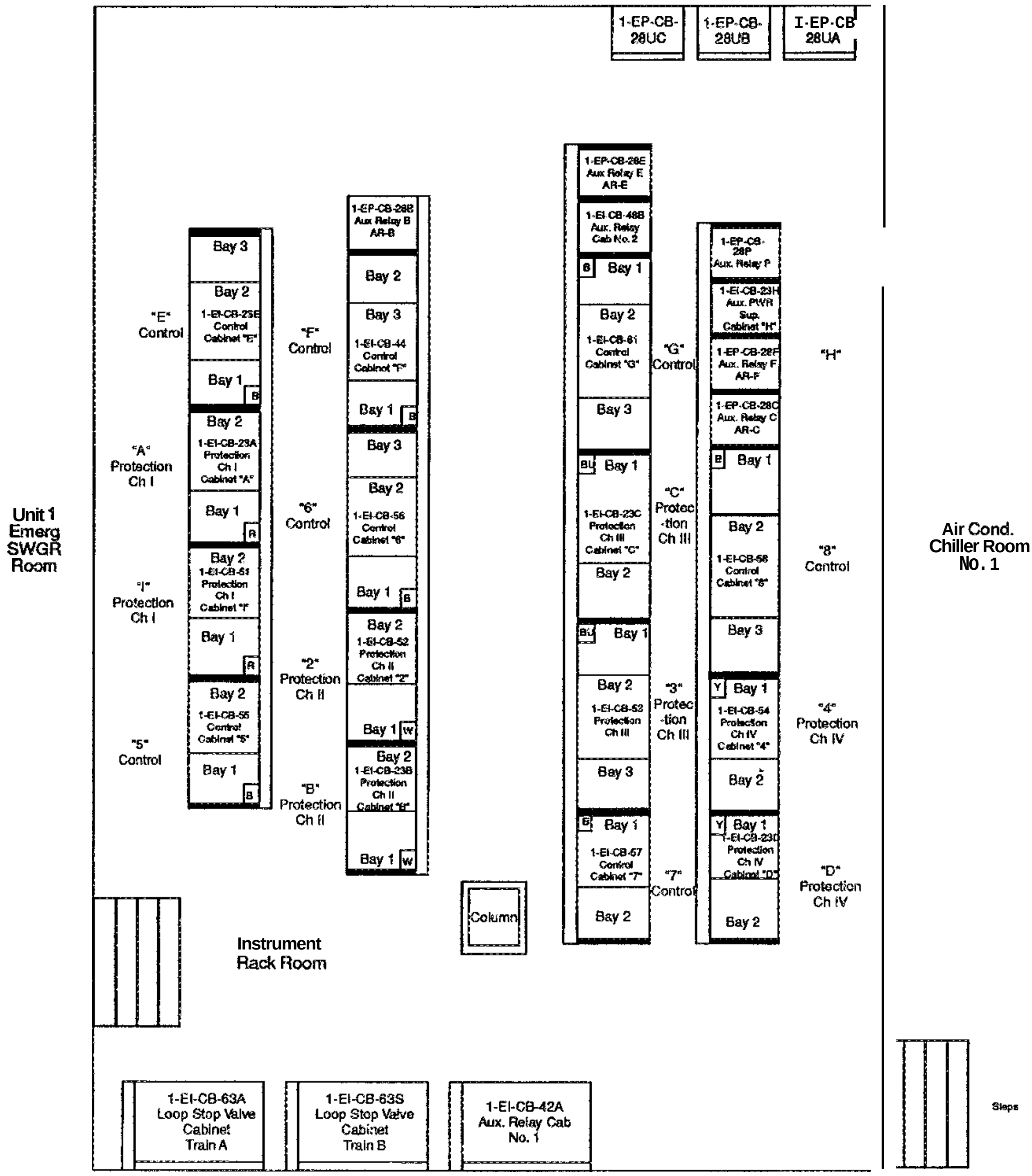
_______ 5.3.15 Notify the Instrument Department that Turbine First Stage Pressure Channel IV (F-1447) failed and has been placed in trip.

Completed by: ______________________________ **Date:** __________

(Page 1 of 1)

Attachment 1

Instrument Rack Room Cabinet Layout

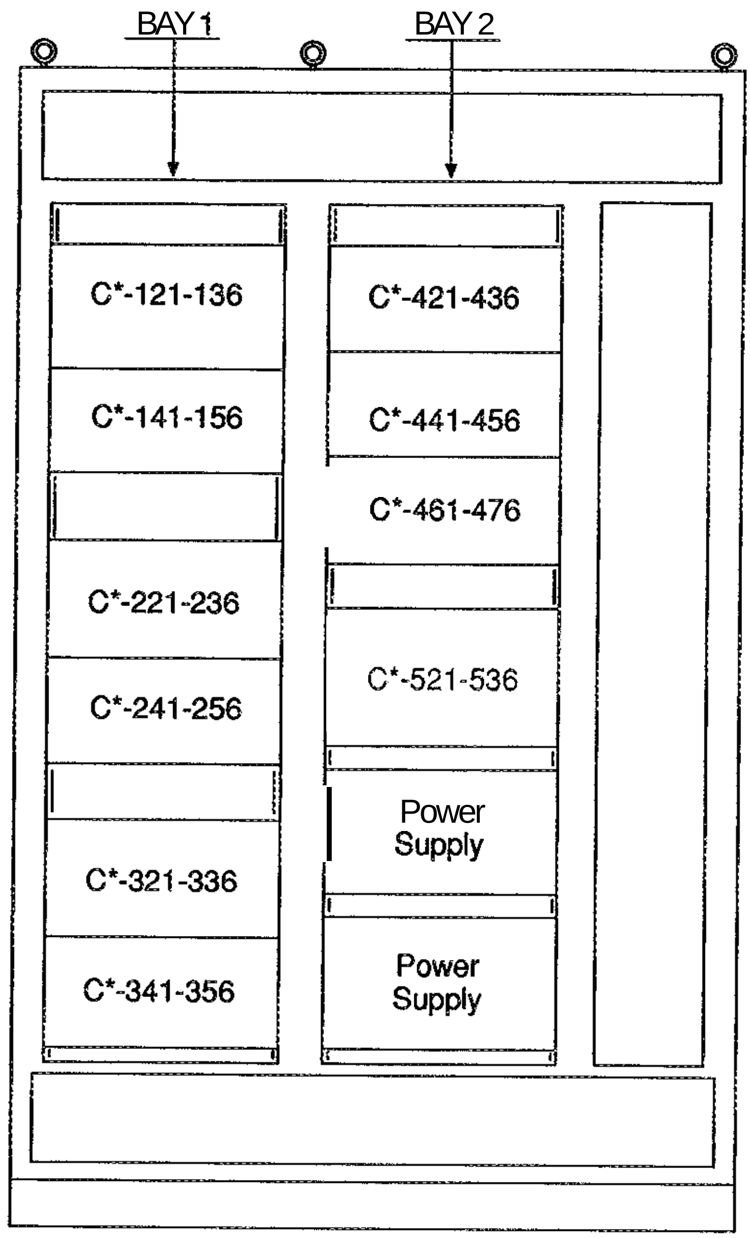


Graphics No: WT122A

INSTRUMENT **RACK** ROOM

(Page 1 of 1)

Attachment 2
Process Typical 2-Bay Cabinet



Graphics No: WT120

Location Code

Cabinet Number
Card Frame Number
Card Slot (Position in Card Frame)

C5 121

Note:

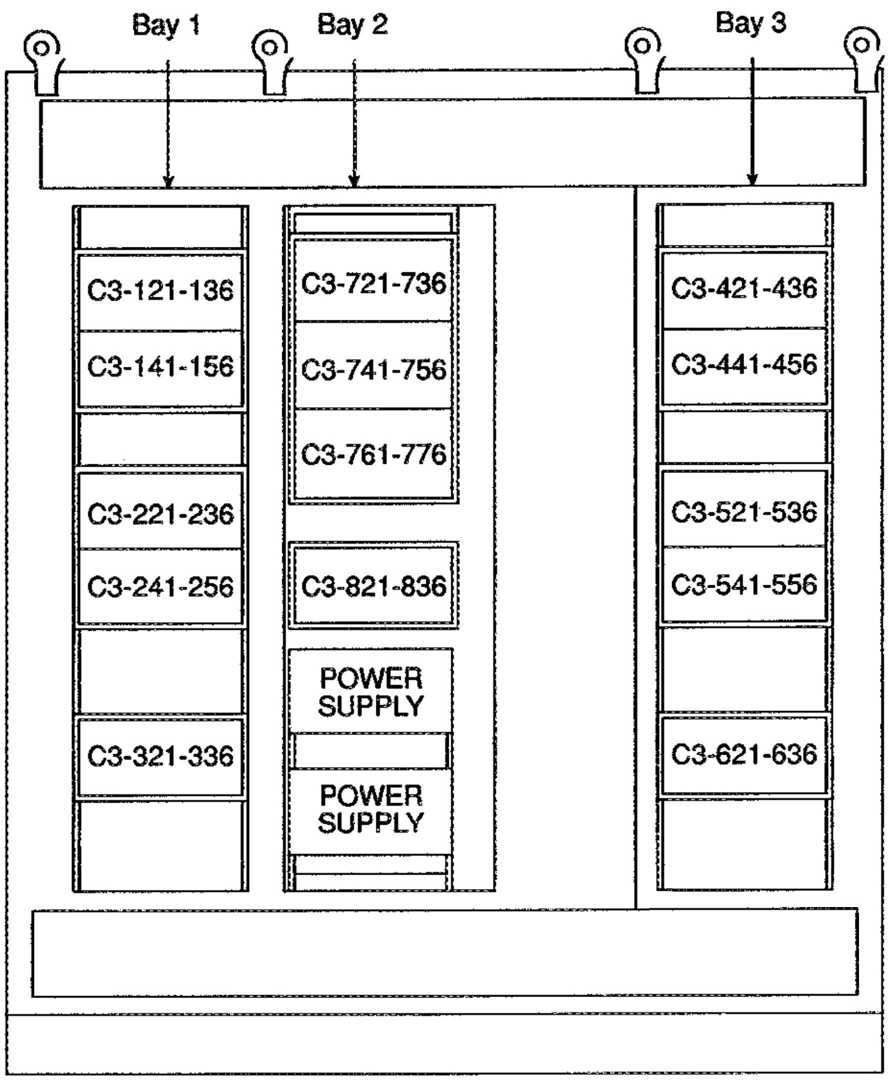
Card positions in a card frame are numbered from RIGHT to LEFT as viewed from the FRONT of the cabinet.

PROCESS TYPICAL 2-BAY **CABINET**

Attachment 3

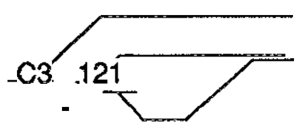
Process Typical 3-Bay Cabinet

NOTE "F" Cabinet Unit 1 *is* Bay 1, 3, 2 and all others *are* Bay 1, 2, 3.



Graphic NO. WT121

Location Code



Cabinet Number
Card Frame Number
Card Slot (Position in Card Frame)

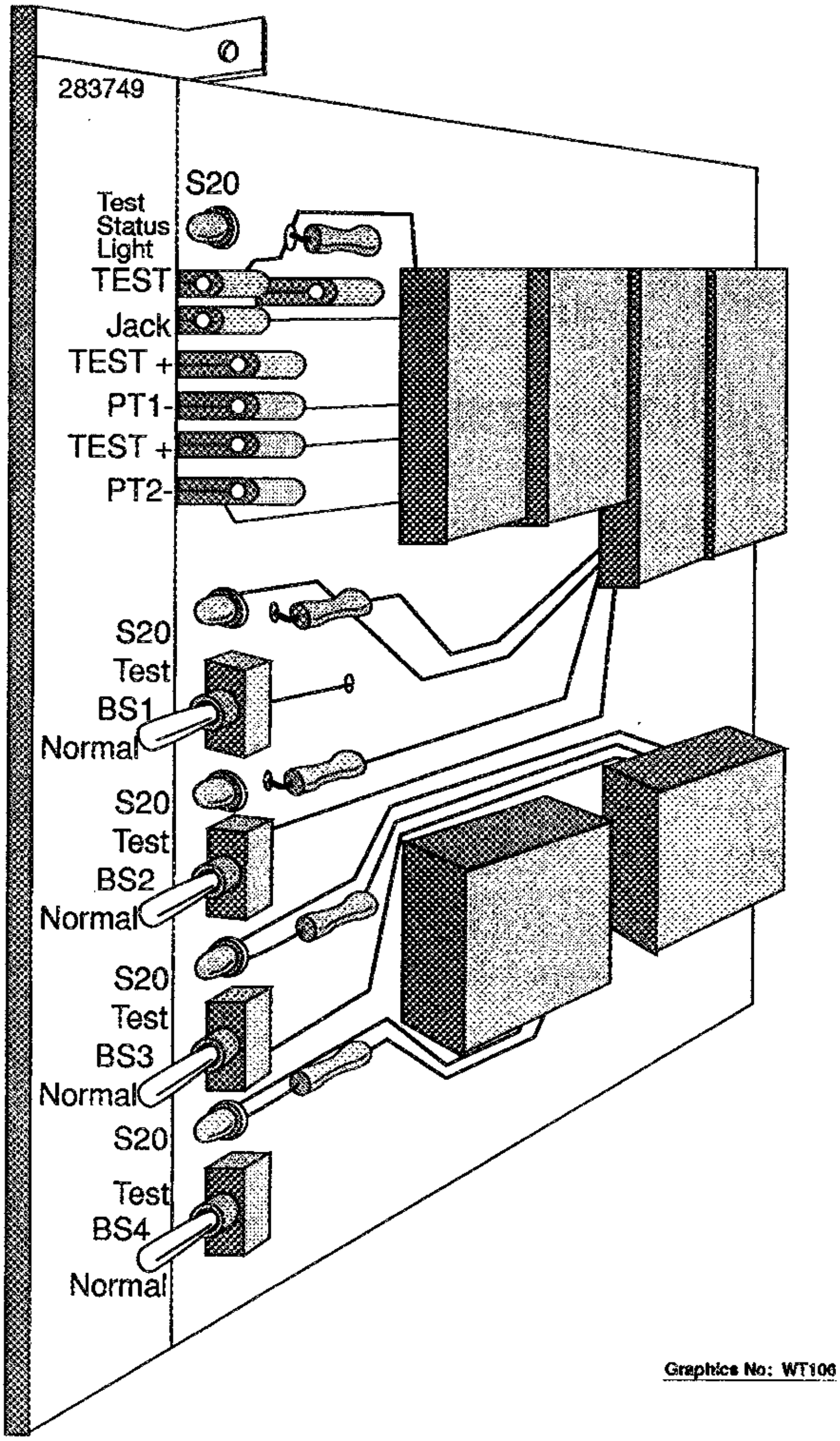
Note: Card positions in a card frame are numbered from RIGHT to LEFT as viewed from the FRONT of the cabinet.

PROCESS TYPICAL %BAY CABINET

(Page 1 of 1)

Attachment 4

Process Typical Channel Test Card



TYPICAL PROCESS CHANNEL TEST CARD

VIRGINIA POWER
NORTH ANNA **POWER** STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-3	LOSS OF VITAL INSTRUMENTATION (WITH TWO ATTACHMENTS)	20
		PAGE 1 of 15

PURPOSE

To provide instructions to follow in the event of a loss of vital instrumentation.

ENTRY CONDITIONS

This procedure is entered when a faulty Indication occurs on any of the following vital instrumentation channels:

- Reactor Coolant Flow, or
- Pressurizer Level. or
- Pressurizes Pressure Protection, or
- DELTA T/TAVE Protection. or
- Containment Pressure Protection, or
- Steam Generator Level. or
- Turbine Stop Valves Indication. or
- Turbine First Stage Impulse Pressure, or
- Turbine Auto Stop Oil Low Pressure Trip Signal. or
- Steam Flow. or
- Feed Flow, or
- Steam Pressure, os
- Station Service Bus Undervoltage. or
- Station Service Bus Underfrequency.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL: APPROVAL - ON FILE	DATE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1] __	VERIFY REDUNDANT INSTRUMENT CHANNEL INDICATION - NORMAL	IF unable to determine Reactor is in a safe operating condition. THEN GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.
[2] __	VERIFY STEAM GENERATOR LEVEL CONTROL PARAMETERS - NOW: • Steam Flow • Feed Flow • Steam Generator Level Gh III • Steam Pressure	Do the following: a) Place the associated valves in **MANUAL**: • Main Feed Reg Valves • Main Feed Reg Bypass Valves b) Control Steam Generator level.
[3] __	VERIFY TURBINE FIRST STAGE PRESSURE INDICATIONS - NORMAL	IF the controlling channel failed. THEN place Control Rod Mode Selector switch in MANUAL.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.__	VERIFY SYSTEMS AFFECTED BY PRESSURIZER LEVEL CHANNELS - NORMAL	
	a) Verify operable Pressurizer level channels - SELECTED	a) Do the following: 1) Place 1-CH-FCV-1122, Charging Flow Control Valve in MANUAL and control level at program. 2) Select operable Pressurizer level channels for control.
	b) Verify Letdown - IN SERVICE	b) Restore letdown using Attachment 2, LETDOWN RESTORATION.
	c) Verify Pressurizer Level Control. . IN AUTO	c) Do the following: 1) Verify level restored to program. 2) Verify expected output of 1-RC-LCV-1459G, Pressurizer Level Control. 3) Place 1-CH-FCV-1122, Charging Plow Control Valve in AUTO.
	d) Verify Pressurizer Control Group Heaters - NOT TRIPPED	d) Reset Pressurizer Control Group Iieaters by placing control switch to START position.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.___	VERIFY BOTH TURBINE FIRST STAGE PRESSURE CHANNELS NORMAL	IF Condenser Steam Dumps are available, THEN transfer to Steam Pressure Mode by doing the following: a) Place both STEAM DUKP INTLK switches to OFF/RESET b) Place STEAM DUMP CONTROLLER to MANUAL c) Place MODE SELECTOR switch to STEAM PRESS d) Ensure Steam Dump demand is ZERO e) Return STEAM DUMP CONTROLLER to AUTO f) Verify Steam Dump demand is ZERO g) Place both STEAK DUMP INTLK switches to ON
6.___	VERIFY OPERABLE CHANNELS SELECTED FOR ALL OF THE FOLLOWING SGWLC INSTRUMENTS: • Turbine First Stage Pressure • "A" SG Steam Flow • "E" SG Steam Flow • "C" SG Steam Flow • "A" SG Feed Plow • "E" SG Feed Flow • "C" SG Feed Flow	Do one of the following as directed by the Unit 1 SRO: • IF desired to swap ONLY the failed channel, THEN GO TO Step a. OR • IF desired to swap ALL SGWLC channels to the same channel. THEN GO TO Step 9.
7.___	GO TO STEP 10	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8.__ SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS:

a) Swap of Turbine First Stage Pressure channel - DESIRED

RNO: a) GO TO Step 8b.

1) Verify Rod Control Mode Selector Switch in MANUAL

RNO: 1) Place Rod Control Mode Selector Switch is in MANUAL.

2) Verify Steam Dumps in one of the following conditions:

- Steam Pressure Mode

<u>OR</u>

- OFF

RNO: 2) Do one of the following with Unit SRO concurrence:

- Place Steam Dumps in - OFF

<u>OR</u>

- <u>IF</u> Condenser Steam Dumps are available. <u>THEN</u> transfer to Steam Pressure Mode by doing the following:

a. Place both STEAM DUMP INTLK switches to OFF/RESET

b. Place STEAM DUMP CONTROLLER to MANUAL

c. Place MODE SELECTOR switch to STEAK PRESS

d. Ensure Steam Dump demand is ZERO

e. Place STEAM DUMP CONTROLLER to AUTO

f. Verify Steam Dump demand is ZERO

g. Place both STEAM DUMP INTLK switches to ON

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED **SGWLC** CHANNEL AS FOLLOWS (Continued):	
	3) Check ALL Bypass Feed Reg valves in **MANUAL**	3) Place ALL Bypass Feed Reg valves are in MANUAL.
	4) Place ALL Main Peed Reg valves in MANUAL	
	5) Select the operable Turbine First Stage Pressure channel for control	
	6) Verify ALL Steam Generator channel III levels - OPERABLE	6) GO TO Step 8a7.
	a. Verify Steam Generator Levels are on program	
	b. Return the Main or Bypass Feed Reg Valves to AUTO. as required	
	7) Verify Condenser Steam Dumps - AVAILABLE	7) GO TO Step 8a9.

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS (Continued):	
	8) Place Steam Dumps in Steam Pressure Kode by doing the following with Unit SRO concurrence:	
	a. Place both STEAM DUMP INTLK switches to OFF/RESET	
	b. Place STEAM DUMP CONTROLLER to MANUAL	
	c. Place MODE SELECTOR switch to STEAM PRESS	
	d. Ensure Steam Dump demand is ZERO	
	e. Place STEAM DUMP CONTROLLER to AUTO	
	f. Verify Steam Dump demand is ZERO	
	g. Place both STEAM DUMP INTLK switches to ON	
	9) Auto Rod Control - DESIRED	9) GO TO Step 8b.
	a. Verify Tave and Tref - MATCHED	
	b. Return Rod Control Mode Selector switch to AUTO	

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS (Continued):	
	b) Swap of Steam Flow channel DESIRED	b) GO TO Step 8c.
	1) Verify affected Main Feed Reg valve in MANUAL	1) Place affected Main Feed Reg valve in MANUAL.
	2) Select the operable Steam Flow channel for control.	
	3) Verify affected Steam Generator Level channel III - OPERABLE	3) GO TO Step 8c.
	a. Verify affected Steam Generator Level is on program	
	h. Return affected Main Feed Reg Valve to AUTO. as required	
	c) Swap of Peed Flow channel - DESIRED	c) GO TO Step 10.
	1) Verify affected Main Feed Reg valve in MANUAL	1) Place affected Main Feed Reg valve in MANUAL.
	2) Select the operable Feed Flow channel for control	
	3) Verify affected Steam Generator Level channel III OPERABLE	3) GO TO Step 10.
	a. Verify affected Steam Generator Level is on program	
	h. Return affected Main Feed Reg Valve tu AUTO. as required	
	d) GO TO Step 10.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.__	SWAP ALL SGWLC CHANNELS AS FOLLOWS:	
	a) Verify Rod Control Mode Selector Switch in MANUAL	a) Place Rod Control Mode Selector Switch in MANUAL.
	b) Verify Steam Dumps in one of the following conditions: • Steam Pressure Mode **OR** • OFF	b) Do one of the following with Unit SRO concurrence: • Place Steam Dumps in - OFF **OR** • **IF** Condenser Steam Dumps are available. **THEN** transfer to Steam Pressure Mode by doing the following: 1) Place both STEAM DUMP INTLK switches to OFF/RESET 2) Place STEAM DUMP CONTROLLER to MANUAL 3) Place MODE SELECTOR switch to STEAM PRESS 4) Ensure Steam Dump demand is ZERO 5) Place STEAM DUMP CONTROLLER to AUTO 6) Verify Steam Dump demand is ZERO 7) Place both STEAM DUMP INTLK switches to ON
	c) Check ALL Bypa Feed R valves in MANUAL	c) Place ALL Bypass Feed Reg valves are in MANUAL.
	d) Place ALL Main Feed Reg valves in MANUAL	

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	SWAP ALL SGWLC CHANNELS AS FOLLOWS (Continued):	
	el Select **ALL** of the following channels to the same channel: • Steam Flow • Feed Flow • First Stage Pressure	
	f) Verify ALL Steam Generator channel III levels - OPERABLE	f) GO TO Step 9g.
	1) Verify Steam Generator Levels are on program	
	2) Return the Main or Bypass Feed Reg Valves to **AUTO**, as required	
	g) Verify Condenser Steam Dumps AVAILABLE	g) GO TO Step 9i.

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. SWAP ALL SGWLC CHANNELS AS FOLLOWS (Continued):

h) Do one of the following with Unit SRO concurrence:

- Place Steam Dumps in Steam Pressure Mode by doing the following:

  1) Place both STEAM DUMP INTLK switches to OFF/RESET

  2) Place STEAM DUMP CONTROLLER to MANUAL

  3) Place MODE SELECTOR switch to STEAM PRESS

  4) Ensure Steam Dump demand is ZERO

  5) Place STEAM DUMP CONTROLLER to AUTO

  6) Verify Steam Dump demand is ZERO

  7) Place both STEAM DUMP INTLK switches to ON

  OR

- Place Steam Dumps in Tave Mode by doing the following:

  1) Verify BOTH channels of Turbine First Stage Pressure are operable

  2) Place both STEAM DUMP INTLK switches to OFF/RESET

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	SWAP ALL SGWLC CHANNELS AS FOLLOWS (Continued):	
	3) VERIFY ANNUNCIATOR PANEL "P" E-4. C-7 PERM STM DUMP ARMED FROM LOSS OF LOAD - NOT LIT	3) Place Steam Dump Mode Selector switch to RESET.
	4) Place MODE SELECTOR switch to TAVE	
	5) Ensure Steam Dump demand is ZERO	
	6) Place both STEAM DUMP INTLK switches to ON	
	i) Auto Rod Control - DESIRED	i) GO TO Step 10.
	1) Verify Tave and Tref - MATCHED	
	2) Return Rod Control Mode Selector switch to AUTO	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: With one instrument channel lost, operations may continue only if the channel is placed in trip condition within the specified time period and the conditions of the applicable Technical Specification are met.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.__	VERIFY OPERATION OF THE FOLLOWING INSTRUMENTS:	
	a) Reactor Coolant Flow Instrumentation indication NORMAL	a) IF unit is in Mode I. THEN complete 1 MOP 55.71. REACTOR COOLANT FLOW INSTRUMENTATION within 72 hours.
	b) Pressurizer Level Instrumentation indication NORMAL	b) IF unit is in Mode 1 or 2, THEN complete 1-MOP 55.72. PRESSURIZER LEVEL INSTRUMENTATION within 72 hours.
	c) Pressurizer Pressure Protection Instrumentation indication - NORMAL	c) IF unit is in Mode 1, 2, or 3. THEN complete 1-MOP-55.73. PRESSURIZER PRESSURE PROTECTION INSTRUMENTATION. Section 5.1 within one hour.
	d) Loop ΔT/TAVE Protection Instrumentation indication NORMAL	d) IF unit is In Mode 1. 7. or 3. THEN complete I-MOP 55.74, LOOP ΔT/TAVE PROTECTION INSTRUMENTATION. Section 5.1 within one hour.
	e) Containment Pressure Protection Instrumentation indication - NORNAL	e) IF unit is in Node 1, 2, 3, or 4, THEN complete 1-MOP-55.75, CONTAINMENT PRESSURE PROTECTION INSTRUMENTATION within 72 hours.
	f) Steam Generator Level Instrumentation indication - NORMAL	f) IF unit is in Mode 1. 2, or 3. THEN complete 1-MOP-55.16. STEAM GENERATOR LEVEL INSTRUMENTATION within 72 hours.
	g) Steam Pressure Instrumentation indication - NORMAL	g) IF unit is in Mode 1. 2, or 3. THEN complete I-MOP-55.79. STEAM PRESSURE INSTRIJMENTATION within 72 hours.

(STEP 10 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	VERIFY OPERATION OF THE FOLLOWING INSTRUMENTS (Continued):	
	h) Steam Plow Instrumentation indication - NORMAL	h) IF unit is in Mode 1, 2, or 3, THEN complete 1-MOP-55.77. STEAM FLOW INSTRUMENTATION within 72 hours.
	i) Feed Flow Instrumentation indication - NORMAL	i) IF unit is in Mode 1 or 2. THEN complete 1-MOP-55.78, FEED FLOW INSTRUMENTATION within 72 hours.
	j) Turbine Stop Valve Closure Signal Instrumentation annunciator indication - NORMAL	j) IF unit is in Mode 1, THEN complete I-MOP 55.80. TURBINE STOP VALVE CLOSURE SIGNAL INSTRUMENTATION within 72 hours.
	k) Turbine First Stage Pressure Instrumentation indication - NORMAL	k) IF unit is in Mode 1. 2. or 3, THEN complete 1-MOP-55.81. TURBINE FIRST STAGE PRESSURE INSTRUMENTATION. Section 5.1 within one hour.
	l) Turbine Auto Stop Oil Pressure annunciator indication - NORMAL	l) IF unit is in Mode 1, THEN complete 1-MOP-55.82. TURBINE AUTO STOP OIL LOW PRESSURE INSTRUMENTATION within 72 hours.
	m) RCP Bus Undervoltage annunciator indication NORMAL	m) IF unit is in Mode 1, THEN complete 1-MOP 55.83. REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 2A, 2B. AND 2C UNDERVOLTAGE within 72 hours.
	n) RCP Bus Underfrequency annunciator indication NORMAL	n) IF unit is in Mode 1. THEN complete 1-MOP 55.84. REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 2A, 2B. AND 2C UNDERFREQUENCY within 72 hours.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.__	VERIFY MAINTENANCE OPERATING PROCEDURE(S) - INITIATED FOR ALL FAULTY INSTRUMENT CHANNELS	IF the failed instrument channel was not in the mode specified. THEN do the following: a) Continue operation. b) Enter Action Statement. c) Do either of the following: • Initiate the appropriate MOP specified in Step 10 for the failed channel(s). OR • Have the 16C department place the failed channel(s) in trip. d) Refer to the applicable Technical Specifications as listed in the Reference Section of the associated MOP. e) DO NOT enter mode specified in Technical Specification until all requirements of Technical Specifications for affected channel have been met. f) Notify Instrument Department to repair faulty channel
12.__	NOTIFY SUPERINTENDENT OF OPERATIONS OR OPERATIONS MANAGER ON CALL OF FAILURE	
13.__	RETURN TO PROCEDURE IN EFFECT	

- END -

- Westinghouse SSP Tech Manual
- Westinghouse Process Instrumentation Manual and Prints
- Instrument Department PTs
- Tech Spec 3.3.1
- Tech Spec 3.3.2
- Tech Spec 3.3.4
- Tech Spec 3.3.3

  CTS 02-92-2506-001, from HPES 92-04
- 1-MOP-55.71. REACTOR COOLANT FLOW INSTRUMENTATION
- 1 MOP 55.72. PRESSURIZER LEVEL INSTRUMENTATION
- 1 MOP 55.73. PRESSURIZER PRESSURE PROTECTION INSTRUMENTATION
- 1-MOP-55.74. LOOP ΔT/TAVE PROTECTION INSTRUMENTATION
- 1-MOP 55.75. CONTAINMENT PRESSURE PROTECTION INSTRUMENTATION
- I-MOP-55.76. STEAM GENERATOR LEVEL INSTRUMENTATION
- I-MOP-55.77. STEAM FLOW INSTRUMENTATION
- 1-MOP-55.78. FEED FLOW INSTRUMENTATION
- 1 MOP 55.79. STEAM PRESSURE INSTRUMENTATION
- 1 MOP 55.80. TURBINE STOP VALVE CLOSURE SIGNAL INSTRUMENTATION
- 1-MOP-55.81. TURBINE FIRST STAGE PRESSURE INSTRUMENTATION
- 1-MOP-55.82. AUTO STOP OIL LOW PRESSURE INSTRUMENTATION
- 1-MOP-55.83. REACTOR PROTECTION SYSTEM INPUT PROM STATION SERVICE BUSES 1A, 1B, AND 1C UNDERVOLTAGE
- 1-MOP-55.84. REACTOR PROTECTION SYSTEM INPUT PROM STATION SERVICE BUSES 1A, 1B, AND 1C UNDERFREQUENCY

- CTS Assignment 02-99-1801-003, Tech Spec Change 290

_____ 1 Ensure Charging Flow is at least 25 gpm.

2 Ensure the following valves are open:

_____ • 1-CH-LCV-1460A, LETDOWN ISOLATION VALVE

_____ • 1-CH-LCV-14608. LETDOWN ISOLATION VALVE

_____ • 1-CH-TV-1204A, LETDOWN ISOLATION VALVE

_____ • 1-CH-TV-1204B, LETDOWN ISOLATION VALVE

_____ 3 Place 1-CH-PCV-1145, LETDOWN PRESSURE CONTROL VALVE, in MAN.

_____ 4 Fully open 1-CH-PCV-1145.

NOTE: To prevent potential overheating of Letdown, Charging flow may need to be increased immediately after establishing Letdown flow.

5 Open the desired Letdown Orifice Isolation Valve(s):

_____ • 1-CH-HCV-1200A, A LETDOWN ORIFICE ISOLATION VALVE

_____ • 1-CH-HCV-12006. B LETDOWN ORIFICE ISOLATION VALVE

_____ • 1-CH-HCV-1200C, C LETDOWN ORIFICE ISOLATION VALVE

_____ 6 Adjust 1-CH-PCV-1145 to obtain 300 psig Letdown pressure as indicated on 1-CH-PI-1145. NONREGCNCRATIVC HEAT EXCH OUTLET PRESS.

_____ 7 Place 1-CH-PCV-1145 in AUTO.

_____ 8 Adjust Charging and Letdcwn to maintain program PRZR level.

-END-

G2.4.45 001

Unit 1 is in Mode 3 when a SI and loss of offsite power occurs. The crew transitioned t(E-1, "Loss of Primary or Secondary Coolant" based on indications of a LBLOCA. The Unit Supervisor is reading the step, "Check if SI Can Be Terminated." The Shift Manager is scanning the annunciator panels. The following illuminated annunciators are noted during this review:

H-F3 4 KV Bus 1H EMR SUP BKR AUTO TRIP

F-E8 AFW SUPPLY 20 MIN WATER REMAINING

J-A2 RWST LO LEVEL

J-F5 LHSI PP 1B LO OR OL TRIP

The Shift Manager uses alternate indications and verifies the conditions that caused these alarms still exist.

Based on these indications the Shift Manager will instruct the crew to ___________

A✓ transition to 1-ES-1.3, "Transfer to Cold Leg Recirculation," then to 1-ECA-1.I, "Loss of Emergency Coolant Recirculation."

B. transition to 1-ES-1.3, "Transfer to Cold beg Recirculation," then back to 1-E-1, " Loss of Primary or Secondary Coolant."

C. perform 1-AP-22.5 "Loss of Emergency Condensate Storage Tank," per the CAP to preclude going to 1-FR-H.1 on a loss of heat sink.

D. verify the LHSI Pump suction transferred to the containment sump and continue in E-1, "Loss of Primary or Secondary Coolant."

A. This is the correct answer. With the H Emergency Bus de-energized and the B LHSI Pump tripped, the crew has lost recirc capability. Transition to 1-ECA-I.1 "Loss of Emergency Coolant Recirculation" will be required.

B. This answer is incorrect. Transfer to ES-1.3 " Cold Leg Recirculation" is required. This will direct you to ECA-1.I with no RHR Pumps running. It does not send you back to E-1 unless you can successfully transfer to cold leg recirc. The transition to ES1.3 off the CAP makes this answer plausible. If examinee doesn't make the connection between the annunciators and a loss of recirc. Capability they will choose this answer.

C. This answer is incorrect. The annunciator setpoint for F-E8 is 46.29%. The CAP directs performance at 40%. It also is not a priority in this accident. H-1 will kick the operator out at step 1 if a LBLOCA is in progress. Steam generators are not needed as a heat sink. Examinee may choose this answer based on CAP criteria and not prioritizing annunciators correctly.

D. This answer is incorrect. The setpoint for annunciator J-A2 is 22.8%. Examinee may associate this with the lowest of setpoints. This would be 3%. Procedural cautions require verification of LHSI Pump suction transfer to the containment sump.

Emergency Procedures/Plan
Ability to prioritize and interpret the significance of **each annunciator or** alarm.

This is a **new question.**

References:

Qbjective 13869 from study guide on Emergency Procedures
1-AR-J-F5 "LHSI **PP 1B LO OR OL TRIP**"
1-AR-J-A2 "**RWST** LO **LEVEL**"
1-AR-F-E8 " **AFW SUPPLY 20 MIN WATER REMAINING**"
1-AR-H-F3 "**4 KV BUS 1H EMR SUP BKR AUTO TRIP**"
1-E1 CAP
1-ES-1.3 step 8

Level(RO/SRO):	SRO	Tier:	3
Group:		Importance Rating:	3.3/3.6
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

3.4: Transfer to Cold-leg Recirculation (1-ES-1.3)

Topic 3.4.1: 1-ES-1.3 General Information 13689

3.4.1a. Objective

List the following information associated with 1-E§-1.3, "Transfer to Cold Leg Recirculation."

- Purpose of the procedure
- Modes of applicability
- Entry conditions
- Major action categories
- Conditions that result in leaving the procedure

3.4.1b. Content

1. Following large-break **loss** of coolant accidents, the inventory in the refueling water storage tank will **be** depleted as water is injected into the RCS and spills to the containment floor via the break.
    1.1. Eventually the RWST level will decrease to 23% which will necessitate a transfer of the **Safety** Injection System to cold leg recirculation.
    1.2. ES-1.3 TRANSFER TO COLD LEG RECIRCULATION provides the operator with the guidance necessary to align the Safety Injection System for cold leg recirculation.

2. ES-1.3 *is* applicable in modes 1, 2, and 3.

3. ES-1.3 may be entered from any of the following Emergency Operating Procedures:
    3.1. E-I, LOSS OF REACTOR OR SECONDARY COOLANT
    3.2. ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS
    3.3. FR-C.I, RESPONSE TO INADEQUATE CORE COOLING
    3.4. FR-C.2, RESPONSE TO DEGRADED CORE COOLING
    3.5. FR-C.3, RESPONSE TO SATURATED CORE COOLING

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

3.7. ES-1.3 may also be entered when RWST level has decreased to 23% and cold-leg recirculation is required.

4. The major action categories associated with ES-1.3 are as follows:

4.4. Align Safety Injection System for recirculation

4.2. Align recirculation spray if necessary

5. After actions have been taken to transfer the Safety Injection System to the recirculation mode, a transition is made from ES-1.3 to the procedure and step in effect.

5.1. If recirculation flow cannot be established, a transition is made from ES-1.3 to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

CONTINUOUS ACTION PAGE FOR 1-E 1

1. ADVERSE CONTAINMENT CRITERIA
   IF either of the following conditions exist. THEN use setpoints in brackets:
   - 20 psia Containment pressure. OR
   - Containment radiation has reached or exceeded 1.E5 R/hr (70% on High Range Recorder).

2. RCP TRIP CRITERIA
   IF both conditions listed below exist, THEN trip all RCPs:
   - Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS, AND
   - RCS subcooling based on Core Exit TCs LESS THAN 20°F [65°F].

3. CHARGING PUMP RECIRC PATH CRITERIA
   - IF RCS pressure decreases to less than 1275 psig [1475 psig] AND RCPs tripped. THEN close Charging Pump Recirc Valves.
   - IF RCS pressure increases to 2000 psig. THEN open Charging Pump Recirc Valves.

4. SI REINITIATION CRITERIA
   IF either condition listed below occurs. THEN manually start Charging Pumps and align BIT:
   - RCS subcooling based on Core Exit TCs - LESS THAN 25°F [75°F], OR
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 21% [40%]

5. ECST LEVEL CRITERIA
   WHEN the ECST level decreases to 40%, THEN initiate 1-AP 22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK 1.

6. SECONDARY INTEGRITY CRITERIA
   IF either of the following conditions exist AND the affected SG has NOT been isolated. THEN GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, STEP 1:
   - Any SG pressure is decreasing in an uncontrolled manner. OR
   - Any SG has completely depressurized.

7. ~~1-E-3~~ TRANSITION CRITERIR
   IF either of the following conditions exist. THEN manually start Charging Pumps. align BIT, and GO TO 1-E-3. STEAM GENERATOR TUBE RUPTURE. STEP 1:
   - Any SG level is increasing in an uncontrolled manner. OR
   - Any SG has abnormal radiation.

8. COLD LEG RECIRCULATION TRANSFER CRITERIA
   IF RWST level decreases to less than 23%. THEN GO TO 1-ES-1.3. TRANSFER TO COLD LEG RECIRCULATION. STEP 1.

9. QS TERMINATION CRITERIA
   IF either condition listed below occurs. THEN perform Attachment 3, TERMINATION OF QUENCH SPRAY:
   - Containment pressure - LESS THAN 12 PSIA. OR
   - RWST level LESS THAN 3% AND QS Pump amps - FLUCTUATING

10. CASING COOLING TANK LEVEL
    WHEN the Casing Cooling Tank level decreases to 4%. THEN close 1-RS-MOV-LOOA and 1-RS-MOV-100B and stop both Casing Cooling Pumps.

11. RCP CRITERIA
    Seal injection flow should be maintained to all RCPs.

12. REACTIVITY CONTROL CRITERIA
    An Operator should be sent to locally close and lock 1 CH 217. PG to Blender Isolation Valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: To prevent pump damage. any pumps taking suction **from** the RWST must **be** stopped before reaching **3%** RWST level.

* *

8. ALIGN SI SYSTEM FOR COLD LEG RECIRCULATION:

a) Verify Low-Head SI Pump - AUTO ALIGNMENT:

1) Low-Head SI Pump Discharge Valves to Charging Pumps - OPEN:

- 1-SI-MOV-1863A
- 1-SI-MOV-1863B

2) Low-Head SI Pump Recirc Valves - CLOSED:

- 1-SI-MOV-1885A
- 1-SI-MOV 1885C
- 1-SI-MOV-1885B
- 1-SI-MOV 1885D

3) Low-Head SI Pump Suction From Containment Sump - OPEN:

- 1-SI-MOV 1860A
- 1-SI-MOV 1860B

4) Low-Head SI Pump Suction From RWST - CLOSED:

- 1-SI-MOV-1862A
- 1 SI MOV 1862B

b) Close Charging Pump Suction From RWST Isolation Valves:

- 1-CH-HOV 1115B
- 1-CH-MOV-1115D

c) Verify continued flow of Charging Pumps - INDICATED:

- 1-SI-FI-1943
- 1-SI-FI-1943-1

RESPONSE NOT OBTAINED (Step 8):

Manually align valves in sequence *as* necessary.

IF at least one flow path from the Containment Sump to the RCS cannot be established OR maintained, THEN do the following:

- Initiate Attachment 2, PRIMARY PLANT VENTILATION ALIGNMENT
- GO TO 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, STEP 1.

4 KV BUS 1H
EMR SUP BKR
AUTO TRIP

.0 Probable Cause

1.1 Generator field over excitation (135A)
1.2 Exciter emergency shutdown button pushed (local or remote)
1.3 Shutdown of diesel (local or remote) with out-put breaker closed
1.4 Engine overspeed (>1050 RPM), lube oil pressure low (<17 PSI), start failure (~662.5RPM and <18 PSIG lube oil press after 7 seconds for start circuit #2 or e250 RPM and coolant pump discharge <10 PSIG after 7 seconds fer start circuit #1), or both emergency stop buttons pushed (local or remote)
1.5 Lube oil temperature high (>230°F), cooling water temp. high (>205°F) or crankcase pressure high (>.6" water), with no auto start signals (Safety Injection or undervoltage on "H" bus;)

.0 Operator Action

2.1 Verify 15H2 trip.

2.2 Verify 1J emergency diesel operating or operable for minimum safeguards.

2.3 Notify Shift Supervisor.

2.4 Determine cause of trip from electrical relay targets or from local diesel alarm panel.

2.5 Submit required Work Requests.

2.6 Refer to 1-MOP-6.90 €or surveillance requirements. 1H EDG is inoperable.

.0 References

3.I LSK-22-12V
3.2 LSK-22-12U
3.3 LSK-22-12S
3.4 LSK-22-12K
3.5 LSK-22-12J
3.6 LSK-22-12L
3.7 Tech Spec 3.8.1.1, 3.8.1.2 (ITS 3.8.1, 3.8.2)
3.8 ET EE-98-037 Rev. 0, Correction to Station Controlled Documents for EDG Speed Switches

.0 Actuation

4.1 Relay 74 - 15H2 auto trip alarm relay
4.2 Refer to emergency diesel generator 1H annunciator responses

'IRGINIA POWER
fORTH ANNA POWER STATION
\PPROVAL: ON FILE

1-EI-CB-ZIF ANNUNCIATOR E8

1-AR-F-E8
REV. 1
Effective Date:07/01/98

AFW SUPPLY 20 MIN WATER REMAINING	46.29% (13 ft)

fOTE: This annuncaitor is based upon an AFW flow rate of 350 gpm. Higher flow rates will result in less than 20 minutes of water remaining.

.0 Probable Cause

1.1 Usage of water from 1-CN-TK-1, Emergency Condensate Storage Tank, to supply Auxiliary Feedwater
1.2 Rupture of 1-CN-TK-1 or Auxiliary Feedwater lines

.0 Operator Action

2.1 Make up to 1-CN-TK-1, Emergency Condensate Storage Tank, from 300,000-gallon CN tanks using 1-OP-31.2, Steam Generator Auxiliary Feedwater System.

2.2 IF Auxiliary Feedwater is NOT in use, THEN verify no leaks in Auxiliary Feedwater System.

2.3 IF 1-CN-TK-1 is inoperable, THEN GO TO 1-AP-22.5, Loss of Emergency Condensate Storage Tank 1-CN-TK-1.

.0 References

3.1 11715-FM-73A, 74A
3.2 11715-ESK-10BAK, 10F
3.3 DCP 92-003-1, Annunciator Windows Engraving and Relocation
3.4 1-AP-22.5, Loss of Emergency Condnesate Storage Tank 1-CN-TK-I
3.5 I-OP-31.2, Steam Generator Auxiliary Feedwater System
3.6 Instrument Loops 11715-CN-001, 071
3.7 CTS Assignment 02-98-2109-008

.0 Actuation

4.1 1-CN-LT-100A input to 1-CN-LC-100A
4.2 1-CN-LT-100B input to 1-CN-LC-100B

RWST LO LEVEL	RWST < 22.8%

1.0 Probable Cause
   1.1 Failure of 1-QS-LT-100A or 100B or 1-QS-LAL-100A-3 or 100B-3
   1.2 CDA in progress
   1.3 Filling Reactor Cavity during refueling
   1.4 Rupture of RP system piping while aligned to RWST
2.0 Operator Action
   2.1 Verify RWST level.
   2.2 IF CDA in progress, THEN *GO* TO 1-ES-1.3, Transfer To Cold Leg Recirc.
   2.3 IF filling Reactor Cavity using LHSI Pump, THEN ensure SI Recirc Mode Reset pushbuttons, Train A and Train B, have been depressed, to prevent swapover of LHSI Pump suction to the Containment Sump.
   2.4 IF borating or filling the RWST *AND* level begins to decrease, THEN secure fill lineup through RP system, UNTIL RP system is verified intact.
   2.5 If level is NOT low, THEN submit Work Request on level transmitter.
3.0 References
   3.1 11715-LSK-29-5B
   3.2 NAPS Instrumentation Manual page QS003 and QS004
   3.3 Loop Diagram 1-L-QS100A
   3.4 EWR 89-571
   3.5 DCP 90-13, Steam Generator Replacement
   3.6 ET CE-96-014, Rev 0, Mode 5 & 6 Compensatory Measures Recommended for Problem Reported in Deviation Report No. N-96-0278
4.0 Actuations
   4.1 1-QS-LSL-100A-2 and 1-QS-LAS-100B-2 RWST low level alarm switches. Feed signals to 1-QS-LAL-100A-3 and 1-QS-LAL-100B-3

IIRGINIA POWER 1-EI-CB-21J ANNUNCIATOR F5 1-AR-J-F5
JORTH ANNA POWER STATION REV. 0
\`\`ROVAL: ON FILE Effective Date:06/13/97

LHSI PP 1B
LO OR OL
TRIP

..0 Probable Cause

1.1 Control switch in "PULL-TO-LOCK"
1.2 Overload trip of 1-SI-P-1B due to mechanical failure, electrical failure, or excessive flow

!.0 Operator Action

2.1 Place control switch in "AUTO AFTER STOP" if 1-SI-P-1B is available.

2.2 Notify Shift Supervisor.

2.3 Submit required Work Requests.

:.0 Reference

3.1 11715-LSK-26-2A
3.2 11715-ESK-10AAX

Actuations

4.1 Instantaneous or inverse time phase to ground overcurrent (64)
4.2 Instantaneous or inverse time overcurrent-Phase A, Phase B, or Phase C (50/51)
4.3 Control switch contacts for 'PULL-TO-LOCK" alarm

009EA2.23 001

Unit 1 has experienced a reactor trip and safety injection. The following plant conditions currently exist:

- RCS temperature is **498** degrees
- RCS pressure is 700 psig
- Steam generator pressures are 980 psig and stable
- Steam generator levels are stable
- No rad monitors are in alarm
- RCP operating parameters are as follows:

	Motor Bearing Temp	Stator Winding Temp	Proximity Vibs
"A RCP	**189°F**	**189°F**	7 mils.
"B" RCP	**154°F**	**194°F**	**4** mils.
"C" RCP	**149°F**	**198°F**	**5** mils.

The crew has just transitioned to **1-E-1, " Loss** of Reactor or Secondary Coolant." The crew should ______

A. trip all RCP's in accordance with the appropriate continuous action page to minimize RCS inventory loss
B. trip **only** the **A** RCP in accordance with the precautions and limitations of **I-QQ-5.2,** "Reactor Coolant Pump Startup and Shutdown," to prevent pump damage
C. trip all RCPs in accordance with the appropriate continuous action page to prevent excessive **WCS** cooldown
D. trip **only** the C RCP in accordance with the precautions and limitations of **1-OP-5.2,** "Reactor Coolant Pump Startup and Shutdown," to prevent pump damage.

A. This **is** the correct answer. Examinee will have to be able to determine subcooling requirements and know the bases for tripping RCP's in Westinghouse **EOP's**.

B. This answer is incorrect. The examinee could choose this answer based on RCP tripping criteria tisted in 1-OP-5.2. The list of criteria is long and the student may recall setpoints but have difficulty determining which setpoint goes to what criteria. The examinee may choose this answer based on "A RCP Motor Bearing Temp being much higher than the other RCP's. The examinee may choose this answer because he got the proximity trip confused with the seismic setpoint.

C. This answer is incorrect. Even if the examinee determines the need to trip all RCP's based on subcooling, they may not **be** clear **as** to the reason why. RCP's are started in the EOP's to give mixing and prevent PTS concerns.

D. This answer is incorrect. The examinee may choose this answer based on C RCP stator temperature being the only one above the Motor Bearing Hi Temp. setpoint of 195 degrees.

Ability to determine and interpret the following as they apply to Small Break LOCA: RCP operating parameters and limits
(CFR: 41.10 / 43.5 / 45.13)

Modified bank question *5996*

References:

Objective 12451 from study guide on Emergency Procedures
1-E-1 Continuous Action Page
Bases document for RCP trip criteria
1-OP-5.2 RCP Startup and Shutdown precautions and [imitations
Steam Tables.

Level(RO/SRO):	SRO	Tier:	1
Group:	1	Importance Rating:	2.8/3.3
Type(Bank/Mod/New):	MOD	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

.–

1 ID: 5996 Points: 1.00

A LOCA has occurred inside the Unit 1 containment. The operating team is implementing I-E-0, Reactor Trip or Safety Injection. Conditions are as follows:

- Containment pressure is 21 psia.
- Containment pressure has remained less than 25 psia
- Core exit temperature is 350°F (average of 5 highest).
- RCS pressure is 276 psig.
- HHSI total flow is 500 gpm.

Which ONE of the following is the required action concerning operation of the RCPs?

A. RCPs should be secured due to low subcooling.

B. RCPs should not be secured.

C. RCPs should be secured due to loss of CC flow.

D. RCPs should be secured due to RCS pressure being < 1275 psig.

Answer: A

Topic 2.1.3: Reactor Coolant Pump Trip Criteria 12451

2.1.3a. Qbjective

Explain the following concepts concerning the reactor coolant pump (RCP) trip criteria in the emergency response guideline procedures.

- Type of accident that requires the RCP trip criteria to be met in order to ensure that core protection is provided
- Consequences of not tripping the RCPs when the criteria are met
- Conditions that require the RCPs to be tripped
- Why **RCPs** should remain running when no high-head safety injection flow is being delivered to the **RCS**
- Why RCS subcooling based on core exit thermocouples is used **as** an RCP trip criteria

2.1.3b. Content

1. Running reactor coolant pumps during accident conditions is beneficial in most cases.
   1.1. Reactor coolant pumps enhance core decay heat removal since they provide forced flew circulation of the reactor coolant.
   1.2. In addition, reactor coolant pump operation allows the use of pressurizer spray for RCS pressure control.
   1.3. However, under certain accident conditions running reactor coolant pumps for extended periods could result in negative consequences.
   1.4. The foldout page of E-0 contains criteria, which directs the operator to trip the reactor coolant pumps based on observed plant conditions.
   1.5. Reactor coolant pump trip criteria is generally required for small-break-loss-of-coolant accidents.
   1.6. Reactor Coolant System inventory depletion exasperated by extended reactor coolant pump operation could cause the core to uncover under certain conditions.

2. During small break loss-of-coolant accidents, the major objective is to leave the reactor coolant pumps running until the point at which the RCS break becomes uncovered.

2.1. When **RCS** inventory has been reduced to the point where the break is uncovered, the reactor coolant pumps are secured in order to allow the coolant to separate into a distinct liquid and vapor phase.

2.2. Once phase separation occurs, steam will vent from the **RCS** break.

2.3. Venting steam will allow for a greater **RCS** pressure reduction with minimum **RCS** inventory depletion.

2.4. As **RCS** pressure is reduced, more high-head safety injection flow will be delivered to the core.

2.5. If reactor coolant pump operation continues beyond the point at which the break uncovers, excessive RCS inventory depletion will occur as a two-phase water steam mixture issue from the break.

2.6. The core would uncover if the reactor coolant pumps were to be tripped later in the event.

2.7. After the phase separation occurs, the core would be blanketed by steam since insufficient liquid inventory would be present.

3. All reactor coolant pumps must be tripped when the following criteria is satisfied:

3.1. At least one charging pump must be running and Rowing to the RCS and;

3.2. **RCS** subcooling based on core exit thermocouples is -25°F [85°] or less.

4. If high head safety injection is not in service, the reactor coolant pumps **should** be left running even when the subcooling criterion is met.

4.1. Without a source of high-pressure makeup to the **RCS.** the core would eventually uncover once the reactor coolant pumps are secured and the phase separation occurs.

4.2. Following the phase separation. depletion of **RCS** inventory will continue with no make up.

4.3. As such, the reactor coolant pumps are left running to allow the circulation of a two-phase mixture and ensure that some degree of core cooling is maintained.

4.4. Additionally, continued reactor coolant pump operation will help expedite **RCS** depressurization.

4.5. Makeup for **RCS** inventory depletion will occur once the **RCS** has been depressurized to the point where injection of the safety injection accumulators occurs followed by LHSI flow.

5. It should be noted that it is not necessary to secure the reactor coolant pumps unless there is a **loss** of **RCS** inventory.

   5.1. The rate at which **RCS** inventory is lost begins to decrease only when the break uncovers and the reactor coolant pumps **are** secured (only steam flow out the break).

   5.2. RCS subcooling is **used** to monltor the extent of **RCS** voiding during a small-break-loss-of-coolant accident.

   5.3. The upper core area will **be** the first region of the core to uncover when the reactor coolant pumps are secured.

   5.4. It is therefore prudent to monitor **RCS** subcooling using the temperature at the core exit.

   5.5. **As** long as adequate **RCS** subcooling exists, **RCS** inventory should be such that the upper region of the core will remain covered if the reactor coolant pumps are tripped.

   5.6. **If** RCS subcooling based on core exit thermocouples **is** adequate, the reactor coolant pumps should be left running.

   5.7. Their continued operation will enhance core cooling and **RCS pressure** control.

   5.8. However, inadequate core subcooling is an indication of excessive **RCS** voiding.

   5.9. In this case. the reactor coolant pumps should be secured **to** minimize **WCS** inventory **loss** and the potential for uncovering the core.

4.5 The first RCP started will initiate forced Flow in all non-isolated RCS Loops, due to reverse flow.

4.6 The RCP **Trip** Criteria are as follows:

- Hot and Cold Leg Isolation Valves for RCP being started open in coincidence with RCP loop **flows** <u>NOT</u> increasing within **30** seconds after closing breaker
- RCP starting current <u>NOT</u> decreasing within 30 seconds after breaker closure
- RCP proximity vibration greater than **20** mils
- RCP seismic vibration greater than 5 mils
- Number 1 Seal ***AP*** less than **200** psid
- Number 1 Seal Leakoff flow less than allowed by the curve in Attachment 1
- Number 1 Seal **Leakoff** flow is ≥ 5.9 gpm
- RCP Motor **Bearing** temperatures greater than **195°F**
- RCP Lower Seal Water Bearing (Pump Bearing) temperature greater than **225°F**
- RCP Stator Winding temperature greater than 300°F
- Loss of **Seal** Injection <u>AND</u> CC to RCP Thermal Barrier

4.7 <u>IF</u> after a pump start, Number 1 Seal ***AP*** is rapidly decreasing <u>AND</u> it is imminent that Number 1 Seal ΔP will decrease to less than 200 psid, <u>THEN</u> the affected RCP <u>MUST</u> be stopped when the Number 1 Seal ***AP*** reaches 240 psid. **This** will ensure enough seal flow is available during pump coastdown.

4.8 Start only ONE **RCP** at a time.

CONTINUOUS ACTION PAGE FOR 1-E-1

1. ADVERSE CONTAINMENT CRITERIA
   IF either of the following conditions exist. THEN use setpoints in brackets:
   - 20 psia Containment pressure. OR
   - Containment radiation has reached or exceeded 1.E5 R/hr *(70%* on High Range Recorder).

2. RCP TRIP CRITERIA
   IF both conditions listed below exist. THEN trip all RCPs:
   - Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS. AND
   - RCS subcooling based on Core Exit TCs - LESS THAN 20°F [65°F].

3. CHARGING PUMP RECIRC PATH CRITERIA
   - IF RCS pressure decreases to less than 1275 psig [1475 psig] AND RCPs tripped. THEN close Charging Pump Recirc Valves.
   - IF RCS pressure increases to 2000 psig. THEN open Charging Pump Recirc Valves.

4. SI REINITIATION CRITERIA
   IF either conditian listed below occurs, THEN manually start Charging Pumps and align BIT:
   - RCS subcooling based on Core Exit TCs - LESS THAN 25°F [75°F]. OR
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 21% [40%].

5. ECST LEVEL CRITERIA
   WHEN the ECST level decreases to 40%, THEN initiate 1-AP-22.5. LOSS OE EMERGENCY CONDENSATE STORAGE TANK I-CN-TK-1.

6. SECONDARY INTEGRITY CRITERIA
   IF either of the following conditions exist AND the affected SG has NOT been isolated. THEN GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION. STEP 1:
   - Any SG pressure is decreasing in an uncontrolled manner. OR
   - Any SG has completely depressurized.

7. 1-E-3 TRANSITION CRITERIA
   IF either of the following canditions exist. THEN manually start Charging Pumps, align BIT. and GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, STEP 1:
   - Any SG level is increasing in an uncontrolled manner. OR
   - Any SG has abnormal radiation.

8. COLD LEG RECIRCULATION TRANSFER CRITERIA
   IF RWST level decreases to less than 23%. THEN GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, STEP 1.

9. QS TERMINATION CRITERIA
   IF either candition listed below occurs, THEN perform Attachment 3, TERMINATION OF QUENCH SPRAY:
   - Containment pressure - LESS THAN 12 PSIA. OR
   - RWST level - LESS THAN 3% AND QS Pump amps - FLUCTUATING

10. CASING COOLING TANK LEVEL
    WHEN the Casing Cooling Tank level decreases to 4%. THEN close 1-RS-MOV 100A and 1-RS MOV-100B and stop both Casing Cooling Pumps

11. RCP CRITERIA
    Seal injection flow should be maintained to all RCPs.

12. REACTIVITY CONTROL CRITERIA
    An Operator should be sent to locally close and lock 1 CH-217. PG to Blender Isolation Valve.

1. ADVERSE CONTAINMENT CRITERIA
   - **IF** either of the following conditions exist, **THEN** use setpoints in brackets:
     - 20 psia Containment pressure. **OR**
     - Containment radiation has reached or exceeded 10^5 R/hr (70% on High Range Recorder).

2. SI FLOW CRITERIA
   IF SI is actuated **AND** High-Head Cold Leg SI flow is **NOT** indicated, **THEN** initiate Attachment 6. MANUAL VERIFICATION OF SI FLOWPATH.

3. RCP TRIP CRITERIA
   IF both conditions listed below exist. **THEN** trip all RCPs:
   - Charging Pumps - AT LEAST ONE RUNNING **AND** FLOWING TO RCS, **AND**
   - RCS subcooling based on Core Exit TCs - LESS THAN 20°F [65°F].

4. CHARGING PUMP RECIRC PATH CRITERIA
   - **IF** RCS pressure decreases to less than 1275 psig [1475 psig] **AND** RCPs tripped. **THEN** close Charging Pump Recirc Valves.
   - **IF** RCS pressure increases to 2000 psig. **THEN** open Charging Pump Recirc Valves.

5. ECST LEVEL CRITERIA
   WHEN the ECST level decreases to 40%. **THEN** initiate 1-AP-22.5, LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1,

6. CDA ACTUATION CRITERIA
   IF Containment pressure exceeds 28 psia. **THEN** do the following:

   a) Manually actuate CDA.
   b) Ensure CC Pumps STOPPED.
   c) Stop all RCPs.
   d) Ensure QS Pumps RUNNING.
   e) Ensure QS Pump Discharge MOVs **OPEN.**
   f) Initiate Attachment 2, VERIFICATION OF PHASE B ISOLATION.
   g) Initiate Attachment 3, PRIMARY PLANT VENTILATION ALIGNMENT.

7. CONTAINMENT RECIRC MODE CRITERIA
   To prevent possible radioactive release from the RWST. VCT level should be maintained greater than 22%.

8. RCP CRITERIA
   Seal injection flow should be maintained to all RCPs.

9. REACTIVITY CONTROL CRITERIA
   An Operator should be sent to locally close and lock 1-CH-217. PG to Blender Isolation Valve.

024AG2.1.32001

Following a normal trip from 100% power, the reactor operator identifies the following

- Shutdown bank "A" rod J3 indicates 45 steps
- Shutdown bank "B" rod G7 indicates 10 steps
- Control bank "C" rod M12 indicates 11 steps
- **Control** bank "D" rod P-8 indicates *22* steps.

Emergency boration was initiated via 1-CH-241, Manual Emergency Borate Valve, at 0845.

Based on these indications, emergency boration must continue until __________

A?' 45% of the boric acid storage tank has been inserted
B. 75 minutes have elapsed since initiation of emergency boration
C. 30% of the boric acid storage tank has been inserted
D. 50 minutes have elapsed since initiation of emergency boration

A. This is the correct answer. Working through the table the answer comes up with 3 equivalent stuck rods. 15% BAST level is what is required per rod to satisfy the requirements of emergency boration. This totals 45%.

B. This answer is wrong. The note at the top of the attachment requires the use of BAST level to verify boration if using 1-CH-241. If examinee doesn't use guidance of the note, this answer would be correct.

C. This answer is wrong. It is the BAST total for 2 stuck rods. Examinee could pick this answer if they incorrectly use the table to derive stuck rod equivalents. The answer was kept divisabfe by 15 to allow examinee to come up with this number.

D. This answer is wrong. This is the boration time for 2 stuck rods. It also doesn't use BAST level to determine adequate boration as required by the note in the front of the attachment. If examinee doesn't use guidance provided by the note or derives stuck rod equivatents wrong they could come up with this answer.

Emergency Boration
Ability to explain and apply all system limits and precautions.

North Anna bank question 5720

References:
Objective 12481 from study guide on Emergency Procedures
1-ES-0.1, Attachment 2
1-ES-0.1, Step 6

QUESTIONS REPORT
for sroquestions

Level(RO/SRO):	SRO	Tier:	1
Group:	2	Importance Rating:	3.4/3.8
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

Topic 2.3.9: Control Rod Insertion Verification in 1-ES-0.1 12481

2.3.9a. Objective

Explain the following concepts concerning verifying that all control rods are fully inserted in 1-ES-0.1, "Reactor Trip Response" (SEN-132, SEN-134).

- **Basis**
- Condition requiring emergency boration
- Three acceptable methods of establishing boration flow
- How the required emergency boration time is determined
- Three conditions that allow termination of emergency boration flow

2.3.9b. Content

1. Following a reactor trip, it is important to verify that all control rods have fully **inserted.**
   - **1.1.** This verification provides the operating crew with assurance that the core is **in** a subcritical condition.
   - **1.2.** The reactor core is designed such that adequate shutdown margin will exist with one rod fully withdrawn after a reactor trip.
   - **1.3.** However, shutdown margin is compromised when one or more control rods are stuck more than 10 steps from the fully inserted position.

2. ES-0.1 directs the operator to initiate an emergency boration when *two* or more control rods indicate that they are stuck more than 10 steps from the fully inserted position.

3. There are three acceptable methods **used** for establishing boration flow.
   - 3.1. Each of these three methods begins with shifting the in-service boric acid transfer pump to fast speed
   - 3.2. The first and most desirable method is to open the emergency borate motor-operated valve electrically by placing its control switch in the OPEN position.

3.3. If this method of emergency boration fails, an operator is dispatched to manually open the emergency borate motor-operated valve locally.

3.4. If the emergency borate motor-operated valve cannot be opened, then the Blender is placed in the Borate Mode to allow the boric acid integrator to function, then the boric acid to blender flow control valve is opened and an operator Is dispatched to locally open the manual emergency borate valve.

4. The amount of emergency boration that is required is determined by using an attachment to ES-0.1.

4.1. The attachment to ES-0.1 has the operator determine "total equivalent stuck rods".

4.2. Any rod indicating greater than 20 steps is considered as one equivalent stuck rod.

4.3. For rods indicating 11 to 20 steps inclusive a table is used to determine the equivalent stuck rods (EQSR)

4.3.1.	1-5 rods	1 EQSR
4.3.2.	6-9 rods	2 EQSR
4.3.3.	10-16 rods	3 EQSR
4.3.4.	17-32 reds	4 EQSR
4.3.5.	33 or more	5 EQSR.

4.4. As an example, assume five control rods failed to fully insert following a reactor trip.

4.4.1. Indicated red position is as follows: 14, 11, 20, 35, and 200 steps.

4.4.2. The two rods that are > 20 steps out have the reactivity worth of 2 EQSR.

4.4.3. The three rods that are between 11 and 20 steps out have the reactivity worth of 1 EQSR.

4.5. The total number of "equivalent stuck rods" is three.

5. Emergency boration is to continue until one of three conditions is satisfied.

5.4. Emergency boration has been in service for the at least 25-minutes for each "equivalent stuck rod."

5.2. Boric acid storage tank level has decreased 15% for each "equivalent stuck rod."

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	VERIFY ALL IRPIs - 10 STEPS OR LESS	IF TWO or more IRPIs indicate greater than 10 STEPS, THEN emergency borate as follows:

a) Place the in-service Boric Acid Transfer Pump in FAST.

b) Open 1-CH-MOV-1350. Emergency Borate Valve.

c) Verify Emergency Boration flow:

- Emergency Boration Flow - 35 GPM OR GREATER
- On-service Boric Acid Tank Level - DECREASING

d) IF Emergency Boration flow NOT verified. THEN locally open 1-CH-MOV-1350 and verify flow.

e) IF Emergency Boration flow STILL NOT verified. THEN open 1-CH-FCV-1113A and locally open 1-CH-241, Manual Emergency Borate Valve.

f) Record the following:

- Time Emergency Boration started: ____________
- Initial on-service BAST level: ____________

g) Initiate Attachment 2, EMERGENCY BORATION FOR CONTROL RODS NOT FULLY INSERTED. to determine when emergency boration can be secured.

h) Have the Shift Supervisor refer to Tech Spec 4.1.1.1.1.a (ITS 3.1.1 and 3.1.4.A).

NOTE: If 1-CH-241 is used **as** the flow path. then the boration amount should **be** verified by the change in BAST level or by a 1-PT-10 series procedure.

1. ___ Determine conditions to stop Emergency Boration:

a) Determine total Equivalent Stuck Rods using the following table:

Actual IRPI Indication	Record IRPI IDs for IRPIs indicating NOT fully inserted	Convert Actual IRPI to Equivalent Stuck Rods (EQSR):	Record Equivalent Stuck Rod Subtotals:
Any Rod >20 steps	______	1 rod = 1 EQSR	______
Rods indicating 11-20 (inclusive) steps withdrawn	______	1-5 rods = 1 EQSR 6-9 rods = 2 EQSR 10-16 rods = 3 EQSR 17-32 rods = 4 EQSR 33 or more = 5 EQSR	______
		Total Equivalent Stuck Rods:	______

b) IF ONLY ONE Total Equivalent Stuck Rod was recorded in Step **la** table, THEN GO TO Attachment **2.** EMERGENCY BORATION FOR CONTROL **RODS** NOT FULLY INSERTED. Step 2 to stop Emergency Baration.

c) IF TWO or more Total Equivalent Stuck Rods were recorded in Step la table. THEN monitor for one of the following conditions to stop Emergency Boration:

- 25 minutes for each Equivalent Stuck Rod has elapsed.

OR

- 15% BAST Level for each Equivalent Stuck Rod has been inserted.

OR

- Adequate shutdown margin has been verified using a 1 PT-10 series procedure.

2. ___ WHEN Emergency Boration is no longer required. THEN stop Emergency Boration as follows:

a) Place Boric Acid Transfer Pump in AUTO.

b) Close valves that were opened:

- ___ 1-CII-MOV-1350
- ___ 1-CH-FCV-1113A
- ___ 1-CH-241

c) Record the following:

- Time Emergency Boration stopped: ________________
- Final on service BAST level: ________________

3. ___ Return to procedure and step in effect

END -

1 ID: 5720 Points: 1.00

Unit 1 was operating at 100% when spurious actuation of the main generator protection relay 86BU resulted in a turbine trip and a rector trip. The operating crew is performing 1-E§-0.4 REACTOR TRIP RESPONSE step 6 "Verify all IRPI's – 10 steps or less. The Reactor Operator identifies the following:

- Shutdown bank " A rod J3 indicates 45 steps
- Shutdown bank "B" rod G7 indicates 10 steps
- Control bank "C" rod M12 indicates 11 steps
- Control bank "D" rod P8 indicates 22 *steps*

The *crew* is unable to open Emergency Boration Valve 1-CH-MOV-1350 electrically or manually and has initiated manual emergency boration via 1-CH-241. Based on the number of stuck rods, manual emergency boration must continue until ________.

A. 45% of the boric acid storage tank has been inserted

B. 30% of the boric acid storage tank has been inserted

C. 75 minutes have elapsed since initiation of emergency boration

D. 50 minutes have elapsed since initiation of emergency boration

Answer: A

Reference Provided. 1-ES-0.1.

vicinity of the A hot leg. This activity is expected to take 20 minutes. The Unit Supervisor___ _______

A. This is the correct answer. Examinee will need to have knowledge of T.S. Bases to determine core mapping is acceptable surveillance to stop the RHR pump. The examinee will also need to know the applicable time frame for this spec.

B. This answer is incorrect. The examinee could choose this answer based on limited knowledge of the bases. Uniform boron cannot be verified without forced circulation. This is mentioned numerous times throughout the bases for RHR, but T.S. Bases allows forced circulation to be stopped for one hour provided nothing is done to dilute the RCS.

C. Phis answer is incorrect. The examinee could choose this answer if they don't differentiate it with the times associated with the low water level spec. The 15 minute time frame bases was added to this distractor to make it plausible.

D. This answer is incorrect. The examinee couid choose this answer based on core outlet temperature requirements. This is the requirement for stopping RHR under the 15 minute time frame. Examinee could think that because cavity level is greater than 23 feet above the flange, RHR is not needed.

Loss of RHR **System**
Knowledge of **bases in technical specifications for limiting conditions** for **operations and safety limits.**

This question is modified from a Diablo Canyon bank question.

References:
Objective 444 from study guide for Residual Heat Removal
Tech Spec 3.9.5 and Bases
Tech Spec 3.9.6 and Bases

QUESTIONS REPORT
for sroquestions

Level(RO/SRO):	SRO	Tier:	1
Group:	1	Importance Rating:	2.5/3.7
Type(Bank/Mod/New):	MOD	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

Questions Marked for Collection

1/1/2000

WEC Diablo Canyon

1

Tech Specs state that in **MODE *6,*** with the water level greater than or equal to 23 feet above the top of the vessel flange, *AT* LEAST one Residual Heat Removal (RHR) **train** shall he operable **and** in operation, except that pump may be de-energized for up to one hour (per eight hour period) under certain conditions. Which **ONE** of the following items describes one of these conditions?

Core alterations are performed in the vicinity of the reactor vessel hot legs.

One RHR pump **remains** operable., and no operations **involving** core alterations are initiated.

No operations **are** permitted that would cause a dilution of the RCS boron concentration to less than 2500 ppm.

Core outlet temperature is maintained at least **40** deg F. below saturation temperature.

Topic 4.1: RHR TS
444

4.1a. Objective

List the following technical specification information as it applies to the Residual Heat Removal System. [444]

- Requirements. including any exceptions, during modes 4 and 5 (ITS 3.4.6, 3.4.7, 3.4.8, TR 3.4.7, SOER-85-4)
- Requirements during modes 1.2, and 3 (TR 3.7.8)
- Requirements, including any exceptions to this requirement, during refueling (ITS 3.9.5, 3.9.6)
- Requirements during dilution (TR 3.1.4)

4.1b. Content

Have trainees refer to copy of TS.

1. There are several technical specifications and one technical requirement that cover RHR requirements in modes 4 and 5.
   1.1. TS-3.4.6 (Mode 4) requires two operable loops consisting of any combination of RCS and RHR loops with one loop operating.
      1.1.1. It *is* permissible for the operating reactor coolant pump (RCP) or RHR pump to be secured up to one hour as long as:
         1.1.1.1. No operations occur which could add water with boron concentration less than required to meet SDM.
         1.1.1.2. Core outlet temperature is at least 10°F below saturation temperature.
            1.1.1.2.1. Prevents a vapor bubble from forming that could disrupt natural circulation flew.
         1.1.1.3. No RCP shall be started with any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2) unless the secondary side temperature of each SG is ≤ 50°F above each of the *RCS* cold leg temperatures.

1.1.1.3.1. Prevents a low temperature overpressurization event due to a thermal transient when an RCP is started.

1.2. **TS 3.4.7** (Mode 5, Loops Filled) requires either two **operable** RHR **loops,** with m e in operation, or one operating RHR loop with the secondary side water level of one SG ≥ 17% (and Its loop stop valves open).

1.2.1. This technical specification is only applicable **if ail** unisolated **loops** are filled (i.e. no air in SG **tubes)**

1.2.1.1. Allows natural circulation to be used as a backup **for** the operating RHR pump

1.2.2. The RHR loop may be removed from operation for ***S*** 1 hour per ***8*** hour period provided:

1.2.2.1. No operations occur which could **add** water with boron concentration **less** than required to meet SDM.

1.2.2.2. Core outlet temperature **is** at least 10°F below saturation temperature.

1.2.3. One RHR loop may be inoperable **for** up to 2 hours for surveillance testing provided that the other RHR loop **is** operable and in operation.

1.2.4. **No** RCP shall be started with any **RCS cold leg** temperatures **235°F** (Unit 1), **270°F** (Unit 2) unless the secondary side **temperature of** each ***SG*** is ≤ 50°F above each of the **RCS** cold **leg** temperatures.

1.2.5. All RHW loops may be removed from operation during planned heatup to Mode **4** when at least one RCS loop in operation.

1.3. TS ***3.4.8*** (Mode 5, Loops Not Filled) requires that two RHR **loops** be operable with one in operation.

1.3.1. Applicable if all loops are isolated, or if any unisolated loop is not filled.

1.3.2. All RHR pumps may be removed from operation for ≤ 15 minutes when switching from one train to another provided:

1.3.2.1. Core outlet temperature is at least 10°F below saturation temperature.

1.3.2.2. No Operations occur which could add water with boron concentration less than required lo meet **SDM.**

1.3.2.3. No draining operations occur to further reduce **RCS** water volume.

1.3.3. One RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is operable and in operation.

1.4. TR 3.4.7 requires that the RHR flow rate of RHR loops required to be operable by TS 3.4.6, 3.4.7, and 3.4.8 be within limits.

1.4.1. 3000 gpm or

1.4.2. 2000gpm, if RCS temperature is ≤ 140°F and time since entry into Mode 3 is ≥ 100 hours.

2. TR 3.7.8 requires two RHR subsystems be available in Modes 1,2, and 3

2.1. An available RHR subsystem is ready to be placed into service and will be operable when the units meets the conditions for system operation.

3. There are two technical specifications that deal with refueling operations.

3.1. TS 3.9.5 states that at least one RHR loop shall be operable and at least one RHR loop shall be in operation during mode 6 with the reactor vessel water level greater than or equal to 23 feet above the top of the reactor vessel Range.

3.1.1. The required RHR loop may be removed from operation for ≤ 1 hour provided that no operations occur which could add water with boron concentration less than required to meet SDM..

3.1.1.1. This permits operations such as core mapping or alterations in the vicinity of the hot leg nozzles.

3.2. TS 3.9.6 requires two independent RHR loops be operable with at least one loop in operation with RCS level less than 23 feet above the top of the reactor vessel flange.

3.2.1. All RHR pumps may be removed from operation for ≤ 15 minutes when switching from one train to another provided:

3.2.1.1. Core outlet temperature is at least 10°F below saturation temperature,

3.2.1.2. No operations occur which could dilute the boron concentration

3.2.1.3. No draining operations occur to further reduce RCS water volume.

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

------------------------------ NOTE ------------------------------

The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System (RCS), coolant of boron concentration less than reauired to meet the minimum reouired boron concentration of LCO 3.9.1.

APPLICABILITY: **MODE** 6 with the rater level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO **3.9.1.**	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	A.5 Close one door in each installed air lock.	4 hours
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.1 Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	12 hours

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation–Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

------------------------------NOTES------------------------------

1. All RHR pumps may be removed from operation for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10°F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS volume are permitted.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONQITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Inmediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loop in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to operation.	Inmediately
	AND		
	B.3	Close equipment hatch and secure with four bolts.	4 hours
	AND		
	B.4	Close one door in each installed air lock.	4 hours
	AND		
	B.5.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of: a. ≥ 3000 gpm, or b. ≥ 2000 gpm if RCS temperature ≤ 140°F and time since entry into MODE 3 ≥ 100 hours.	12 hours
SR 3.9.6.2	------------------------NOTE------------------------ Not required to be performed until 24 hours after a required RHR pump is not in operation. -- Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Hater Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F. boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit removal of the RHR loop from operation for short durations. under the condition that the boron concentration is not diluted. This conditional removal from operation of the RHR loop does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be

(continued)

LCO (continued)

OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping. instruments, and controls to ensure an OPERABLE flow path and to determine the RHR discharge temperature. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

BASES

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation. except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS far minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling. decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly. is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE **6** and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and **A.6.2**

If LCO 3.9.5 is not met, the following actions must be taken:

a. the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;

b. one door in each installed air lock must be closed; and

(continued)

BASES

ACTIONS

A.4, A.5, **A.6.1,** and **A.6.2** (continued)

c. each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve. blind flange, or equivalent. or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. UFSAR, Section 5.5.4.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F. boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE. and one train in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

(continued)

BASES

LCO (continued)

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flaw is stopped. Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing unit configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves. piping, instruments and controls to ensure an OPERABLE flow path and to determine the RHR discharge temperature. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MORE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level."

ACTIONS

A.l and A2

If less than the required number of RHR loops are OPERABLE, action shall be imnediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation

(continued)

ACTIONS

A.1 and A.2 (continued)

or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator tu initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore une RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

If no RHR is in operation, the following actions must be taken:

a. the equipment hatch or equipment hatch cover must be closed and secured with at least Pour bolts;

b. one door in each installed air luck must be closed; and

c. each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent. or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant tu boil and release radioactive gas to the containment atmosphere. Performing the actions described

(continued)

BASES

ACTIONS

8.3. B.4, B.5.1, and B.5.2 (continued)

above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENYS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level lowered to the level of the reactor vessel nozzles. the RHR pump net positive suction head requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

The SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. UFSAR, Section 5.5.4.

029GG2.1.28 001

When removing the containment purge from service, 1-OP-21.2, " Containment Purge" closes the purge supply and exhaust valves and locks open their power supply. What is the basis behind this action?

A. Purge supply and exhaust valves are not qualified for automatic closure during a DBA due to their large size.
B. 1-WM-RMS **162,** MANIPULATOR CRANE RADIATION MONITOR is not available for automatic isolation in modes 1 through ***4.***
C. Piping outside containment **is** not seismically qualified.
D. Iodine filters do not have enough capacity to handle a release through this flowpath during a LOCA.

A. This is the correct answer. Per the basis of T.S. 3.6.3 credit cannot be given for automatic closure of these valves during a DBA due to their large size. Locking the breaker open satisfies the action for an inoperable automatic containment isolation valve.

B. This answer is incorrect. 1-RM-RMS-162, MANIPULATOR CRANE RADIATION MONITOR is out of service in modes 1 through 4, The examinee could choose this answer based on this knowledge. 1-RM-RMS-159/160, CONTAINMENT PARTICULATE AND GASEOUS RADIATION MONITORS still provide isolation in modes **1** through 4.

C. This answer **is** incorrect. Certain sample lines are not seismically qualified and have *to* be isolated in the EOP's during a tube rupture. The examinee may use this same logic to choose this answer.

D. This answer **is** incorrect. There are flow limitations put on the iodine filters to handle *a* release during a BBA, however the release **is** through the safeguards fans not the containment purge.

Containment Purge
Knowledge of **the purpose and function** of major **system components and controls**

Based on combination of several North Anna bank questions

References:
Objective 4492 and 4502 from study guide on Primary Ventilation
1-OP-21.2 Containment Purge

Level(RO/SRO):	**SRO**	**Tier:**	*2*
Group:	*2*	Importance **Rating:**	3.2/3.3
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	**Last Exam(Y):**	**N**

valve, which permits reduced purge flow if required. Motor operated valve MOV-HV-100C is located in the purge exhaust line inside Containment. When low purge flow rates are required, MOV-HV-IO1 is opened and MOV-HV-100D is shut. All containment purge system containment isolation MOVs for a unit automatically shut when a Hi Hi radiation condition exists in the Containment Structure for that unit. The valves trip shut on a signal from radiation monitor RM-159, -160, or -162 (refer to module NCRODP-46-NA, Radiation Monitoring System).

These motor-operated valves are driven by 1/7-HP motors. MOV-HV-100A and -100C are powered from MCC 1A1-1. MOV-HV-100B and -1000 are powered from MCC 1B1-2.

Containment Pressure Equalizing Valve. Before the purge equipment can be placed in service, containment pressure must be equalized with atmospheric pressure. An 1&inch pressure equalizing valve (MOV-HV-102) provides that capability and is located between the purge supply containment isolation valve and the containment wall. This MOV automatically shuts when a Hi Hi radiation condition in Containment is detected by radiation monitor RM-159, -160, or -162. This motor-operated valve is driven by a 1/7-HP motor and is powered from MCC 1C1-1. A temporary spoolpiece connection is made at MOV-HV-102/202 to allow portable air compressors to pressurize the Containment during Type-A leakage testing of the Containment Structure.

Containment Purge Exhaust Iodine Filtration Equipment Bypass Dampers. Four dampers are used to control the direction of ventilation exhaust air. Exhaust air is normally discharged directly to the environment from the containment via two air-operated dampers (AOD-HV-104-1 and 104-2) in line with the exhaust fan suction. If the air becomes contaminated, the exhaust air is redirected through the Auxiliary Building iodine filter banks by way of iodine filter supply and return dampers AOD-HV-104-4 and 104-3. These dampers can be controlled from either unit's Vent Panel with a handswitch.

Containment Elevator Exhaust Fan. Fan HV-F-53 exhausts air from the elevator machinery room at a rate of 2,800 cfm. The fan is powered from MCC 1C1-1 and starts the elevator fan when the space temperature exceeds 110°F (as sensed by local temperature switch TS-HV-161).

Appendix R Ventilation Subsystem

The Appendix R Ventllation Subsystem is designed to ventilate critical areas in the Auxiliary Building and fuel Building if the normal ventilation equipment which ventilates these areas is disabled by fire. The system is normally shutdown, and operated only during emergencies addressed by FCA procedures.

The critical areas ventilated are:

1. Auxiliary Building Component Cooling Area
2. Auxiliary Building Charging Pump Cubicles
3. Fuel Building Auxiliary Monitoring Panel Area

MCC 1B2-4 respectively. The fans operate in parallel to provide step purge control and each supplies 11,000 cfm of air to Containment. The air is distributed to the Containment confines by ring ducts which run the entire circumference of each of the Containment Structures.

The fans are manually operated by individual handswitches on the ventilation control panels in the MCR (HV-F4A from Unit 1 and HV-F-48 from Unit 2). The fans trip or will not start if both units' containment purge supply isolation valves are not fully open, the air temperature at the air handiing unit outlet drops below 35°F, or on a fan motor overload. They also trip if a Hi Hi radiation condition (Containment Atmosphere Gaseous or Particulate or Fuel Manipulator Area RMS) occurs in either or both containment Structures being purged.

Also associated with each supply fan is a self-actuating discharge damper. The force of air being discharged by the supply fan opens the discharge damper. Gravity forces the damper shut when the fan is stopped. The dampers are in place to prevent air from being short cycled through either of the parallel fans in the event one should fail.

Containment Purge Exhaust Fans. The Containment equipment uses two vane axial type fans to expel stale air from either OF both containment Structures when they are opsned for maintenance. The fans are physlcaily located in the Auxiliary Building 291-foot elevation Southwest. Fans 1-HV-F-5A and -5B are driven by 40-HP, 480V ac motors, powered from MCC 1C1-3 and MCC 1B2-4 respectively. The capacities, control, trips, and operation of these fans are the same as those described for the purge supply fans except isolation damper interlocks associated with the exhaust lines. The fans exhaust to the environment by way of exhaust stack B. The fans are controlled from MCR Vent Panels (HV-F-5A from Unit 1 & HV-F-5B from Unit 2).

Containment Purge Fan Exhaust Dampers. Motor-operated exhaust dampers MOD-HV-101A and -101B are interlocked with their associated exhaust fan's operation. Each damper is located on the fan's discharge side. When the fan is started, the damper opens. When the fan is stopped, a spring returns the damper to the shut position. As such, the exhaust fan discharge dampers fail shut.

Containment Purge Isolation Valves. The containment purge equipment supplies air to both Containment Structures via separate supply headers which connect to the containment ventilation ring ducts mentioned previously. Each containment supply header has motor-operated butteffly isolation valves located on each side of the Containment Structure wail. The arrangement for Unit I is described; Unit 2 is identical with only valve number prefixes differing (MOV-HV-100B for Unit 1 and MOV-HV-200B for Unit 2). Containment purge air supply isolation valve MOV-HV-100B is located outside Containment, while isolation valve MOV-HV-IOOA is located inside Containment.

Isolation valves are also located in the exhaust lines on both sides of the Containment wall. There are two isolation valves (MOV-HV-100D and -101) in parallel outside the Containment Structure. MOV-HV-101 is an 8-inch bypass

.029000.K **1.04** 6/22/1998

WEC Point Reach 1

Which ONE of the following is the reason the Containment Purge Supply and Exhaust Valves ***are*** required to **be** locked closed during operations at power?

The valves' capability to close during a design basis loss-of-coolant accident has **NOT** been demonstrated

The related piping systems outside containment **are** NO?' seismically qualified

The valves **are NOT** seismically qualified to operate during a design basis earthquake.

The valve actuators do NOT **have** class 1E penetration conductor overcurrent protection devices

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow path(s) except for 36 inch purge and exhaust valves, 18 inch Containment vacuum breaking valve, 8 inch purge bypass valve, and steam jet air ejector suction flaw paths may be unisolated intermittently under administrative controls.

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

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment,' when leakage for a penetration flow path results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE--------- Only applicable to penetration flow paths with two or more containment isolation valves. .... One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

BASES

BACKGROUND (continued)

Containment Purge System (36 inch purge and exhaust valves, 18 inch containment vacuum breaking valve, and 8 inch purge bypass valve)

The Containment Purge System operates tu supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 36 inch purge valves are not qualified for automatic closure from their open position under Design Basis Accident (DBA) conditions. Therefore, the 36 inch purge valves are maintained closed in MODES 1, 2, 3, and 4 tu ensure the containment boundary is maintained. The 18 inch containment vacuum breaking valve and 8 inch bypass valve are also maintained closed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it in assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 36 inch purge and exhaust valves are closed at event initiation.

The DBA analysis assumes that. within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, La. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke 'times.

(continued)

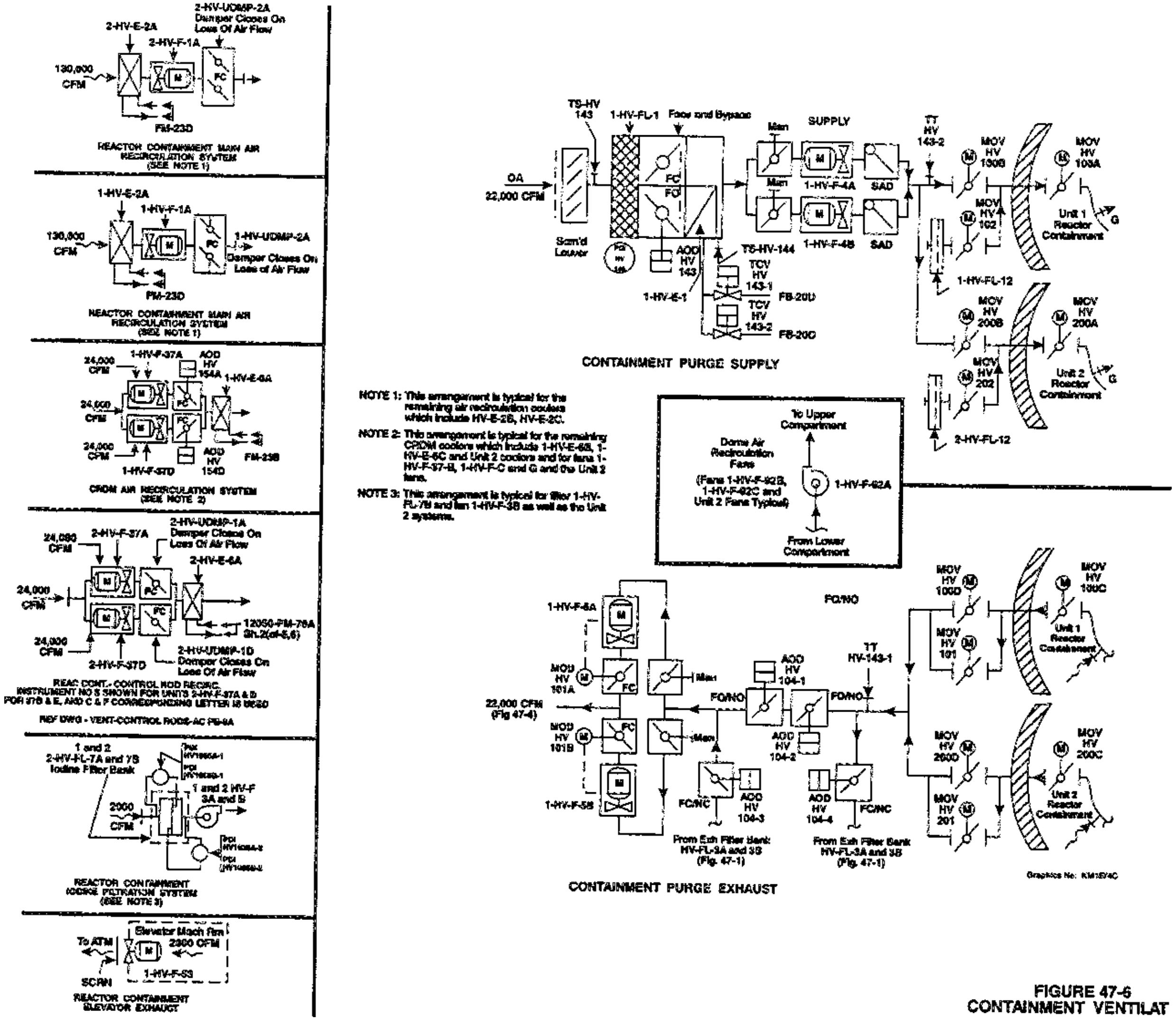


FIGURE 47-6
CONTAINMENT VENTILAT

Figure 47-6-NA
Containment Ventilation Subsystem

Dominion	PROCEDURE NO: 1-OP-21.2
NORTH ANNA POWER STATION	REVISION NO: 27
PROCEDURE TYPE: OPERATING PROCEDURE	UNIT NO: 1

PROCEDURE TITLE:

CONTAINMENT PURGE.

REVISION SUMMARY:

FrameMaker Template Rev. 030.

Incorporated Operations DR N-2003-1252:

- Added Reference for PI N-2003-1252-R1.
- Added Step 5.2.3 to notify Health Physics Shift Supervisor prior *to* securing Containment Purge.

PROBLEMS ENCOUNTERED: ☐ NO ☐ YES *Note:* If YES. note problems in remarks.

REMARKS:

(Use back for additional remark.)

SHIFT SUPERVISOR: BATE:

TABLE OF CONTENTS

1.0 PURPOSE

1.1 To provide instructions for placing the containment purge system in operation.

1.2 To provide instructions for removing the containment purge system from operation.

2.0 REFERENCES

2.1 Source Documents

2.1.1 **UFSAR,** Section 9.4.9, Containment Structure

2.1.2 UFSAR, Section 12.2.2.2, Ventilation, Containment Structure

2.2 Technical Specifications

2.2.1 Tech Spec 3.6.1

2.2.2 Tech Spec 3.6.3

2.2.3 Tech Spec 3.6.4

2.2.4 Technical Requirements Manual 3.3.7

2.2.5 Technical Requirements Manual 3.9.5

2.3 Technical References

2.3.1 11715-FM-006A, Air Cooling and Purge System

2.3.2 0-OP-21.5, Operation of Auxiliary Building Iodine Filters

2.3.3 1-PT-61.3.2, Containment Purge Valves As Found Pressure Test

2.3.4 **Engineering** Transmittal ET SE-96-0035, included with REV. 18

2.3.5 ICP-RMS-1-RM-159, RMS - 159 Containment Particulate Radiation Monitor Calibration

2.3.6 ET SE-99-046, Rev. 0, Terms used in Tech Spec Sections 4.6.4.3.b, 4.7.7.1.b, and 4.7.8.B.b Regarding Charcoal Filters

2.3.7 Safety Evaluation 96 SE-OT-28, included with Rev. 18

2.4 **Commitment Documents**

2.4.1 CTS Assignment 02-91-2805-003, SER 1-91, Spent Fuel Pool Overflow Events

2.4.2 CTS Assignment 02-94-2220-001, Surry Enforcement Conference - 9/30/94, Charcoal Filter Degradation

2.4.3 PI N-2003-1252-R1, Mis-Communications Between Ops and HP.

3.0 INITIAL CONDITIONS

3.1 Review the equipment status to verify station configuration supports the performance of this procedure.

3.2 A known atmospheric condition exists in the containment as sampled by Health Physics.

3.3 Reactor is shutdown, in either mode 5 or 6 OR defueled.

3.4 IF in Mode 6, THEN at least one Containment Air Recirc Fan is in operation to provide representative sample for 1-RM-RMS-159 and 1-RM-RMS-160.

3.5 IF in Mode 6, THEN the following Radiation Monitors are OPERABLE:

- 1-RM-RMS-159, CONTAINMENT PARTICULATE RADIATION MONITOR
- 1-RM-RMS-160, CONTAINMENT AREA GAS RADIATION MONITOR
- 1-RM-RMS-162, MANIPULATOR CRANE RADIATION MONITOR

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Comply with ***the*** following guidelines when marking steps N/A:

- IF the conditional requirements of a step do not require the action **to** be performed, THEN mark the step N/A.
- IF any other step is marked N/A, THEN have the Shift Supervisor (or designee) approve the N/A **and** justify the N/A on the procedure Cover Sheet.

4.2 The reactor containment shall not be purged while the reactor coolant system temperature is > 200°F and containment is sub atmospheric.

4.3 All automatic containment isolation valves in that unit shall be operable or at least one valve in each line shall be closed except in those systems which must be operated during refueling.

4.4 To prevent the possible collapse of purge system duct work, when it is desired to estabiish atmospheric conditions in the containment, do NOT under any circumstances open 1-HV-MOV-100B, CONT PURGE SUPPLY VALVE (outside valve) while vacuum *is* being broken.

4.5 Purges shall go through filter unless written authorization from Health Physics states otherwise.

4.6 Ventilation changes made when the fuel transfer **tube** is open may cause a pressure differential between Containment and the Fuel Building and result in Bevel changes in the Spent Fuel Pit or Reactor Cavity. (Reference **2.4.1)**

4.1 Due to possible **Charcoal** Filter degradation, the Auxiliary Building Iodine Filters MUST **NOT** be placed in service during or following painting, fie, or chemical release in **any** ventilation zone communicating with the inlet to **any** Auxiliary **Building** Iodine Filters which will be operated. **IF** the Auxiliary Building Iodine Filters MUST be placed in service under these conditions, such as to mitigate radiological events, **THEN** ***the*** surveillance requirements of Tech **Spec** 5.5.10 and Ventilation Filter Testing Program (VFTP) must be satisfied. **(References 2.4.2** and **2.3.6)**

4.8 At least one Containment Air Recirc Fan must be in operation to provide a representative sample for 1-RM-RMS-159 and 1-RM-RMS-160.

4.9 Sample **flow** for **1-RM-RMS-159** and 1-RM-RMS-160 must be between 8 cfm and 12 cfm to be considered operable **to** satisfy Technical Requirements Manual TR 3.3.7. **(References 2.3.4. and 2.3.7)**

..nit Verif

5.0 INSTRUCTIONS

5.1 Placing Purge System in Operation

______ 5.1.1 Verify Initial Conditions are satisfied.

______ 5.1.2 Review Precautions and Limitations.

NOTE The reactor containment shall NOT be purged while the reactor coolant system temperature is > 200°F AND containment is sub-atmospheric.

NOTE All automatic containment isolation valves in the unit shall be operable OR at least one valve in each line shall be closed except in those systems which must be operated during refueling.

5.1.3 IF H. P. form has NOT been initiated, THEN initiate H. P. form for "Reactor Containment Release Permit."

______ 5.1.4 IF Reactor is to be refueled or maintenance outage is to be commenced, THEN PRIOR to breaking containment vacuum by movement of any of the Purge and Exhaust Valves, 1-PT-61.3.2 MUST be completed. Type "C" Engineer may be contacted for further guidance.

5.1.5 Unlock **and** close the following breakers:

______ ______ SV
- 1-EP-BKR-1C1-1 B1, Purge Exhaust Bypass Valve Circuit Breaker, 1-HV-MOV-101

______ ______ SV
- 1-EP-BKR-1C1-1 B2, Cntmt Purge Supply Bypass line Isol Vv Circuit Bkr, 1-NV-MOV-I02

______ ______ SV
- 1-EP-BKR-1A1-1 C3, Purge Sply Fans to **Reac** Cntmt Isol Vv Circuit Bkr, 1-HV-MOV-100A

______ ______ SV
- 1-EP-BKR-1A1-1C4, Reac Cntmt to Purge Exh Isol Valve **Circuit** Bkr, 1-HV-MOV-100C

______ ______ SV
- 1-EP-BKR-1B1-2 C3, Cntmt Supply Line Isol Vv Circuit Breaker, 1-HV-MOV-100B

______ ______ SV
- 1-EP-BKR-1B1-2 C4, Cntmt Purge **Exhaust** Line Isol Vv Circuit Breaker, 1-HV-MOV-100D

5.1.6 <u>IF</u> required, <u>THEN</u> remove locks from the following valve handwheels:

______ ______
- 1-HV-MOV-100D, Containment Purge **Return Line** Isolation Valve

______ ______
- **1-HV-MOV-101, 1-HV-MOV-100D** Bypass Valve

______ ______
- 1-HV-MOV-IWB, Containment Purge Supply Line Isol Valve

______ ______
- 1-HV-MOV-102, Containment Purge Supply Vent Line Isol Valve

______ ______
- 1-HV-MOV-100A, Purge Supply Fans To Reactor Containment Isol

______ ______
- **1-HV-MOV-100C,** Reactor Containment **To** Purge Exhaust Isolation

______
5.1.7 Open 1-HV-MOV-100A, CONT PURGE SUPPLY VALVE (inside valve).

NOTE: **When** it is desired to establish atmospheric conditions in the containment, do NOT under any circumstances open **1-HV-MOV-100B,** CONT PURGE SUPPLY **VALVE** (outside valve), while vacuum is being broken. This to prevent the possible collapse of purge system duct work.

5.1.8 IF containment is NOT at atmospheric pressure, THEN do the following:

a. Open **1-IN-MOV-102, CONT** PURGE RELIEF **VALVE,** to raise the containment pressure to atmospheric.

b. **WHEN** containment AND atmospheric pressure are equalized as indicated on 1-PI-LM-100A, B, C or D, THEN close **1-EIV-MOV-102,** CONT PURGE RELIEF **VALVE.**

5.1.9 Ensure the switch for 1-HV-3A/3B, Containment Iodine Filter Fans, is in the OFF position.

5.1.10 IF Mode 6 entry is anticipated, THEN do the following: **(References 2.3.4 and 2.3.7)**

a. Notify the I&C Department to perform the section of **ICP-RMS-1-RM-159, RMS-159** Containment Particulate Radiation Monitor Calibration, for Rotameter Calibration for Operation.

b. Enter in the Action Statement Log **to** verify that 1-RM-RMS-159 and 1-RM-RMS-160 are operable with a sample flow rate of between 8 cfm and **12 cfm** prior to movement of recently irradiated fuel within the containment per TRM 3.3.7.

NOTE HP Release Form must be obtained prior to proceeding with this OP.

NOTE: Purge shall go through filters unless written authorization from Health Physics states otherwise.

5.1.11 Align containment purge through the iodine filters or to bypass the filter as per 0-OP-21.5, Operation of Auxiliary Building Iodine Filters.

_____ 5.1.12 Open 1-HV-MOV-100B, CONT PURGE SUPPLY VALVE (outside valve.).

_____ 5.1.13 Open 1-HV-MOV-100C, CONT PURGE EXH VALVE (inside valve).

5.1.14 Open applicable Purge Exhaust valve. Mark valve not opened N/A:

_____ • IF max allowable release rate is less than 11,000 CFM, THEN open 1-MOV-HV-101, CONT PURGE EXH BYPASS VALVE.

_____ • IF max allowable release rate is greater than or equal to 11,000 CFM, THEN open 1-MOV-HV-100D, CONT PURGE EXH VALVE (outside valve).

NOTE: *An* additional suppiy and exhaust fan may be used for additional ventilation if allowable release rate ≥ 22,000 CFM.

5.1.15 Start one OR both containment purge exhaust fans. Mark fan not started N/A:

• 1-IIV-F-SA,5A REACTOR CNTMT HV PURGE EXHAUST FAN

• 1-HV-F-5B, 5B REACTOR CNTMT HV PURGE EXHAUST FAN

_____ 5.1.16 Note. flow increase on flow recorder, 1-HV-FR-1212B, MISC AREA EXHAUST AIR FLOW TO VENT STACK "B."

_____ 5.1.17 IF flow is greater than maximum allowable release rate, THEN throttle close 1-MOV-HV-101, CONT PURGE EXH BYPASS VALVE.

5.1.18 IF additional ventilation is required, THEN start one containment supply fan.

_____ • 1-HV-F-4A, REACTOR CNTMTS HV PURGE SUPPLY FAN

_____ • 1-HV-F-4B, REACTOR CNTMTS HV PURGE SUPPLY FAN

NOTE The following parameters initiate automatic functions whish affect the purge and exhaust system.

- High radiation levels in containment (1-RM-RMS-159, 160 or 162) will dose the containment purge system butterfly valves.
- Air temperature leaving the steam heating coils at < 35°F will trip the containment purge supply fan...
- Purge Supply Fans 1-HV-F-4A AND 4B will trip IF 1-MOV-HY-IOOA OR 1-MOV-HV-100B are NOT FULL OPEN.
- **Purge** Exhaust **Fans** I-HY-F-SA AND 5B will **trip** IF 1-MOV-HV-100C OR 1-MOV-HV-100D are NOT FULL OPEN AND 1-MOV-HV-101 IS FULL **CLOSED.**

NOTE Steam Heating is required if Containment temperature is less than **65°F.**

______ **5.1.19** IF directed by **Health** Physics, THEN remove containment purge **flow** from the iodine filter as per 0-OP-21.5, Operation of Auxiliary Building Iodine Filters.

Completed by: ______________________________ Date: __________

5.2 Removing Purge System from Operation

______ 5.2.1 Verify Initial Conditions **arc** satisfied.

______ 5.2.2 Review Precautions and Limitations.

______ 5.2.3 Notify the HP **Shift** Supervisor that the Containment Purge will be **secured.**

5.2.4 Ensure containment purge supply fans are secured:

______ • **1-HV-F-4A,** REACTOR CNTMTS HV PURGE SUPPLY **FAN**

______ • 1-HV-F-4B, REACTOR CNTMTS HV PURGE **SUPPLY** FAN

5.2.5 Close containment purge supply isolation valves:

______ ______ • **1-HV-MOV-100A, CONT** PURGE SUPPLY **VALVE** (inside valve)

______ ______ • 1-HV-MOV-100B, CONT PURGE SUPPLY VALVE (outside valve)

5.2.6 Ensure containment purge exhaust fans are secured:

______ • 1-IW-F-SA, 5A REACTOR **CNTMT** HV PURGE EXHAUST FAN

______ • 1-HV-F-5B, 5B REACTOR CNTMT HV PURGE **EXHAUST FAN**

5.2.7 Close containment **purge** exhaust isolation valves:

______ ______ • 1-HV-MOV-100C, CONT PURGE EXH VALVE (inside valve)

• 1-HV-MOV-100D, CONT PURGE EXH VALVE (outside valve)

5.2.8 Ensure the following are closed

- 1-MOV-HV-101, CONT PURGE EXH BYPASS VALVE
- 1-HV-MOV-102, CONT PURGE RELIEF VALVE

5.2.9 IF containment purge is aligned through the iodine filters, THEN remove containment purge from the iodine filter per 0-OP-21.5, Operation of Auxiliary Building Iodine Filters.

5.2.10 IF Containment Purge will NOT be restarted under the same release permit, THEN route completed HP form for Reactor Containment Release Permit to Health Physics.

5.2.11 Have a second person perform independent verification of Steps 5.2.5, 5.2.7, and 5.2.8.

5.2.12 IF Containment Purge will NOT be restarted, THEN lock open the following breakers:

- 1-EP-BKR-1C1-1 B1, Purge Exhaust Bypass Valve Circuit Breaker, 1-MV-MOV-101
- 1-EP-BKR-1C1-1 B2, Cntmt Purge Supply Bypass Line Isol Vv Circuit Bkr, 1-HV-MOV-102
- 1-EP-BKR-1A1-1 C3, Purge Sply Fans to Reac Cntmt Isol Vv Circuit Bkr. 1-HV-MOV-100A
- 1-EP-BKR-1A1-1 C4, Reac Cntmt to Purge Exh Isol Valve Circuit Bkr, 1-HV-MOV-100C
- I-EP-BKR-1Bl-2 C3, Cntmt Supply Line Isol Vv Circuit Breaker, 1-HV-MOV-100B
- 1-EP-BKR-1B1-2 C4, Cntmt Purge Exhaust Line Isol Vv Circuit Breaker, 1-HV-MOV-100D

5.2.13 IF Containment Purge will NOT be restarted, THEN lock the following valve handwheels:

_____ _____ • 1-HV-MOV-100D, Containment Purge Return Line Isolation Valve

_____ _____ • 1-HV-MOV-101, 1-HV-MOV-100D Bypass Valve

_____ _____ • 1-HV-MOV-100B, Containment Purge Supply Lie Isol Valve

_____ _____ • 1-HV-MOV-102, Containment Purge Supply Vent Line Isol Valve

_____ _____ • 1-HV-MOV-100A, Purge Supply Fans To Reactor Containment Isol

_____ _____ • 1-HV-MOV-100C, Reactor Containment To Purge Exhaust Isolation

Completed by: _______________ _______________ . Date: _______

038EA2.09 001

Unit 2 has experienced a loss of offsite power. The subsequent reactor trip ruptured a tube in the "B" steam generator. The crew has terminated safety injection in accordance with 2-E-3, " Steam Generator Tube Rupture." The following conditions exist on Unit 2:

- RCS hot leg temperature **490°F** and stable
- CETCs are 488°F and stable
- RCS pressure 900 psig
- RCS cold leg temperature is 488°F
- The following SG parameters exist:

SG	Pressure	Level
"A"	**530** psig and stable	9% NR
"B"	900 psig and stable	70% NR
"C"	528 psig and stable	8% NR

Based on the above plant parameters all natural circulation indications listed in Attachment 1 of **2-E-3**, "Steam Generator Tube Rupture," are __________

A✓ not met because RCS cold leg is not at saturation temperature for "A" and "C" steam generator pressures
B. met because all parameters indicate natural circulation
C. not met because "A" and "C" steam generators are not above the level required for heat sink in the EOP's
D. not met because CETC temperatures are not decreasing.

A. This is the correct answer. The examinee will have to know the natural circulation criteria listed in Attachment 1 of 2-E-3, "Steam Generator Tube Rupture." They will then have to use Steam Tables to arrive at the conclusion.

B. This is incorrect. The examinee could choose this answer. Four of five natural circulation criteria are met. If the examinee doesn't use the steam tables correctly they could arrive at this answer.

C. This is incorrect. The examinee could choose this answer because 11% is considered adequate level in the EOP's to consider it a heat sink. A heat sink is required for natural circulation. The attachment doesn't use this parameter but if all of the Attachment 1 criteria are met a heat sink has been established.

D. This answer is incorrect. Examinee may think CETC temperatures need to be decreasing to have natural circulation.

Ability to determine and interpret the following as they apply to SGTR: Existence of natural circulation, using plant parameters.

(CFR: 41.10 / 43.5 / 45.13)

References:
1-E-3, " Steam Generator Tube Rupture" page 32
1-E-3," Steam Generator Tube Rupture" Attachment 1 Natural Circulation Verification
Steam Tables

Level(RO/SRO):	SRO	Tier:	1
Group:	1	Importance Rating:	4.2/4.2
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

NOTE: The following conditions support or indicate natural circulation flow.

1. ___ VERIFY NATURAL CIRCULATION PLOW	Increase dumping stem.

- RCS subcooling based on Core Exit TCs - GREATER THAN 25°F [75°F]
- SG pressures - STABLE OR DECREASING
- RCS Hot Leg temperatures STABLE OR DECREASING
- Core Exit TCs - STABLE OR DECREASING
- RCS Cold Leg temperatures - AT SATURATION TEMPERATURE POR SG PRESSURE

2. ___ CONTINUE TO MONITOR NATURAL CIRCULATION PLOW UNTIL FORCED CIRCULATION IS ESTABLISKED

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Following a Loss of all seal cooling. affected RCPs should not be started without prior status evaluation.

* *

*40.__ CHECK RCP STATUS:

a) At least one RCP - RUNNING

a) Try to start one RCP:

1) IF RVLIS upper range indication less than 95%, THEN perform the following:

- Increase PRZR level greater than 69% [75%].
- Increase RCS subcooling based on core exit TCs greater than 50 °F [100°F]
- Use PRZR heaters. as necessary to saturate the pressurizer water.

2) Try to start one RCP using 1-OP-5.2. REACTOR COOLANT PUMP STARTUP AND SHUTDOWN

IF an RCP sannot be started. THEN do the following:

- Initiate Attachment 1 to verify natural circulation.
- GO TO Step 41.

b) Stop all but one RCP

Given the following conditions:

- A small-break LOCA occurred
- RCPs were tripped as required by EOPs
- RCS pressure is 1650 psig
- Wide-range Tcs are 508°F and slowly decreasing
- Wide-range Ths are 525°F and slowly decreasing
- CETCs are 530°F and stable
- S/G narrow-range levels are approximately 15%
- S/G pressures are 715 psig and slowly decreasing
- Containment radiation peaked at 120,000 R/hr and is now 30,000 R/hr
- Containment pressure peaked at 22 psia and is now 15 psia

In accordance with 1-ES-1.2, Post-LOCA Cooldown and Depressurization. the requirements for natural circulation:

A. are met.

B. are not met, since there is inadequate subcooling

C. are not met, since CETCs are not decreasing.

D. are not met, since S/G parameters are not satisfied.

Answer: A

The following plant conditions exist.

- A small break LOCA has occurred
- The crew responded in accordance with **EOPs** and tripped the **RCPs** when required
- The crew is currently in 1-ES-1.2, "Post LOCA Cooldown and Depressurization"
- RCS pressure is 1490 psig
- Wide range Tcs are 505° F and slowly decreasing
- Wide range Ths are 515° F and slowly decreasing
- CETCs are 581° F and stable
- Containment pressure is 10 psia
- Containment radiation is 5.0 X 10E+1 R/hr
- SG narrow-range levels are being maintained at a minimum of 40%
- SG pressures are 715 psig and decreasing slowly

According to 1-ES-1.2, the requirements for natural circulation ________.

A. are not met since there is inadequate *subcooling*

B. are not met since **CETCs** are not decreasing

C. are not met since SG parameters are not satisfied

D. are met

Answer: A

058AG2.4.4001

Unit 1 is at 100% power when DC bus 1-1 de-energizes. Subsequently, a loss of offsite power occurs.

Which ONE of the following actions should be directed by the Unit Supervisor?

A✓ Re-energize 1H Emergency bus **and** DC bus 1-1 in accordance with 0-AP-10," Loss of Electrical Power."

B. Re-energize **only** DC bus 1-1 in accordance with 0-AP-10," Loss of Electrical Power."

C. Re-energize 1H Emergency bus in accordance with I-ECA-0.0, Attachment IV, "Attempting to Restore Power to 1H (1J) Emergency Bus."

D. Re-energize 1H **and** 1J busses in accordance with 1-ECA-0.0, Attachment IV, "Attempting to Restore Power to 1H(1J) Emergency bus."

A. This is the correct answer. The loss of DC Bus 1-1 wil take away control power to the breaker for 1H Emergency Bus. This means on a loss of offsite power, the IH Emergency Bus will not have power from the EDG.

B. This answer is incorrect. You do use AP-10 to re-energize DC Bus 1-1 but you also need it to re-energize 1H Emergency Bus. ECA 0.0 is used for a loss of all AC. 1J Emergency Bus is energized from its EDG. The examinee could choose this answer if they fail to make the connection between a loss of DC Bus 1-1 and control power for the EDG.

C. This answer is incorrect. ECA 0.0 is only entered if both emergency buses are de-energized. The attachment name implies it is used for both buses. This makes this selection plausible.

D. This answer is incorrect. ECA 0.0 is only used during a loss of all AC. 1J Emergency Bus is powered. Examinee may not be able to recall which vital bus supplies control power. They could choose both.

Loss of DC Power
Ability io recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

This is a new question.

References:
NCRODP module 35 page 10
AP-10 " Loss of Elecrical Power"
ECA 0.0 Attachment IV, "Attempting to Restore Power to 1H(1J) Emergency Bus"
E-0 Step 3 RNO

QUESTIONS REPORT
for sroquestions

Level(RO/SRO):	SRO	Tier:	I
Group:	1	Importance Rating:	4/4.3
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

To provide electrical safety during periods of maintenance. each 4160V ac breaker has the following thres interlocks:

1. racking screw interlock.
2. breaker lock-open interlock, and
3. breaker closed interlock.

The racking screw interlock is a mechanical device that protrudes around the racking screw, It must be rotated 45 degrees to allow the racking screw to be turned. Before the racking screw interlock can be turned to disengage from the racking screw, the breaker must be tripped and the breaker lock-open interlock must not be pulled out. Once the racking screw interlock is rotated, the breaker cannot be closed. The racking screw interlock automatically engages when the breaker is in the connect, test, and disconnect positions.

The breaker lock-open interlock is operable in only the test and disconnect positions. When pulled out, it performs the following functions:

1. Mechanically prevents the racking-screw interlock from being disengaged, thus preventing the breaker from being racked to any other position;
2. mechanically prevents the breaker closing; and
3. provides a means for plaang a padlock on the breaker for additional personnel safety while work is being done.

The breaker lock-open interlock cannot be pulled out when the breaker is in the connect position, closed, or when the racking screw interlock is in the disengage position.

The breaker closed interlock mechanically prevents the racking screw interlock from being turned and the breaker lock-open interlock from being pulled out when the breaker is closed.

The 4160V ac breakers are operated by 125V dc power supplied by the battery chargers under normal conditions and supplied by the batteries under loss of power conditions. The power for the H bus breakers comes from battery 1-I (2-1) and the J bus breakers are powered from battery 1-III (2-111). The batteries for the H and J buses are in locked rooms for security. Control power is fused by two sets of fuses inside the breaker cabinets. One is for closing power and the other is for trip power. The trip power fuses also supply the indicator lights.

480V ac Breakers. The breakers in the 480V ac Emergency Bus 1H and 1J panels operate similarly to the 4160V ac breakers described above. Each 480V ac circuit breaker in Emergency Bus 1H and 1J is housed in a cubicle totally enclosed by metal.

The breakers have four positions that are identical to their 4160V ac counterparts, as follows:

1. conned.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE:	If 1557 was closed prior to a loss of 1H bus control power, then 1-CH-P-1C will continue to run. If 1557 is subsequently opened. then 1557 cannot be reclosed.	
NOTE:	During an emergency. if no smoke. fire. or obvious damage exists. then one attempt to re-energize the bus may be made.	
1. ___	LOCALLY CHECK DC BUS 1-I (1-EP-CB-12A) AND BATTERY 1-I (1-BY-B-01):	
	a) Battery - INTACT	a) Have Electrical Department disconnect battery from bus.
	b) Bus - INTACT AND ABLE TU BE ENERGIZED	b) GO TO Step 6.
2. ___	LOCALLY CHECK BATTERY CHARGERS 1 I (1-BY C-2) AND 1C-I (1-BY-C 3):	GO TO Step 6.
	a) 1 EP-KCC 1H1-4 (1-EP-MC 41) - ENERGIZED	
	b) Battery Chargers - INTACT AND ABLE TO BE ENERGIZED: • 1-I (1 BY-C-2) OR • 1C I (1-BY C 3)	
3. ___	PLACE: BATTERY CHARGER 1-I (1-BY-C 2) IN SERVICE USING 1 OP-26.4.1, MAIN STATION BATTERY CHARGERS 1 I AND 1 II OPERATION	Place Battery Charger 1C-I (1-BY C-3) in service using 1-OP 26.4.3, MAIN STATION SWING BATTERY CHARGERS 1C I AND 1C-II OPERATION.
4. ___	CHECK DC BUS 1-I (1-EP-CB-12A) - ENERGIZED	GO TU Step 6.

STEP	~~ACTION/EXPECTED~~ RESPONSE	RESPONSE NOT OBTAINED
5. ___	RETURN TO PROCEDURE IN EFFECT	
6. ___	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-I (1-EP-CB-12A):	
	a) Verify Generator Output Breaker 6-12 - OPEN	a) IF NOT open within 30 seconds of unit trip. THEN manually open G-12.
	b) As required. locally operate breakers on the following busses using 0-MOP-26.11. 4160-VOLT BREAKER LOCAL MANUAL OPERATION. or 0-MOP 26.10. 480-VOLT BREAKER LOCAL **MANUAL** OPERATION. because of loss of control power: • 4160-Volt Bus 1C (1-EP-SW-03) • 4160-Volt Bus 1H (1-EE-SW-01) • 4160-Volt Bus 2G (2-EP-SW-04) • 480-Volt Bus 1C1 (1-EP-SS-07) • 480-Volt Bus 1C2 (1-EP-SS-04) • 480-Volt Bus 1H (1-EE-SS-01) • 480-Volt Bus 1H1 (1-EE-SS-03) • 480-Volt Bus 162 (1-EP-SS-11) • 480-Volt Bus 1G3 (1-EP-SS 12) • 480-Volt Bus 2G2 (2-EP-SS-09) • Exciter Field Breaker • 15F1, 4160 Volt 1F Transfer Bus Feed	
	c) Feed SG with AFW because of loss of MFW Reg Valve and MFW Bypass Reg Valve	

(STEP 6 CONTINUED OM NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-I (1-EP-CB-12A) (continued):	
	d) Check SG PORVs - CONTROLLING IN AUTO OR MANUAL	d) Place SG PORVs in AUTO because of loss of Condenser Steam Dump.
	e) Check Blender - IN OFF	e) Place Blender in OFF because of loss of flow indication.
	f) Check Terry Turbine REQUIRED TO BE IN SERVICE	f) Locally close 1-MS 268. inlet to 1-MS-TV-111A, because of 1-MS TV 111A failed **open.**
	g) Establish continuous fire watches within 1 hour of bus loss for the following areas: • 1H EDG Room • 2H EDG Room • 1J EDG Room • 2J EDG Room • Unit 1 **Cable** Vault • Unit 1 Cable Tunnel • Unit 2 Cable Vault • Unit 2 Cable Tunnel • 1-HV-FL-3A. Charcoal Filter Bank • 1-HV-FL 3B. Charcoal Filter Bank • Control Room Under floor	
	h) Check PRZR Level - *28%* OR LESS	h) As required. place excess letdown in service because of loss of normal letdown. using 1-OP-8.5, OPERATION OF EXCESS LETDOWN.
7. ___	CHECK DC BUS i-I (1-EP CB 12A) CAN BE ENERGIZED	Continue efforts 'to restore. RETURN TO Step 6.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. ___	RETURN TO STEP 1	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3] __	VERIFY BOTH AC EMERGENCY BUSSES ENERGIZED	Do the following: a) IF no AC Emergency Bus is energized. THEN immediately restore power to at least one AC Emergency Bus. IF power cannot be restored. THEN GO TO 1 ECA-0.0, LOSS OF ALL AC POWER. STEP 1. b) Try to restore power to de-energized AC Emergency Bus using 0-AP 10, LOSS OF ELECTRICAL POWER. as time permits. Continue with Step 4.

1. IF attempting to start 1H (1J) Emergency Diesel Generator, THEN do the following:

   a) Place 1H (1J) EDG Mode Selector switch to MANUAL-LOCAL.

   b) Push Emer Gen 1H (1J) Alarm & Shutdown Reset button.

   c) Verify 1H (1J) Shutdown Relay Status Light is LIT.

   d) IF neither 1H nor 1J Shutdown Relay Status Light is LIT. THEN GO TO 1-ECA-0.0. LOSS OF ALL AC POWER. STEP 9.

   e) Wait 1 minute.

   f) WHEN one minute has elapsed. THEN place 1H (1J) EDG Mode Selector switch to MANUAL-REMOTE.

   g) Verify 1H (1J) EDG starts.

   h) IF neither 1H nor 1J EDG starts, THEN GO TO 1 ECA-0.0. LOSS OF ALL AC POWER. STEP 9.

2. ~~IF~~ attempting to energize 1H (1J) 4160-Volt Emergency Bus. THEN do the following:

   a) Place the 4160V Emer Gen Supply Feed To Bus 1H (1J) Synchronizing 15H2 (15J2) switch to ON.

   b) Verify Incoming Voltage is indicated.

   c) IF voltage is NOT indicated. THEN push Exciter Reset button AND control voltage.

   d) IF voltage is still NOT indicated, THEN turn off Synch Scape.

   e) IF neither 1H nor 1J incoming voltage is indicated. THEN GO TO 1-ECA-0.0. LOSS OF ALL AC POWER. STEP 9.

   f) Verify 15H2 (15J2) is closed.

   g) IF 15H2 (1552) is NOT closed. THEN manually close 15H2 (15J2) and turn off Synch Scope.

   h) IF neither 15H2 nor 15J2 can be closed. THEN turn off Synch Scope and GO TO 1-ECA-0.0, LOSS OF ALL AC POWER. STEP 9.

- END

061A2.02 001

Unit 1 was at 100% power when a trip occurred due to a loss of off-site power. The following conditions exist at this time:

- 1-MS-TV-111A, Steam Supply to Turbine-Driven AFW Pump is isolated for diaphragm replacement
- The 3B AFW Pump will not start
- The air supply line to 1-MS-TV-1 11B, Steam Supply to Turbine-Driven: bias broken off.

The crew is currently in I-E§-0.1, "Reactor Trip Response," and the Unit Supervisor desires to feed all three steam generators with auxiliary feedwater.

Which ONE of the following procedures should be used, including the proper flow path?

At' 1-AP-22.3, " boss of 3B AFW Pump." This procedure will use 1-FW-P-2 to feed "A" and "B" steam generators through the MOV header. 1-FW-P-3A will feed the "C" steam generator through the HCV header.

B. 1-AP- 22.6, " Loss of FW- P-2 and One Motor-Driven AFW Pump." 1-FW-P-3A will feed all three steam generators through the HCV header.

C. 1-AP-22.3, " boss of 38 AFW Pump." I-FW-P-2, Turbine-Driven AFW Pump will feed "A" steam generator through 1-FW-MOV-100D and the "B" steam generator through the MOV header. 1-FW-P-3A will feed the "C" steam generator through the HCV header.

D. 1-AP- 22.6, " Loss of FW- P-2 and One Motor-Driven AFW Pump." 1-FW-P-3A will feed all three steam generators through the MOV header,

A. This is the correct answer. TV-111A/B are fail open valves. The only AFW pump not available is the 3B. Attachment 4 will align the Turbine-Driven AFW Pump to the MOV header.

B. This is incorrect. The crew will not lose the Turbine-Driven AFW Pump. If examinee doesn't realize TV-11 1B is a fail open valve they will choose this answer.

C. This answer is incorrect. The Turbine-Driven AFW Pump. will not be aligned both through the MOV header and through 100D. Examinee might pick this bemuse the Turbine-Driven AFW Pump is normally aligned to the A steam generator through 100D.

D. This is incorrect. The crew will not lose the Turbine-Driven AFW Pump. If examinee doesn't realize TV-1115 is a fail open valve they, will choose this answer.

Ability to (a) predict the impacts of the following on the AFW system and (b) based on those predictions, use procedures to correct, control, or mitigate the Consequences of those abnormal operation:
Loss of air to steam supply valve

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

This is a new question

References:
Objective 5978 from study guide on Auxiliary Feedwater
1-AP-22.6 Loss of Turbine Driven AFW Pump step 4
1-AP-22.6 Loss of Turbine Driven AFW Pump Attachment 4
1-AP-22.3 Loss of 35 Motor-Driven AFW Pump step 5
1-AP-22.3 Loss of 3B Motor-Driven AFW Pump Attachment 4

Level(RO/SRO):	SRO	Tier:	2
Group:	1	Importance Rating:	3.2/3.6
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

Topic 2.11: Steam supply valves
5970

2.11a. Objective

List the following information associated with steam supply valves TV-MS-111A and 111B.

- Locations from which the steam supply valves can be operated
- Conditions which will cause the TURBINE DRIVEN AUXILIARY FEEDWATER PUMP TRAIN A(B) NON-AUTOMATIC CONTROL alarm to actuate
- Position in which the steam supply valves will fail on a loss of instrument air
- Function of the small air flasks located near steam supply trip valves
- Methods which can be used to fail these valves open, if required

2.11b. Content

1. AFW pump steam supply valves 1-MS-TV-111A and 111B can be operated from either:
   1.1. Benchboard 1-1 in the MCR.
   1.2. Aux. S/D panel.

2. The TURBINE DRIVEN AFW PUMP TRAIN A(B) NON-AUTO CONTROL annunciator will alarm if the associated control switch (1-MS-TV-111A or B) is taken out of AUTO.

3. The steam supply valves will fail open following a loss of instrument air or SOV power

4. The small air Rasks located near 1-MS-TV-111A and 111B provide a limited amount of air to force the TVs open during a loss of instrument air pressure.

5. The TVs can be failed open by opening the SOV power supply breakers or by isolating and venting the instrument air supply lines.

Topic 2.11: Steam supply valves
5970

2.11a. Objective

List the following information associated with steam supply valves TV-MS-111A and 111B.

- Locations from which the steam supply valves can be operated
- Conditions which will **cause** the TURBINE DRIVEN AUXILIARY FEEDWATER PUMP TRAIN A(B) NON-AUTOMATIC CONTROL alarm to actuate
- Position in which the steam supply valves will fail on a **loss** of instrument air
- Function of the small air flasks located near steam supply trip valves
- Methods which can be used to fail these valves open, if required

2.11b. Content

1. AFW pump steam supply valves 1-MS-TV-111A and 111B can be operated from either:
   1.1. Benchboard 1-1 in the MCR.
   1.2. Aux. S/D panel.

2. The TURBINE DRIVEN AFW PUMP TRAIN A(B) NON-AUTO CONTROL annunciator will alarm if the associated control switch (1-MS-TV-111A or B) is taken out of AUTO.

3. The steam supply valves will fail open following a loss of instrument air or SOV power.

4. The small air flasks located near I-MS-TV-111A and 111B provide a limited amount of air to **force** the TVs open during a loss of instrument air pressure.

5. The TVs can be failed open by opening the SOV power supply breakers or by isolating and venting the instrument air supply lines.

Topic 2.11: Steam supply valves
5970

2.11a. Objective

List the following information associated with steam supply valves TV-MS-111A and 111B.

- Locations from which the steam supply valves can be operated
- Conditions which will cause the TURBINE DRIVEN AUXILIARY FEEDWATER PUMP TRAIN A(B) NON-AUTOMATIC CONTROL alarm to actuate
- Position in which the steam supply valves will fail on a loss of instrument air
- Function of the **small** air flasks located near steam supply trip valves
- Methods which can be used to fail these valves open, if required

2.11b. Content

1. ARM pump steam supply valves 1-MS-TV-111A and 111 B can be operated from either:
   1.1. Benchboard 1-1 in the MCR.
   1.2. Aux. S/D panel.

2. The TURBINE DRIVEN AFW PUMP TRAIN A(B) NON-AUTO CONTROL annunciator will alarm if the associated control switch (1-MS-TV-111A or B) is taken out of AUTO.

3. The steam supply valves will fail open following a loss of instrument air or SOV power.

4. The small air Rasks located near 1-MS-TV-111A and 114B provide a limited amount of air to force the TVs open during a **loss** of instrument air pressure.

5. The TVs can be failed open by opening the SOV power supply breakers or by isolating and venting the instrument air supply lines.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.__	CHECK ECST LEVEL - GREATER THAN 40%	Initiate 1-AP 22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN TK-1.

NOTE: The AFW lineup drawings of Attachments 3 and 4 should he retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.__	PERFORM ATTACHMENT 4 TO ALIGN THE ACV HEADER FOR FEEDING ALL SGs	Perform Attachment 3 to align the MOV header for feeding all SGs. GO TO Step 7.
* 5.__	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING IICV HEADER: • 1 FW HCV 100A, AFW HCV HEADER TO A SG • 1-FW HCV-100B, AFW HCV HEADER TO B SG • 1 FW-HCV 100C, AFW HCV HEADER TO C SG	
6.__	GO TO STEP 8	
* 7.__	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING THE MOV HEADER: • 1 FW-MOV 100A, AFW MOV HEADER TO A SG • 1 FW-MOV 100B, AFW MOV HEADER TO B SG • 1-FW-MOV 100C, AFW MOV HEADER TO C SG	

NUMBER 1 AP-22.6	ATTACHMENT TITLE ALIGNING THE OPERABLE MOTOR DRIVEN APW PUMP TO THE HCV HEADER	REVISION 3
ATTACHMENT 4		PAGE 1 of 3

NOTE: Some of the AFW System valves have admin locks.

1. Have the CRO close the following AFW Valves:

   ___ • 1-FW-HCV-100A, AFW HCV HEADER TO A SG

   ___ • 1-FW-MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG

   ___ • 1-FW-HCV-1COB. AFW HCV HEADER TO B SG

   ___ • 1-FW-MOV-100B, APW MOV HEADER TO B SG

   ___ • 1-FW-HCV-100C, AFW HCV HEADER TO C SG

2. ___ IF 1-FW-P-3A is the operable pump. THEN unlock and close 1-FW-184. Discharge Valve €or 1-FW-P-3B to MOV Header.

3. IF 1-FW P-3R is the operable pump, THEN do the following:

   ___ a) Unlock and close 1 FW 172, Discharge Valve for 1 FW P 3A to HCV Header

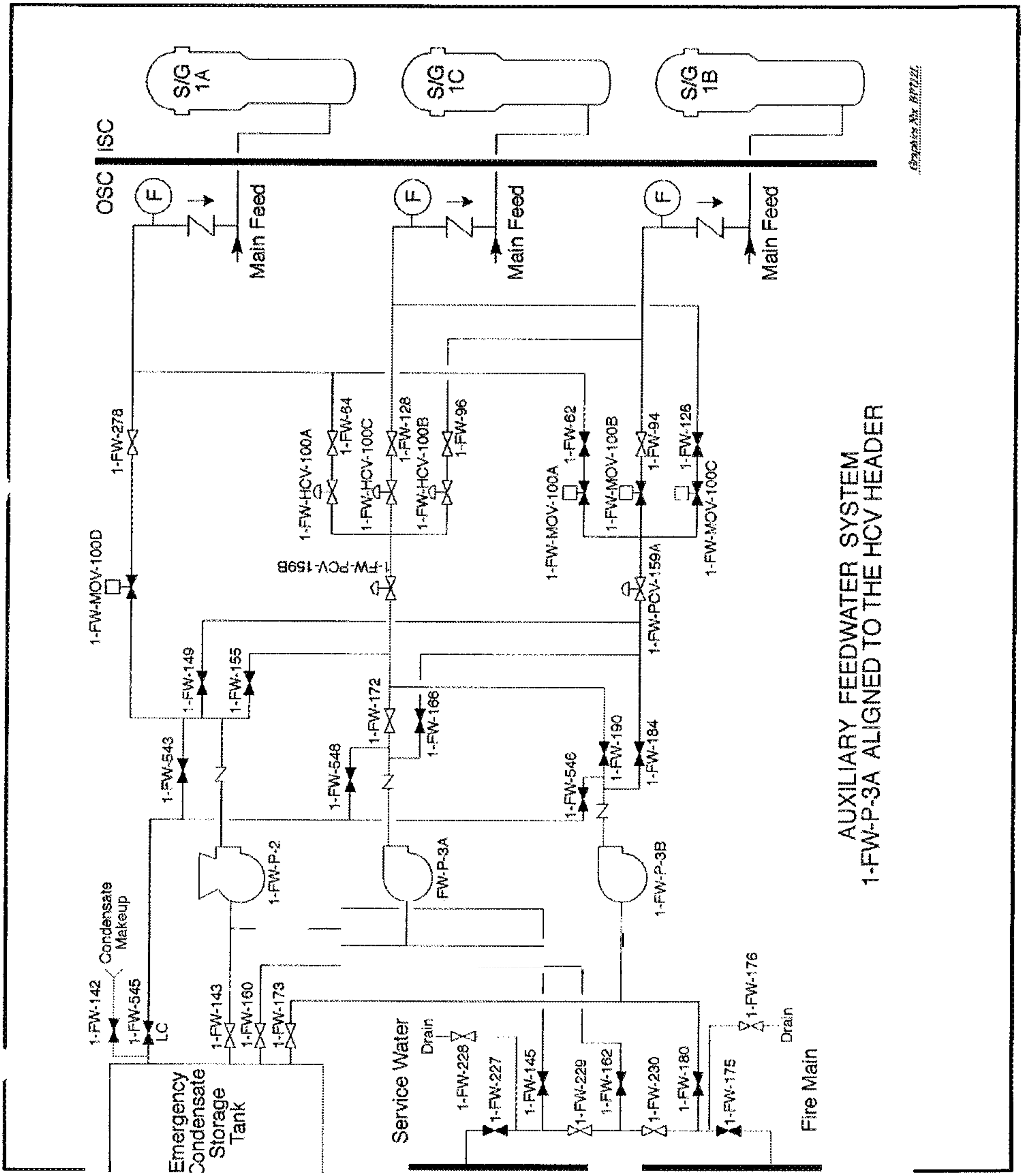
   ___ b) Unlock and open 1-FW-190. 3B Mtr Drvn AFW Pp to HCV Hdr Inlet Isol Valve

4. Locally unlock and open the following valves (located in the Motor-Driven AFW Pumphouse):

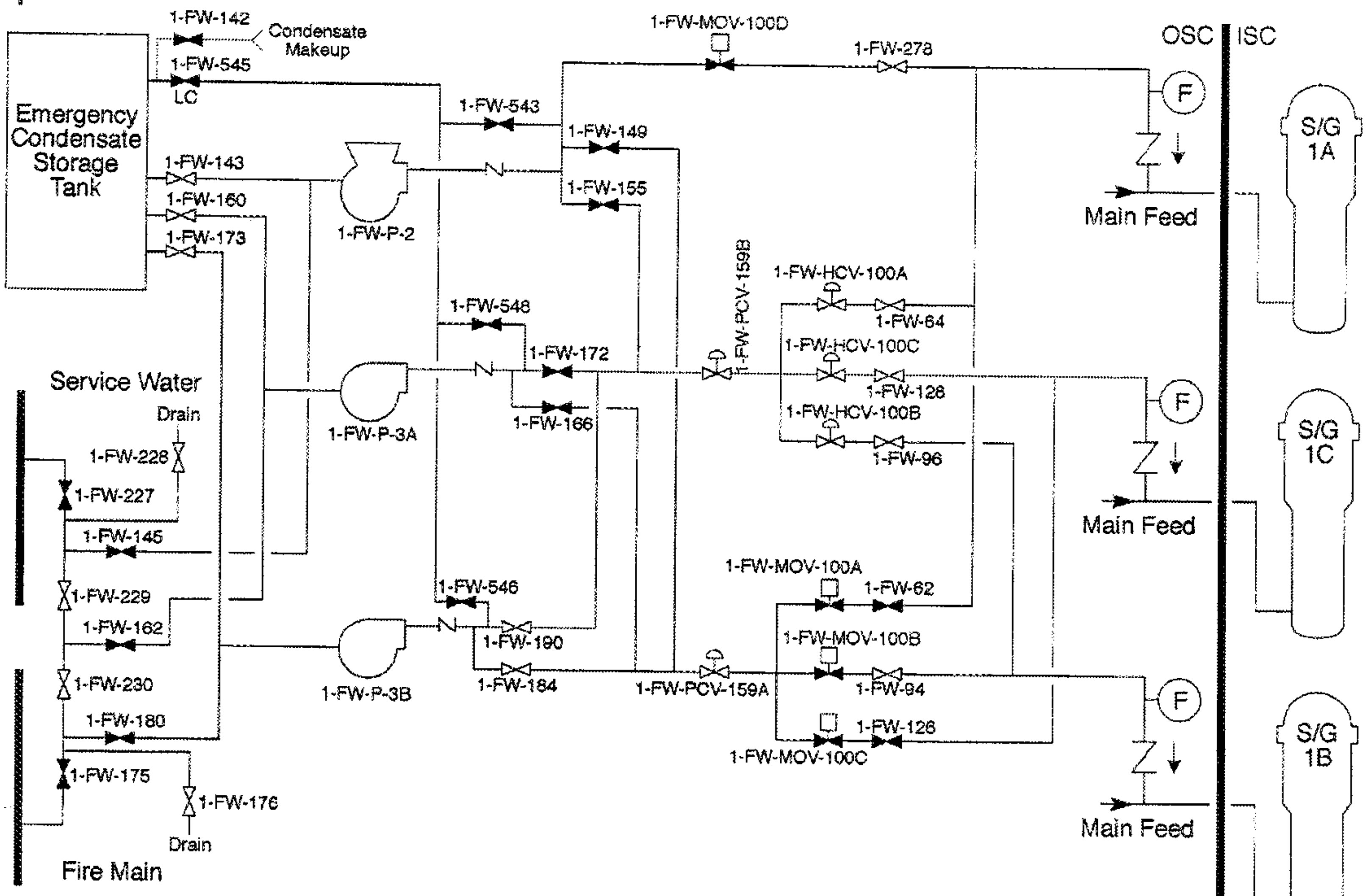
   ___ • I-FW-64. AFW A HCV 1-FW-HCV-100A Outlet Isolation Valve

   ___ • 1-FW-96. AFW B HCV 1 FW HCV-100B Outlet Isolation Valve

5. — Notify the Control Room that Attachment 4 is complete and to return to 1-AP-22.6. step in effect.



AUXILIARY FEEDWATER SYSTEM
1-FW-P-3A ALIGNED TO THE HCV HEADER



AUXILIARY FEEDWATER SYSTEM
1-FW-P-3B ALIGNED TO THE HCV HEADER

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.__	CHECK TURBINE DRIVEN AFW PUMP - RUNNING	Manually start pump: a) **Open** 1-FW-MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG. b) Place Steam Supply Valves for Turbine Driven AFW Pump in OPEN: • 1-MS-TV-111A • 1-MS TV 111B
3.__	CHECK 1-FW-P 3A. MOTOR-DRIVEN AFW PUMP RUNNINC	Manually start 1 FW P-3A. <u>IF</u> 1 FW P 3A cannot be started, <u>THEN</u> GO TO 1-AP-22.4. LOSS OF BOTH KOTOR-DRIVEN AFW PUMPS.
4.__	CHECK ECST LEVEL GREATER THAN 40%	Initiate 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1.

<u>NOTE</u>: The AFW lineup drawings of Attachments 3 and 4 should be retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.__	PERFORM ATTACHMENT 4 TO ALIGN THE MOV HEADER FOR FEEDING THE A AND B SGs	Perform Attachment 3 to align HCV Header for feeding all SGs. GO TO Step *8*.
• 6.__	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING: • 1 FW MOV 100A for A SG • 1-FW-MOV-100B for B SG • 1-FW-HCV 100C for C SG	
7.__	GO TO STEP 9	

NOTE: Some of the AFW System valves have admin locks.

1. Nave the CRO close the following AFW Valves:

___ • 1-FW-MOV-100A, AFW MOV HEADER TO A SG

___ • 1 FW-MOV-100B, AFW MOV HEADER TO B SG

2. Locally perform the following valve lineup (located in the Motor-Driven AFW Pumphouse):

___ a) Unlock and close I-FW-184. Discharge Valve €or 1-FW P 3B to MOV Header.

___ b) Unlock and open 1-FW-149. Turb Drvn AFW Pump to S/G MOV Hdr Disch Isol Valve.

___ c) Unlock and open 1-FW 62. AFW A MOV 1-FW MOV-100A Outlet Isolation Valve.

3. Have the CRO open the following AFW Valves:

___ • 1-FW MOV-100A, AFW MOV HEADER TO A SG

___ • 1-FW-MOV-100B, AFW MOV HEADER TO E SG

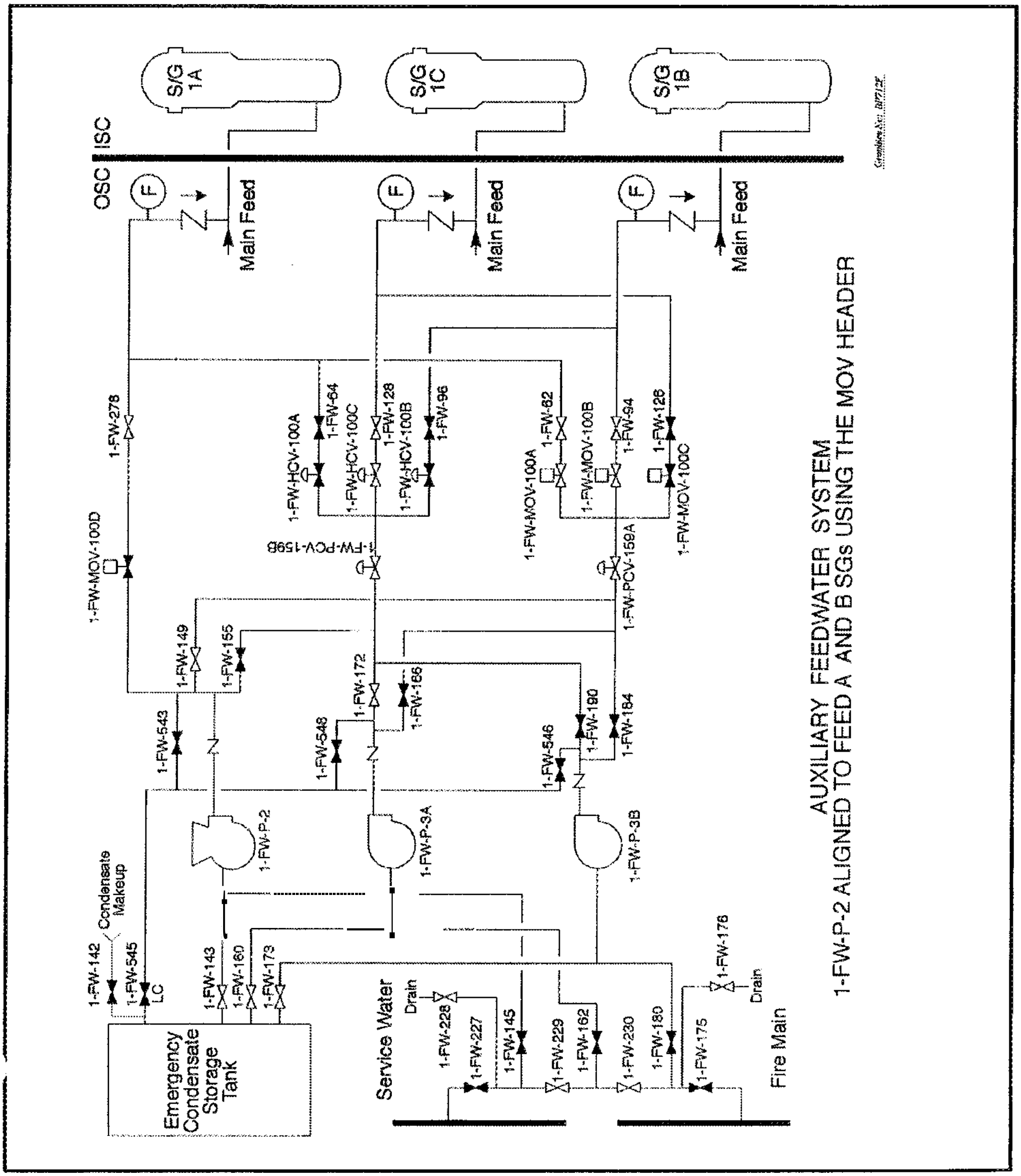
4. Have the CRO close the following AFW Valves:

___ • 1-FW-MOV 100D, TURBINE DRIVEN AFW PUMP TO A SG

___ • 1 **FW-HCV** 100B, AFW HCV HEADER TO B SG

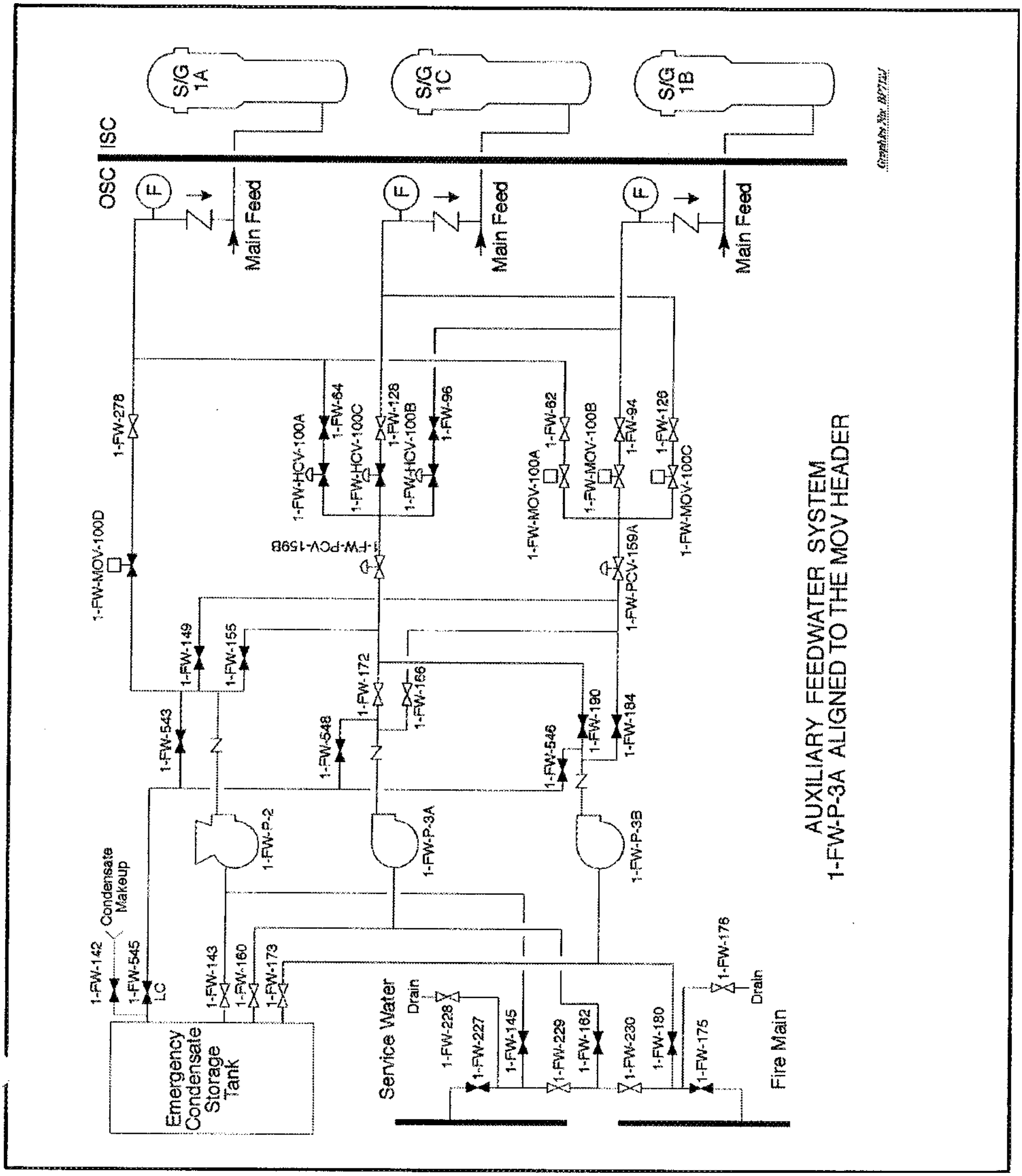
5. — Notify the Control Room that Attachment 4 is complete and Lo return to 1-AP-22.3. step in effect.

NUMBER	ATTACHMENT TITLE	REVISION
1-AP-22.3	ALIGNING TURBINE-DRIVEN AFW PUMP TO FEED A AND B SGs USING TME MOV HEADER	10
ATTACHMENT 4		PAGE 2 of 2



AUXILIARY FEEDWATER SYSTEM
1-FW-P-2 ALIGNED TO FEED A AND B SGs USING THE MOV HEADER

NUMBER	ATTACHMENT TITLE	REVISION
1-AP-22.6	**ALIGNING THE OPERABLE MOTOR-DRIVEN AFW PUMP TO THE MOV HEADER**	3
ATTACHMENT 3		PAGE 2 of 3



VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-22.3	LOSS OF 1-FW-P-3B MOTOR-DRIVEN AFW PUMP (WITH FOUR ATTACHMENTS)	10
		PAGE 1 of 7

PURPOSE

To provide instructions for placing an AFW source in service when the B Motor-Driven AFW Pump is inoperable.

ENTRY CONDITIONS

This procedure is entered when AFW flow is required to the B SG and any of the following conditions exists:

- 1-FW-P-3B is inoperable. or
- The MOV header is inoperable. or
- Annunciator Panel "F" C-5, AUX FD PP 3A-3B AUTO TRIP. is LIT.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL: APPROVAL - ON FILE	DATE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- When ECST level decreases to 40%. then 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1. should be initiated to provide an alternate water source to the AFW Pumps.
- To prevent heating of the ECST above 120°F, the AFW Pumps should not be run on recirc for extended periods of time.
- To prevent possible degradation to the AFW Pump. the amount of time spent on minimum recirc flow should be minimized.
- To prevent lifting relief valve 1-FW-RV-100 when reducing feed flow any MOV or HCV supplied by the Turbine-Driven AFW Pump should be slowly throttled.
- To prevent lifting relief valve 1-FW-RV-100, a discharge flowpath must be available to feed an SG from the Turbine-Driven AFW Pump.

* *

NOTE:
- Normal PRZR spray should be isolated from any RCP that is stopped.
- The C RCP provides the best PRZR spray capability. The A RCP also provides PRZR spray capability.
- If AFW Pumps are lost due to loss of control from the Control Room, then evaluate using 1-AP-20. OPERATION FROM THE AUXILIARY SHUTDOWN PANEL. to start the affected AFW Pumps from the Auxiliary Shutdown Panel.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK MAIN FEEDWATER - IN SERVICE	Do the following: a) Stop all but one RCP. b) Initiate attempts to restore Main Feedwater.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.__	CHECK TURBINE-DRIVEN AFW PUMP - RUNNING	Manually start pump: a) Open 1-FW-MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG. b) Place Steam Supply Valves for Turbine Driven AFW Pump in OPEN: • 1-MS-TV-111A • 1-MS-TV-111B
3.__	CHECK 1-FW-P-3A. MOTOR RRIVEN AFW PUMP - RUNNING	Manually start 1-FW P 3A. <u>IF</u> I-FW-P-3A cannot be started. <u>THEN</u> GO TO 1-AP-22.4. LOSS OF BOTH MOTOR-DRIVEN AFW PUMPS.
4.__	CHECK ECST LEVEL - GREATER THAN 40%	Initiate I-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1.

<u>NOTE</u>: The AFW lineup drawings of Attachments 3 and 4 should he retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.__	PERFORM ATTACHMENT 4 TO ALIGN THE MOV HEADER FOR FEEDING THE A AND B SGs	Perform Attachment 3 to align HCV Header for feeding all SGs. GO TO Step 8.
* 6.__	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING: • 1-FW MOV 100A for A SG • 1 FW MOV 100B for B SG • 1-FW-HCV-100C for C SG	
7.__	GO TO STEP 9	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
• 8.___	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING: • 1-FW-HCV-100A for A SG • 1-FW-HCV-100B for B SG • 1-FW-HCV-100C for C SG	
9.___	DETERMINE IF AFW PUMPS CAN RETURN TO NORMAL DISCHARGE ALIGNMENT	Continue with other procedures and steps in effect. <u>WHEN</u> normal alignment can be established, <u>THEN</u> GO TO Step 10.
10.___	RAISE SG NARROW RANGE LEVELS TO 45% TO 50%	
11.___	STOP AFW PUMPS: a) Reset SI b) Reset AMSAC c) Place Motor-Driven AFW Pump in PTL: • 1-FW-P-3A • 1-FW-P-3B d) Place Steam Supply Valves for Turbine Driven AFW Pump (Terry Turbine) to CLOSE: • 1-MS-TV-111A • 1-MS-TV-111B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12. — MANUALLY CLOSE HCV AND MOV HEADER DISCHARGE VALVES:

- 1-FW-HCV-100A
- 1-FW-HCV I00B
- 1-FW-HCV-100C

- 1-FW-HOV-100A
- 1-FW-MOV-100B
- 1-FW-MOV-100C
- 1-FW-MOV-100D

NOTE: The AFW lineup drawing of Attachment 2 should be retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

13.__ PERFORM ATTACHMENT 2 TO RETURN THE AFW PUMPS TO A NORMAL DISCHARGE ALIGNMENT

14. — CHECK ECST LEVEL - GREATER THAN 40%

Response Not Obtained: Initiate 1-AP-22.5, LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1.

15.__ MANUALLY OPEN DISCHARGE VALVES:

- 1 FW-MOV-100D
- 1-FW-MOV-100B
- 1-FW-HCV-100C

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.__	VERIFY MAIN FEEDWATER IN SERVICE	Do the following: a) Start all AFW Pumps: • I-FW-P-3A • 1-FW-P-3B • 1-FW-P-2 b) Maintain SG Narrow Range Levels between 23% to 50%. c) GO TO Step 18.
17.__	PLACE AFW PUMPS IN AUTO: a) Place Motor-Driven AFW Pump control switches in AUTO: • 1-PW-P-3A • 1-FW P 3B b) Place Steam Supply Valves for Turbine-Driven AFW Pump (Terry Turbine) in AUTO: • 1-MS TV-111A • 1-MS TV-111B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. —	VERIFY THE FOLLOWING ANNUNCIATORS - NOT LIT • Panel "F" A-8. AFW PUMP DISCH FW-MOV 100B/D NOT FULL OPEN • Panel "F" E-8. AFW PUMP DISCH FW-MOV-100A/C NOT FULL CLOSE • Panel "F" C-5, AUX FB PP 3A 3B AUTO TRIP • Panel "F" C-6. AFW PUMP DISCH FW HCV-100C NOT FULL OPEN • Panel "F" C-7. AFW PUMP DISCI1 FW-HCV 100A/B NOT FULL CLOSE • Panel "F" C-8. AFW BUMP DISCH FW-PCV-159A/B NOT OPEN • Panel "F" D-5. AUX FD PP LOCAL CONTROL • Panel "F" D-6. TURBINE DRIVEN AFW PUMP TRAIN A NON-AUTO CONT • Panel "F" D 7, TURBINE DRIVEN AFW PUMP TRAIN B NON AUTO CONT • Panel "F" D-8. TURBINE DRIVEN AFW PUMP TROUBLE OR LUBE OIL TRBL	Refer to the applicable annunciator response procedure.
19. —	PERFORM APPLICABLE PORTIONS OF 1-OP-31.2A, VALVE CHECKOFF AUXILIARY FEEDWATER	
20. —	RETURN TO PROCEDURE IN EFFECT	

END

- UFSAR
- 11715-FM-74A
- Safety System Functional Inspection on Auxiliary Feedwater System at North Anna Power Station, May 20, 1987
- DCP 88-18. Full Flow Recirc
- Response to IEA 88-04, Potential Safety Related Pump Loss (1-C 3)
- 1-AP-20. OPERATION FROM THE AUXILIARY SHUTDOWN PANEL
- 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1
- 1-OP-31.2A, VALVE CHECKOFF AUXILIARY FEEDWATER
- DCP 92-003 1, Annunciator Window Engraving and Relocation. Unit 1
- The following EOPs reference this procedure:

  1-ECA-0.1. LOSS OF ALL AC BOWER RECOVERY WITHOUT SI REQUIRED
  1-ECA-0.2. LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED
  1 FR-H.l. RESPONSE TO LOSS OF SECONDARY HEAT SINK

NOTE: Some of the AFW System valves have admin locks.

1. Locally perform the following valve lineup (located in the Motor-Driven APW Pumphouse):

   ___ a) Open and lock 1-FW-184. Discharge Valve for 1-FW P 3B to MOV Header.

   b) Close and lock the following valves:

   ___ • 1-PW-155. Turb Drvn AFW Pump to S/G HCV Hdr Outlet Isol Valve

   ___ • 1-FW 64. AFW A HCV 1-FW-HCV-100A Outlet Isolation Valve

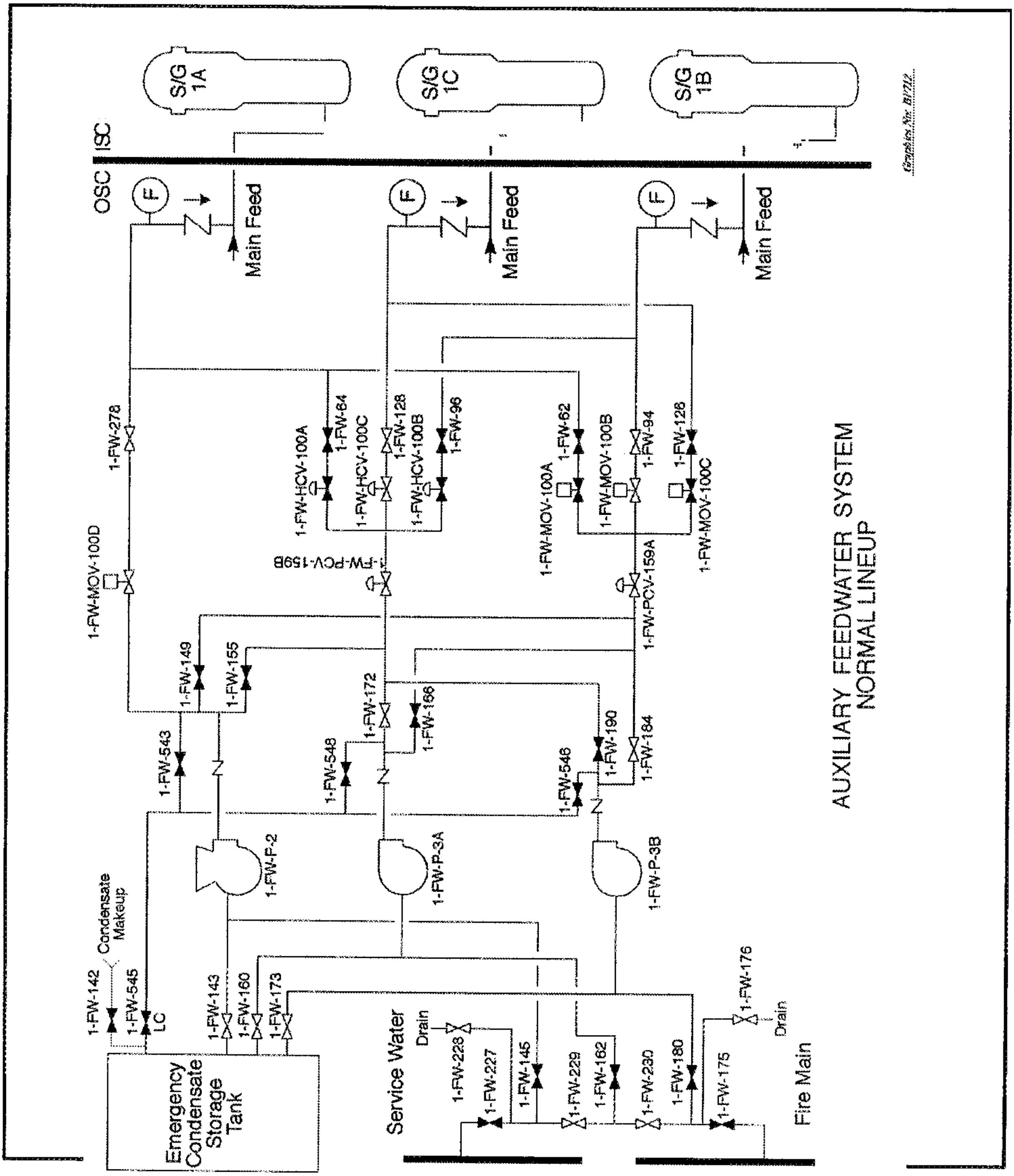
   ___ • 1-FW-96. AFW B HCV 1-PW-HCV-IOOB Outlet Isolation Valve

   ___ • I FW-149. Turb Drvn AFW Pump to S/G MOV Hdr Disch Isol Valve

   ___ • 1-FW-62. AFW A MOV 1-FW-MOV-1008 Outlet Isolation Valve

2. ___ Notify the Control Room that Attachment 2 is complete and to return 'to 1-AP-22.3. step in effect.

NUMBER	ATTACHMENT TITLE	REVISION
1-AP-22.3	RETURNING APW PUMPS TO NORMAL DISCHARGE ALIGNMENT	10
ATTACHMENT 2		PAGE 2 of 2



NOTE: Some of the AFW System valves have admin locks.

1. Have the CRO close the following AFW Valves:

__ • 1-FW-HCV-100A, AFW HCV HEADER TO A SG

__ • 1-FW-HCV-100B, AFW HCV HEADER TO B SG

2. Locally unlock and open the following valves (located in the Motor-Driven AFW Pumphouse):

__ • 1-FW 155. Turb Drvn AFW Pump to S/G HCV Hdr Outlet Isol Valve

• 1-FW 64. AFW A HCV 1-FW-HCV-100A Outlet Isolation Valve

__ • 1 FW-96, AFW B HCV 1-FW-HCV-100B Outlet Isolation Valve

3. Have the CRO open the following AFW Valves:

__ • 1-FW-HCV-100A, AFW HCV HEADER TO A SG

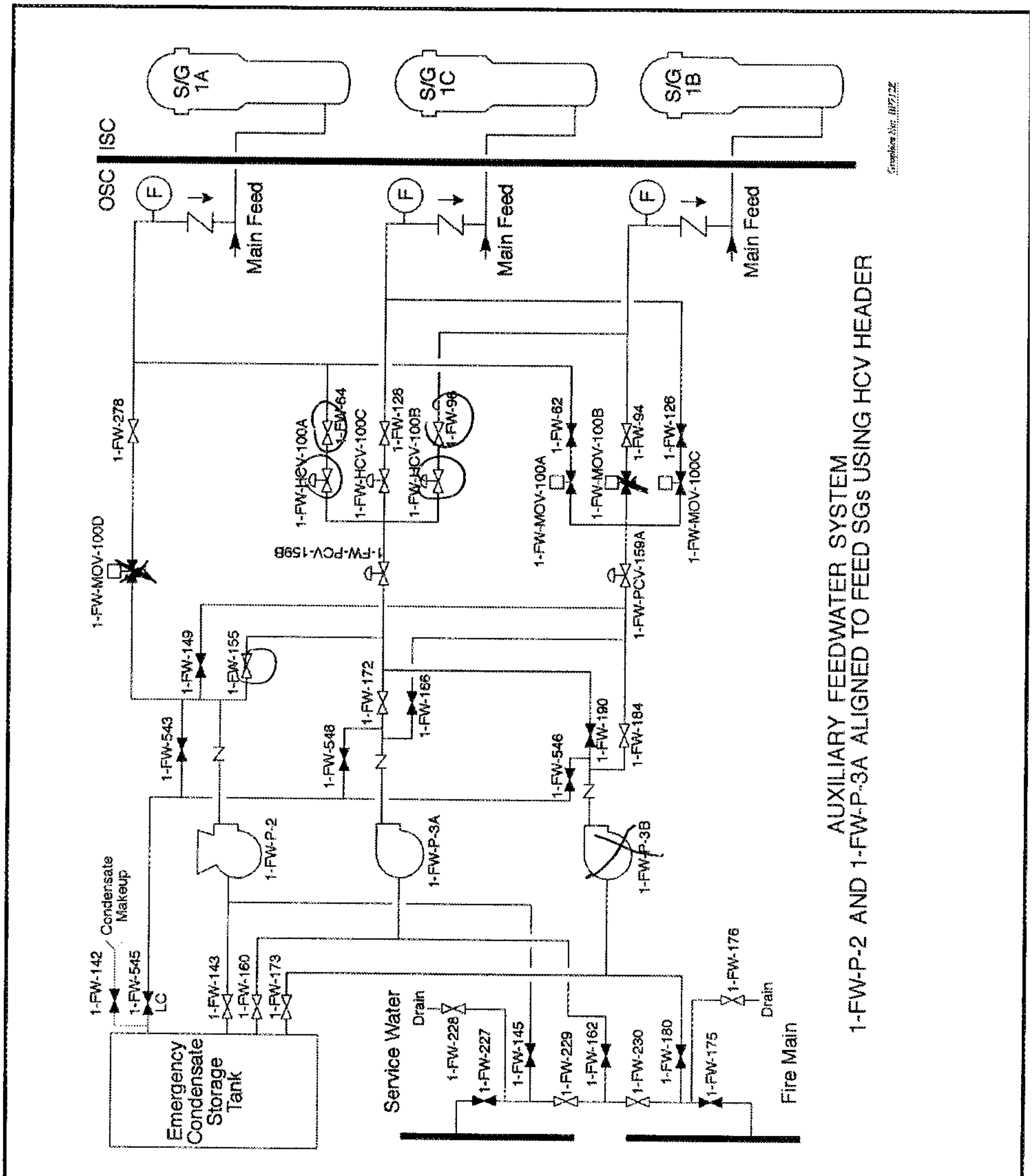
__ • 1-FW-HCV-100B, AFW HCV HEADER TO B SG

4. Have the CRO close the following APW Valves:

__ • 1-FW-MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG

__ • 1-FW MOV-100B, AFW MOV HEADER TO E SG

5. __ Notify the Control Room that Attachment 3 is complete and to return to 1-AP-22.3, step in effect.



AUXILIARY FEEDWATER SYSTEM
1-FW-P-2 AND 1-FW-P-3A ALIGNED TO FEED SGs USING HCV HEADER
S/G 1A
S/G 1C
S/G 1B
ISC
OSC
Main Feed
Main Feed
Main Feed
1-FW-278
1-FW-MOV-100D
1-FW-HCV-100A
1-FW-64
1-FW-HCV-100C
1-FW-128
1-FW-HCV-100B
1-FW-96
1-FW-PCV-159B
1-FW-62
1-FW-MOV-100A
1-FW-MOV-100B
1-FW-94
1-FW-126
1-FW-MOV-100C
1-FW-PCV-159A
1-FW-149
1-FW-155
1-FW-543
1-FW-172
1-FW-166
1-FW-548
1-FW-190
1-FW-184
1-FW-546
1-FW-P-2
1-FW-P-3A
1-FW-P-3B
Condensate Makeup
1-FW-142
1-FW-545
LC
1-FW-143
1-FW-160
1-FW-173
Emergency Condensate Storage Tank
Service Water
Drain
1-FW-228
1-FW-227
1-FW-145
1-FW-229
1-FW-162
1-FW-230
1-FW-180
1-FW-175
1-FW-176
Drain
Fire Main

NOTE: Some of the AFW System valves have admin locks.

1. Have the CRO close the following AFW Valves:

   - 1-FW-MOV-100A, AFW MOV **HEADER** TO **A** SG
   - ___ 1 FW-MOV-100B, AFW MOV HEADER TO B SG

2. Locally perform the following valve lineup (located in the Motor Driven AFW Pumphouse):

   ___ a) Unlock and close 1 FW-184, Discharge Valve for 1 FW-I-LIB to MOV Header.

   ___ b) Unlock and open 1-FW-149, Turb Drvn AFW Pump to S/G MOV Hdr Disch Isol Valve.

   ___ c) Unlock and open 1-FW-62, AFW A MOV 1-FW MOV-100A Outlet Isolation Valve.

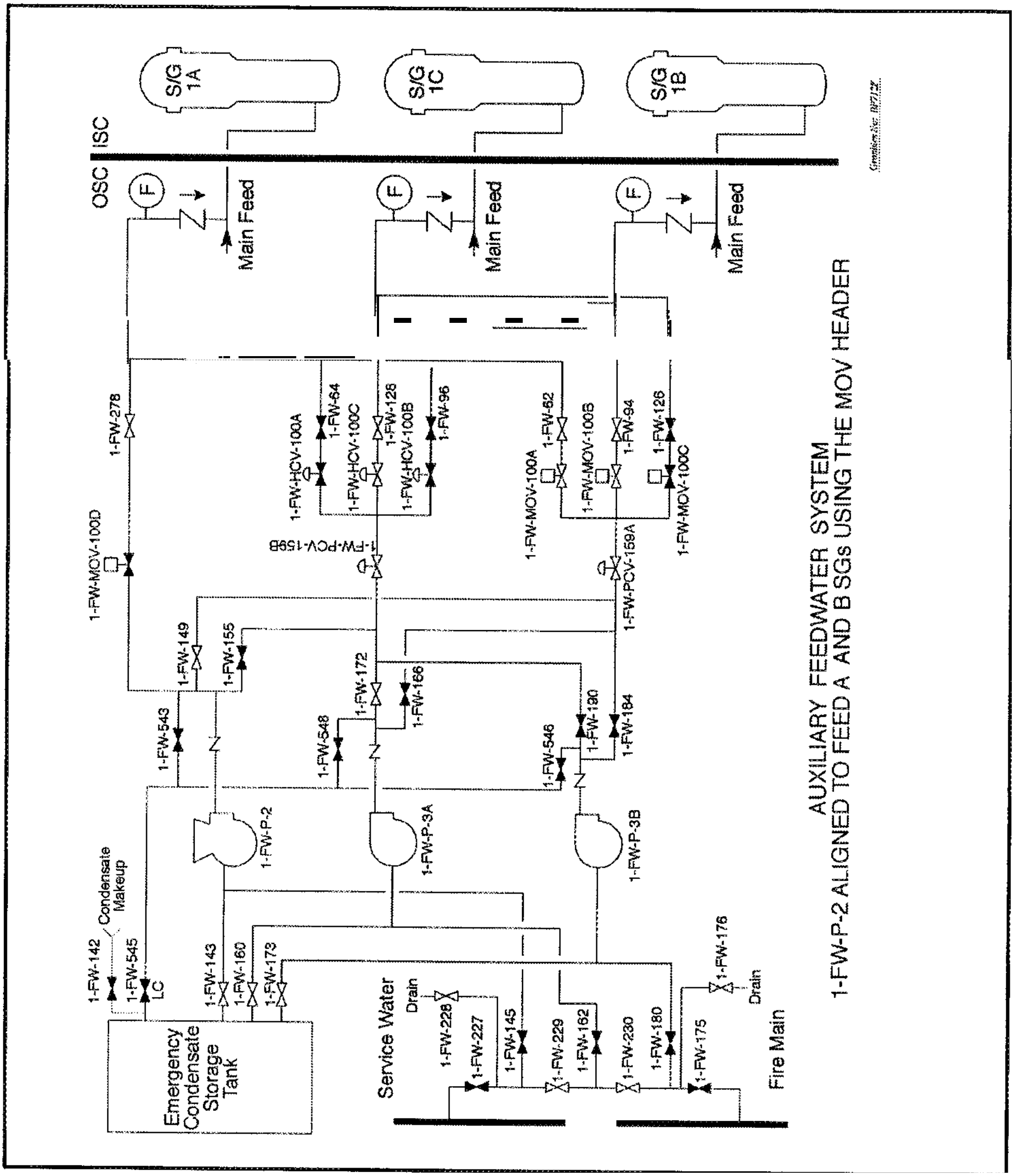
3. Have the CRO open the following AFW Valves:

   - ___ 1-FW-MOV-100A, AFW MOV HEADER TO A SG
   - ___ 1-FW-MOV-100B, AFW MOV HEADER TO B SG

4. Have the CRO close the following AFW Valves:

   - ___ 1-FW-MOV 100D, TURBINE DRIVEN AFW PUMP TO A SG
   - ___ 1-FW-HCV-100B, AFW HCV HEADER TO B SG

5. ___ Notify the Control Room that Attachment 4 is complete and to return to 1-AP-22.3, step in effect.



S/G 1A
S/G 1C
S/G 1B
ISC
OSC
Main Feed
Main Feed
Main Feed
1-FW-278
1-FW-MOV-100D
1-FW-64
1-FW-HCV-100A
1-FW-128
1-FW-HCV-100C
1-FW-96
1-FW-HCV-100B
1-FW-PCV-159B
1-FW-62
1-FW-MOV-100A
1-FW-94
1-FW-MOV-100B
1-FW-126
1-FW-MOV-100C
1-FW-PCV-159A
1-FW-149
1-FW-155
1-FW-172
1-FW-166
1-FW-543
1-FW-548
1-FW-130
1-FW-184
1-FW-546
1-FW-P-2
1-FW-P-3A
1-FW-P-3B
Condensate Makeup
1-FW-142
1-FW-545
LC
1-FW-143
1-FW-160
1-FW-173
Emergency Condensate Storage Tank
Service Water
Drain
1-FW-228
1-FW-227
1-FW-145
1-FW-229
1-FW-162
1-FW-230
1-FW-180
1-FW-175
1-FW-176
Drain
Fire Main
AUXILIARY FEEDWATER SYSTEM
1-FW-P-2 ALIGNED TO FEED A AND B SGs USING THE MOV HEADER

VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-22.2	LOSS OF 1-FW-P-3A MOTOR-DRIVEN AFW PUMP (WITH FOUR ATTACHMENTS)	12
		PAGE 1 of 7

PURPOSE

To provide instructians for placing an AFW source in service when the A Motor Driven AFW pump is inoperable.

ENTRY CONDITIONS

This procedure is entered when AFW Plow to the C SG is required and any of the following conditions exists:

- 1 FW P-3A is inoperable. or
- The HCV header is inoperable. or
- Annunciator Panel "F" C-5. AUX FD PP 3A 3B AUTO TRIP. is LIT.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL:	DATE	
APPROVAL - ON FILE		

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- When ECST level decreases to 40%. then 1-AP-22.5, LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1. should be initiated to provide an alternate water source to the AFW Pumps.
- To prevent heating of the ECST above 120°F, the AFW Pumps should not be run on recirc for extended periods of time.
- To prevent possible degradation to the AFW Pump. the amount of time spent on minimum recirc flow should be minimized.
- To prevent lifting relief valve 1-FW-RV-100 when reducing feed flow. any NOV or HCV supplied by the Turbine-Driven APW Pump should be slowly throttled.
- To prevent lifting relief valve 1-FW-RV-100. a discharge flowpath must be available to feed an SG from the Turbine-Driven AFW Pump.

* *

NOTE:
- Normal PRZR spray should be isolated from any RCP that is stopped.
- The C RCP provides the best PRZR spray capability. The A RCP also provides PRZR spray capability.
- If AFW Pumps are lost due to less of control from the Control Room. then evaluate using 1-AP-20. OPERATION FROM THE AUXILIARY SHUTDOWN PANEL, to start the affected AFW Pumps from the Auxiliary Shutdown Panel.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	CHECK MAIN FEEDWATER - IN SERVICE	Do the following: a) Stop all but one RCP b) Initiate attempts to restore Main Feedwater.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.__	CHECK TURBINEDRIVEN AFW PUMP (1-FW-P-2) - RUNNING	Manually start pump: a) Open 1-FW-MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG. b) Place Control switches for both Turbine-Driven AFW Pump Steam supply valves to OPEN: • 1-MS-TV-111A • 1-MS-TV-111B
3.__	CHECK 1-FW-P-3B. MOTOR-DRIVEN AFW PUMP - RUNNING	Manually start 1-FW P 3B. IF I-FW-P-3B cannot be started. THEN GO TO 1-AP-22.4, LOSS OF BOTH MOTOR-DRIVEN APW PUMPS.
4.__	CHECK ECST LEVEL GREATER THAN *40%*	Initiate 1-AP 22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK I-CN-TK-1.

NOTE: The AFW lineup drawings of Attachments 3 and *4* should be retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.—	PERFORM ATTACHMENT *4* TO ALIGN THE HCV HEADER FOR FEEDING THE A AND C SGs	Perform Attachment 3 to align MOV Header for feeding all SGs. GO TO Step **8**.
• 6.__	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING: • 1-FW-HCV-100A for A SG • 1-FW-MOV-100B for B SG • 1-FW-HCV 100C for C SG	
7.__	GO TO STEP *9*	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8.___	CONTROL AFW FLOW TO MAINTAIN SG NARROW RANGE LEVELS BETWEEN 23% AND 50% USING: • 1-FW-MOV-100A for A SG • 1-FW-MOV-100B for B SG • 1-FW-MOV-100C for C SG	
9.___	DETERMINE IF AFW PUMPS CAN BE RETURNED TO NORMAL DISCHARGE ALIGNMENT	Continue with other procedures and steps in effect. <u>WHEN</u> normal lineup can be established. <u>THEN</u> GO TO Step 10.
10.___	RAISE SG NARROW RANGE LEVELS TO 45% TO 50%	
11.___	STOP AFW PUMPS: a) Reset SI b) Reset AMSAC c) Place Motor-Driven AFW Pumps in PTL: • 1-FW-P 3A • 1-FW P 3B d) Place Steam Supply Valves for Turbine-Driven AFW Pump (Terry Turbine) to CLOSE: • 1-MS-TV-111A • 1-MS-TV 111B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12.__	MANUALLY CLOSE HCV AND MOV HEADER VALVES: • 1-FW-HCV-100A • 1-FW-HCV-100B • 1-FW-HCV-100C • 1-FW-MOV-100A • 1-FW-MOV-100B • 1-FW-MOV-100C • 1-FW-MOV-100D	

NOTE: The AFW lineup drawing of Attachment 2 should be retained in the Control Room to provide Control Room personnel with a graphical representation of the AFW lineup.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. —	PERFORM ATTACHMENT 2 TO RETURN THE AFW PUMPS TO A NORMAL DISCHARGE ALIGNMENT	
14.__	CHECK ECST LEVEL - GREATER THAN 40%	Initiate 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK I-CN-TK-1.
15.__	MANUALLY OPEN DISCHARGE VALVES: • 1-FW-MOV-100D • 1-FW-MOV-100B • 1-FW-HCV-100C	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.__	VERIFY MAIN FEEDWATER IN SERVICE	Do the following: a) Start all AFW Pumps: • 1-FW-P-3A • 1-FW-P-3B • 1-FW-P-2 b) Maintain SG Narrow Range Levels between 23% and 50%. c) GO TU Step 18.
17.__	PLACE AFW PUMPS IN AUTO: a) Place Motor-Driven APW Pump control switches in AUTO: • 1 FW-P-3A • 1-FW-P-3B b) Place Steam Supply Valves for Turbine-Driven AFW Pump (Terry Turbine) In AUTO: • 1-MS-TV-111A • 1-MS-TV 111B	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. —	VERIFY THE FOLLOWING ANNUNCIATORS - NOT LIT • Panel "F" A-8. AFW PUMP DISCH FW-MOV-100B/D NOT FULL OPEN • Panel "F" B-8, AFW PUMP DISCH FW-MOV 100A/C NOT PULL CLOSE • Panel "F" C-5, AUX FD PP 3A-3B AUTO TRIP • Panel "F" C-6. AFW PUMP DESCH FW-HCV-100C NOT FULL OPEN • Panel "F" C-7, AFW PUMP DISCH FW-HCV-100A/B NOT FULL CLOSE • Panel "F" C 8. AFW PUMP DISCH FW-PCV-159A/B NOT OPEN • Panel "F" D-5. AUX FD PP LOCAL CONTROL • Panel "F" D-6. TURBINE DRIVEN AFW PUMP TRAIN A NON-AUTO CONT • Panel "F" D-7. TURBINE DRIVEN AFW PUMP TRAIN B NON-AUTO CONT • Panel "F" D-8. TURBINE DRIVEN AFW PUMP TROUBLE OR LUBE OIL TRBL	Refer to the applicable annunciator response procedure.
19. —	PERFORM APPLICABLE PORTIONS OF 1-OP-31.2A, VALVE CHECK OFF AUXILIARY FEEDWATER	
20. __	RETURN TO PROCEDURE IN EFFECT	

- END -

- UFSAR
- 11715-FM-74A
- Safety System Functional Inspection on Auxiliary Feedwater System at North Anna Power Station. May 20. 1987
- EWR 89-380A, Replacement of 01 PW-126
- DCP 89-18 1. Full Flow Recirc

  1-AP 20, OPERATION PROM THE AUXILIARY SHUTDOWN PANEL
- 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1 CN-TK-1
- 1 OP 31.2A, VALVE CHECK OFF AUXILIARY FEEDWATER
- DCP 92 003-1, Annunciator Window Engraving and Relocation. Unit 1
- Response to IEB 88-04. Potential Safety Related Pump Loss (1 C 3)
- The following EOPs reference this procedure:

  1 ECA-0.1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED
  1-ECA 0.2. LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED
  1-FR H.1, RESPONSE TO A LOSS OF SECONDARY KEAT SINK

NOTE: Some of the AFW System valves have admin locks.

1. Locally perform the following valve lineup (located in the Motor Driven AFW Pumphouse):

   a) Close and lock the follcwing valves

   ___ • 1-FW-149, **Turb** Drvn APW Pump to S/G MOV Hdr Disch Isol Valve

   ___ • 1-FW-62. APW A MOV 1-FW-MOV-100A Outlet Isolation Valve

   ___ • 1-FW-155. Turb Drvn AFW Pump to S/G HCV Hdr Outlet Isol Valve

   ___ • 1-FW 64, AFW A HCV 1-FW-HCV-100A Outlet Isolaticn Valve

   ___ • 1-FW-126. Aux Feedwater C MOV Outlet Isolation Valve

   a) Open and lock the following valves:

   ___ • 1-FW 172, Discharge Valve for 1-FW-P-3A to KCV Header.

   ___ • 1-FW-128, Aux Feedwater C HCV Outlet Isolation Valve

2. ___ Notify the Control Room that Attachment 2 is complete and to return to 1-AP-22.2. step in effect.

NOTE: Some of the AFW System valves have admin locks.

1. Have the CRO close the following AFW Valves:

   ___ • 1-FW-MOV-100A, AFW MOV HEADER TO A SG

   ___ • 1-FW-MOV-100C, AFW MOV HEADER TO C SG

2. Locally unlock and open the following valves (located in the Motor-Driven AFW Pumphouse):

   ___ • 1-FW-149. Turb Drvn APW Pump to S/G Hov Hdr Disch Isol Valve

   ___ • 1-FW-62. AFW A MOV 1-FW-MOV-100A Outlet Isolation Valve

   ___ • 1-FW-126, Aux Feedwater C MOV Outlet Isolation Valve

3. Have the CRO **open** the following AFW Valves:

   ___ • 1-FW-MOV-100A, AFW MOV HEADER TO A SG

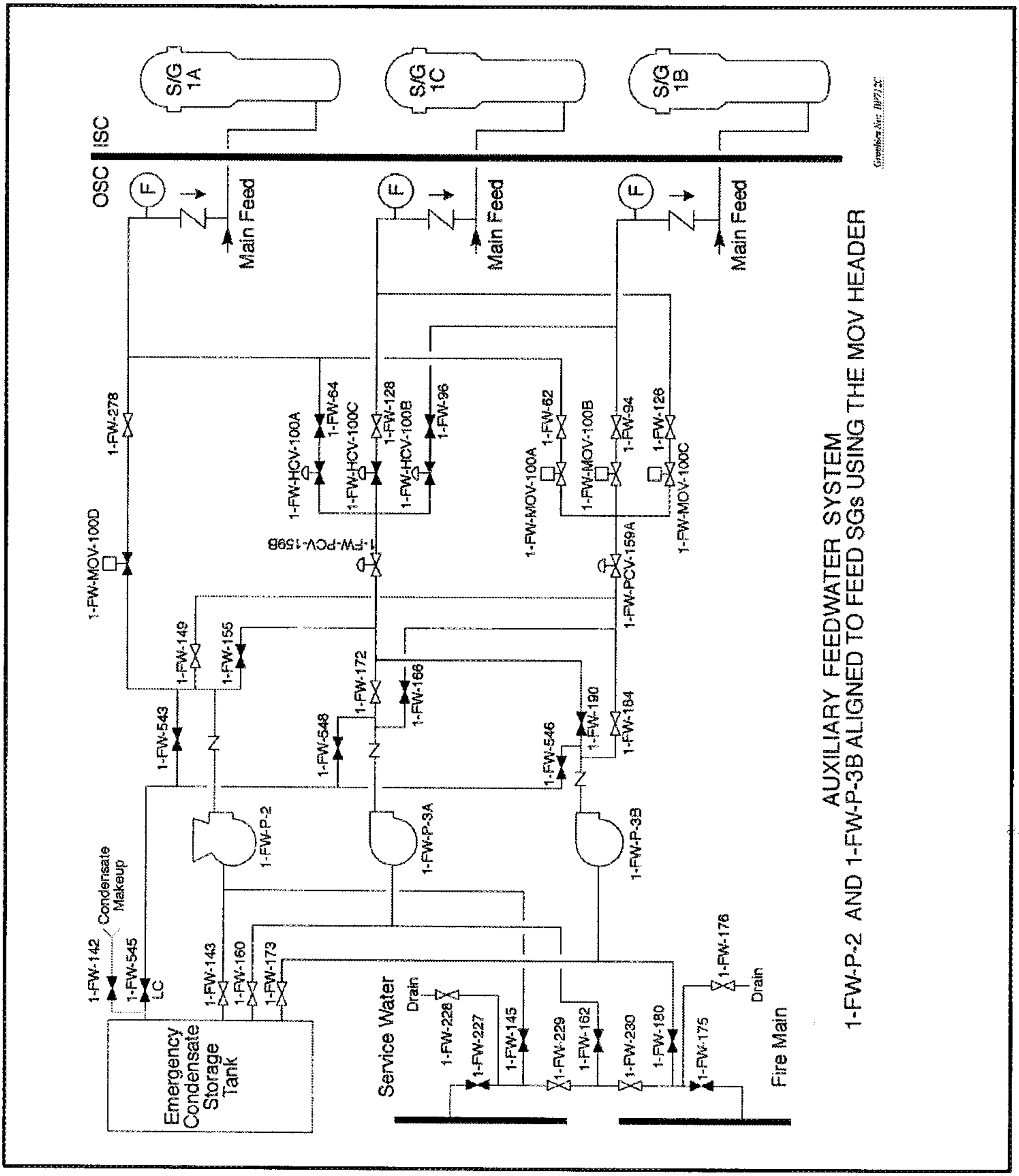
   ___ • 1-FW-MOV-IOOC. AFW MOV HEADER TO C SG

4. Have the CRO close the following AFW Valves:

   ___ • 1-FW-HCV-100C, AFW HCV HEADER TO C SG

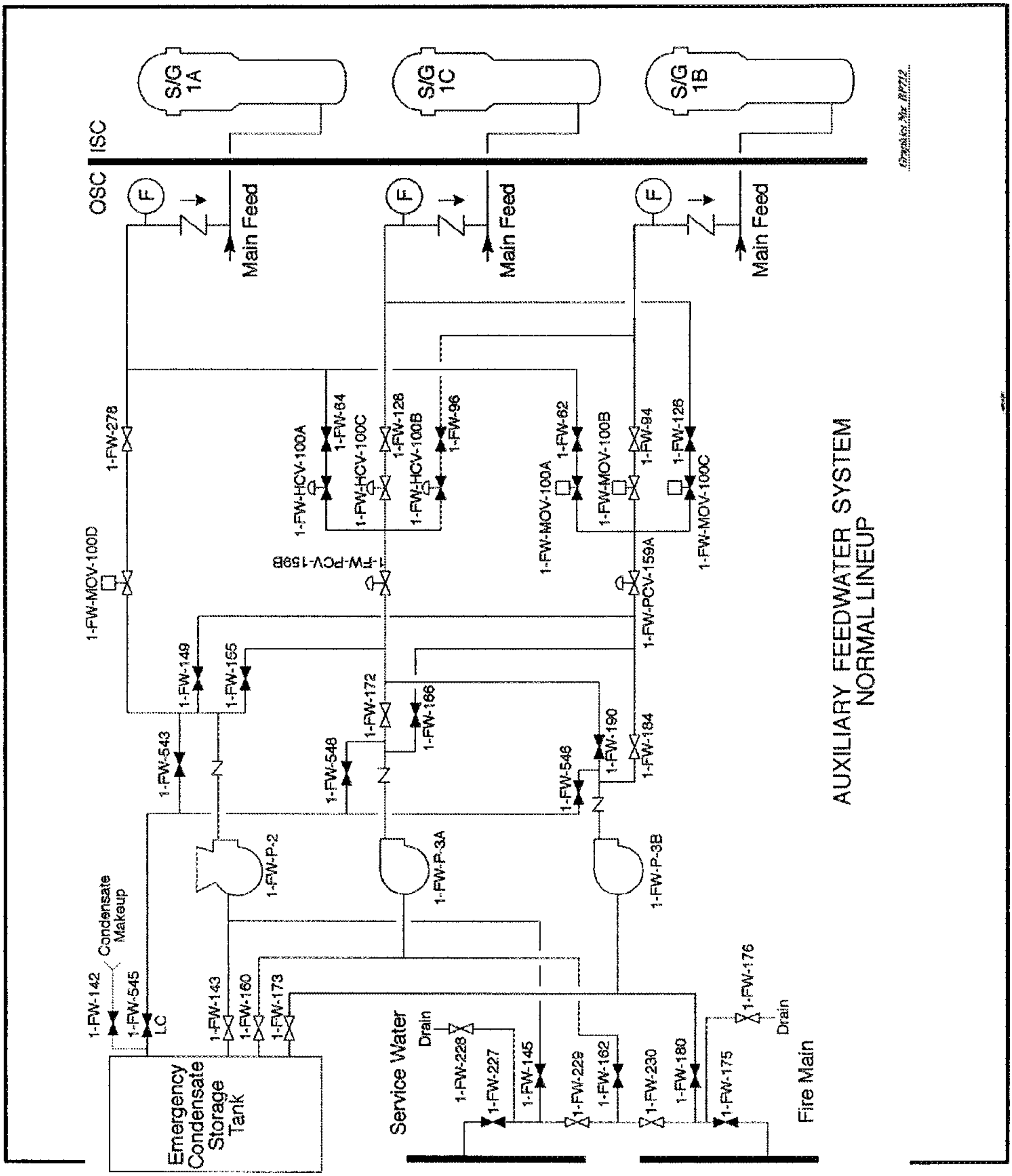
   ___ • I FW MOV-100D, TURBINE DRIVEN AFW PUMP TO A SG

5. ___ Notify the Control Room that Attachment 3 is complete and to return to 1-AP-22.2. **step** in effect.



S/G 1A
S/G 1C
S/G 1B
ISC
OSC
F
Main Feed
1-FW-278
1-FW-MOV-100D
1-FW-64
1-FW-128
1-FW-96
1-FW-HCV-100A
1-FW-HCV-100C
1-FW-HCV-100B
1-FW-62
1-FW-94
1-FW-126
1-FW-MOV-100A
1-FW-MOV-100B
1-FW-MOV-100C
1-FW-PCV-159B
1-FW-PCV-159A
1-FW-149
1-FW-155
1-FW-172
1-FW-166
1-FW-190
1-FW-184
1-FW-543
1-FW-548
1-FW-546
1-FW-P-2
1-FW-P-3A
1-FW-P-3B
Condensate Makeup
1-FW-142
1-FW-545
LC
1-FW-143
1-FW-160
1-FW-173
Emergency Condensate Storage Tank
Service Water
Drain
1-FW-228
1-FW-227
1-FW-145
1-FW-229
1-FW-162
1-FW-230
1-FW-180
1-FW-175
1-FW-176
Fire Main
AUXILIARY FEEDWATER SYSTEM
1-FW-P-2 AND 1-FW-P-3B ALIGNED TO FEED SGs USING THE MOV HEADER

NUMBER 1-AP-22.2	ATTACHMENT TITLE **RETURNING AFW PUMPS TO NORMAL DISCHARGE ALIGNMENT**	REVISION 12
ATTACHMENT 2		PAGE 2 of 2



<u>NOTE</u>: Some of the AFW System valves have sdmin locks.

1. Have the CRO close the following AFW Valves:

___ • 1-FW-HCV-100A, APW HCV HEADER TO A SG

___ • 1-FW-HCV-100C, AFW HCV HEADER TO C SG

2. Locally perform the following valve lineup (located in the Motor Driven AFW Pumphouse):

___ a) Unlock and close 1-FW-172, Discharge Valve for 1-FW-P-3A to HCV Header

___ b) Unlock and open 1 FW-155, Turb Drvn APW Pump to S/G HCV Hdr Outlet Isol Valve.

___ c) Unlock and open 1-FW-64. AFW A EICV 1 FW-HCV-100A Outlet Isolation Valve.

3. Have the CRO open the following AFW Valves:

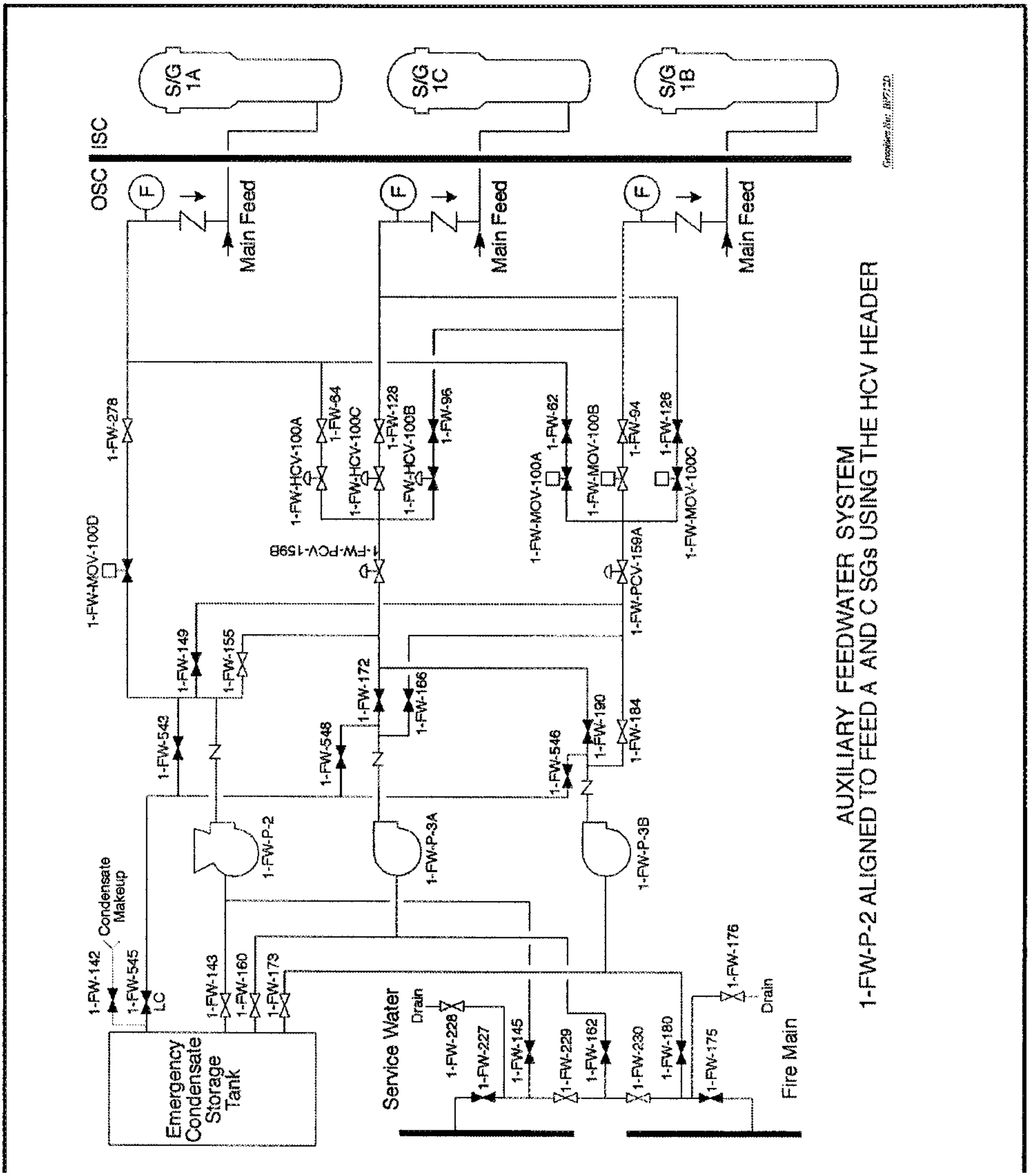
___ • 1-FW-HCV-100A, AFW HCV HEADER TO A SG

___ • 1 FW-HCV-100C, AFW HCV HEADER TO C SG

4. ___ Have the CRO close 1-FW-MOV-100D, TURBINE DRIVEN APW PUMP TO A SG.

5. ___ Notify the Control Room that Attachment 4 is complete and to return to 1-AP 22.2. step in effect.

NUMBER 1-AP-22.2	ATTACHMENT TITLE **ALIGNING THE TURBINE-DRIVEN AFW PUMP TO FEED A AND C SGs USING THE HCV HEADER**	REVISION 12
ATTACHMENT 4		PAGE 2 of 2



062AG2.1.14 001

Unit 1 is operating at 100% power when it sustains a loss of all Service water. The reactor is tripped and the crew has transitioned to 1-ES-0.1, " Reactor Trip Response." 0-AP-12, " Loss of Service Water" is being performed in conjunction with the EOP's. The Shift Manager must ________

A?' make a four hour notification to the NRC, and is required to notify the Manager of Nuclear Operations or a Director immediately
B. declare a NOUE, and is required to notify the Manages of Nuclear Operations or a Director after notifying the State and NRC.
C. make a eight hour notification to the NRC, and is required to notify the Manager of Nuclear Operations or a Director immediately
D. make a one hour notification to the NRC and is also required to notify the Manager of Nuclear Operations or a Director within one hour

A. This is the correct answer. The examinee will have to classify the event and then be able to implement VPAP-2802 to make the correct notifications. Reactor trip is a four hour reportable under VPAP-2802.

B. The examinee could choose this answer. A Boss of service water in Mode 5 is classifiable under €PIP-1.01, Tab A-5, however this has to be combined with a loss of secondary feed to be correct.

C. The examinee could choose this answer, An actuation of the RPS system is mentioned under the eight hour reportable but in the body refers you back to the four hour notification if the reactor is critical.

D. The examinee could choose this answer based on the one hour time frame. They have one hour to notify the Manager of Nuclear Operations but this does not make it a one hour notification.

Loss of Nuclear Svc Water
Knowledge of system status criteria which require the notification of plant personnel.

References: **VPAP 2802**
EPIP 1.01

This is a new question.

Level(RO/SRO):	SRO	Tier:	1
Group:	1	Importance Rating:	2.5/3.3
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

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 WAP-1501, Deviations, establishes responsibilities and processing requirements for initiating and obtaining determinations of reportability for most non-periodic events. **[Commitment 3.2.3]**

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—(707) 452-3769
- 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 (do)468-8892 (Surry)
- State Corporation Commission—(804) 371-961 1
- Area Director of Occupational Safety and Health Administration (OSHA)—
  (804) 371-2327
- Nuclear Mutual Limited and Nuclear Electric Insurance Limited—
  (302) 888-3000; after hours or no answer (302) 479-5222
- American Nuclear Insurers—(860) 561-3433
- Local Emergency Planning Coordinator (LEPC)—
  - Louisa County—(540) 967-0401
  - Surry County—(757) 294-5271

Fax numbers that may not be included in the above listed documents are:

- NRC Operations Center—(301) 816-5151
- NRC Regional Office—(404) 562-4900
- State Department of Environmental Quality —
  - Air/Water/Waste/Pollution Response Regional Office-
    (804) 527-5106 (Surry)
  - Air/Water/Waste/Pollution Response Regional Office-
    (703) 583-3831 (North Anna)
- Nuclear Mutual Limited and Nuclear Electric Insurance Limited
  (302) 888-3008
- American Nuclear Insurers—(860) 561-4655

b. Notifications to other departments inside Dominion for consideration of additional action(s) to be taken include Environmental Policy & Compliance.

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
Sam Nunn Atlanta Federal Center
61 Forsyth St., S.W., Suite 23T85
Atlanta, GA 30303-8931

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-Invel 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
Parkridge 85 North Building
3125 Presidential Parkway, Suite 300
Atlanta, GA 30340

i. Documents to be submitted to the Virginia Department of Emergency Management shall be sent to:

Virginia Department of Emergency Management
10501 Trade Count
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
Town Center, Suite 300S
29 South Main Street
West Hartford, CT 06017-2445

m. Documents to be submitted to Nuclear Mutual Limited shall be sent to:

David Scott or Greg Wilks
Nuclear Mutual Limited/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 20209

6.2 Non-Scheduled Notifications and Reports

NOTE: Reportability determinations for items included in 6.2.1 are initiated and processed in accordance with WAF-1501, Deviations.

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 6.3.4.a.4.

a. **Emergency Plan Activation**—See **6.3.2**, **6.3.5** and **6.3.7**.

NOTE: Operability (availability) is established by the controlling procedure (e.g., Technical Specifications, Station Administrative Procedure). Requirements in this procedure to report inoperable systems or equipment generally rely on other procedures to establish the basis for determining operability.

b. Systems **and Components**

- Reactor trip—See **6.3.3**, **6.3.4.a.**, **6.10.11**, **6.27.1.a.** and **6.27.2**
- Inoperable (including unavailable or out of service) systems or components—See **6.3.3**, **6.3.3.e.**, 6.7.2, 6.10.2, 6.10.11, 6.24.14.b., 6.24.14.c., 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—See **6.3.5.d.**, **6.3.6.a.2.**, 6.25.1 and 6.28.4
- Defective systems or components—See **6.3.4.1.**, **6.3.3.e.**, 6.7.2, 6.10.2, 6.10.11 and h.27.3.e.
- Unacceptable containment leak rate test results—See 6.10.17
- Significant changes in projected values of RT_{PTS}—See 6.10.7
- Reactor Vessel Overpressure Mitigating System is used to mitigate an RCS transient—See 6.24.14.a. **(Surry)**
- Unscheduled outages—See 6.23.1.a. and 6.27.2
- Dissolved gases in transformers exceed limits—See 6.28.2
- Conditions affecting the safety of Lake Anna Dam or its works—See **6.3.2.i.** and 6.18.1
- Planned removal from service, and restoration to service, of Lake Anna Dam safety devices—See 6.18.8

c. **Operating** Limitations

- Technical Specification safety limit exceeded—See **6.3.2.a.5.**, **6.3.6.g.**, 6.7.2, 6.10.2, and6.23.3.
- Limiting Condition for Operation not met—See **6.3.4.1.**, 6.10.2, and 6.10.11.
- Departure **from** license conditions **or** Technical Specificationspermitted by 10CFR 50.54(x)—See **6.3.3.a.** and 6.10.11.
- Excess oxygen in waste gas holdup system—See 6.24.14.d. **(Surry)**
- Excessive quadrant to average power tilt—See 6.24.14.e. **(Surry)**

d. **Radiation or Exposure Events**

- Accidental criticality—See **6.3.3.c.**, 6.7.2, 6.17.1 **and** 6.27.2
- Personnel **contamination**—See **6.3.2.a.4.**, 6.6.4, 6.17.1 **and** 6.27.2
- **Radiation overexposures—See 6.3.2.a.4., 6.6.4, 6.7.2, 6.17.1** and 6.27.2
- Planned special exposures—See 6.6.5
- At receipt, contaminated or excessively radioactive packages—See **6.3.2.b.** and **6.7.2**
- **Radioactive effluent releases—See 6.3.2.a.4., 6.3.6.c.**, 6.6.4, 6.7.2, 6.10.11, 6.10.16, 6.17.1, 6.26.2, 6.27.2, and 6.28.3
- Radioactive **materials transport** incident—See **6.3.2.g.** and 6.28.3
- Twenty **Four** Hour Notification—See 6.3.6.a.1.

e. **Security or Safeguards Events**

- Attempted or actual **unauthorized** entry—See **6.3.3.e.**, 6.15.3 and 6.27.2
- Acts, attempts, or threats to interrupt normal operation—See **6.3.3.e.**, 6.15.3 and 6.27.2
- Loss, theft, or attempted theft of special nuclear material- See **6.3.2.a.3.**, **6.3.3.c., 6.3.3.e.**, 6.6.2.b., 6.7.2, 6.15.3, 6.16.1 **and6.27.2**
- Involving byproduct, source, or special nuclear material—See **6.3.3.e.**, 6.6.3 and 6.15.3
- Attempted or actual introduction of contraband- See **6.3.3.e.**, **6.3.6.b.**, 6.8.1 and 6.15.3
- Lass of shipment of special nuclear material or spent fuel—See **6.3.3.d.**
- Violations **of** requirements of NRC-approved physical security, guard training and qualification, and safeguard contingency plans—See **6.3.6.d.** **(North** Anna)

f. Fitness **for Duty Events**

- Significant Fitness for Duty events—See **6.3.6.b.**, **6.8.1** and 6.27.2
- NRC employee suspected to be unfit for duty—See **6.3.2.c.**
- Fitness for Duty Program false positive test results or unsatisfactory laboratory performance—See 6.8.3 and 6.8.4

g. Environmental Events

- Toxic gas releases—See **6.3.6.c.**, 6.26.2.b. **(North Anna)** and 6.27.2.a.
- Oil or hazardous **material** spills or releases—See **6.3.2.d.**, **6.3.2.e.**, **6.3.6.c.**, 6.20.3, 6.26.2.b., 6.27.2.a., 6.27.3.1. and 6.27.3.n. **(North** Anna)
- Smoke releases from **Station—See 6.3.4.b.**
- Significant increase in nuisance organisms or conditions **(North Anna)**—See **6.3.6.c.** and 6.26.2
- Failure to comply with **VPDES** permit requirements—See **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 6.3.6.f.** and 6.27.3.n.
- **Unpermitted**, unusual, or extraordinary discharge—See **6.3.6.e.** and 6.27.3.n.
- Unanticipated or emergency discharge of waste water or chemical substances See **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** 6.22.4
- Disturbance of **an** osprey nest—See 6.22.4
- Excessive bird impactions **(North Anna)—See** 6.26.2
- Fish **kills—See 6.3.6.c.**, 6.26.2.b. **and** 6.27.2 (North **Anna**)On-site plant or **animal** disease **outbreaks—See** 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 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 **6.3A.a.5.**
- A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask—See 6.3.4.a.5.

i. **Miscellaneous** Events **or Conditions**

- Special circumstances that may be considered media significant—
  See **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 6.27.2.a.
- **Station** fires—See **6.17.1, 6.27.2.a.,** 6.28.2 and 6.28.3
- Demonstrations, picketing, civil disturbances, **strikes**, work stoppages——
  See **6.3.3.e., 6.3.4.a.4.**, and 6.27.2.a.
- Earthquakes, storms, floods, forest or brush **fires**- –
  See **6.3.2.i., 6.10.11, 6.18.1, 6.27.2.a.** and 6.28.3
- Injuries or accidental **deaths—see 6.3.2.g., 6.3.5.c.**, 6.17.1 and 6.27.2.a.
- Deaths or serious injuries at, or alleged to be related to, Lake *Anna* Dam—
  See **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 **6.3.2.g.**,
  6.17.1, 6.21.2, **and** 6.28.3
- Unanalyzed condition that significantly compromises Station safety—
  See **6.3.5.2.**
- Failure to notify NRC of planned removal or significant changes to equipment
  that controls amount of radioactivity in effluents—See 6.3.6.d.
  (North **Anna, Unit 2)**
- **Ambulance** transport of personnel to **an** off-site medical facility—See 6.27.2
- Mishaps involving low-level waste forms—See 6.29.5
- A failure to comply, potentially associated with a significant safety hazard—
  See 6.7.2
- Nonreceipt of hazardous waste shipment manifest from receiver—See 6.20.6.b.
  and 6.27.3.b.
- Planned or emergency removal of asbestos or asbestos containing material—
  See 6.27.3.b.
- Actual or expected unavailability of licensed waste treatment operator—
  See 6.27.3.m.
- **Pump** and haul of bulk-storage-tank bottom waters-See 6.27.3.p. (Surry)
- Operation of auxiliary boiler—See 6.27.3.f. (Surry)
- Licensed material package effectiveness reduction or with safety-significant
  defectsSee 6.13.3

6.2.2 Special Commitments; Administrative Matters

a. **Outages** and Refueling

- Outages—See 6.27.1.a. and 6.27.1.b.
- Refueling—See 6.23.10
- Removal of Reactor Vessel Material Surveillance Program coupons—See 6.10.15
- Restart after refueling, fuel movement, license modification authorizing a power level increase, or Station modifications—See 6.24.4 **(Surry)**
- Inservice inspections—See 6.23.4, 6.23.8.b., **6.24.5 and 6.24.15.a.**
- Steam generator tube inspection sample classified C-3—See **6.3.5.a.1.**

b. Legal & Commercial

1. Program & Procedure Changes
   - Changes to the security plans without prior NRC approval—See 6.10.5.b.
   - Revisions to the Emergency Pian or implementing procedures without prior NRC approval—See 6.10.5.c.
   - Changes to Chemical Test Program procedures—See 6.8.5.
   - Significant changes in the operation of equipment that controls the amount of radioactivity in effluents (North Anna, Unit 2)—See 6.23.6
   - Changes in discharge or management of pollutants—See 6.27.3.j.
   - Significant changes from upstream or downstream conditions addressed in the North Anna Hydroelectric Project Emergency Action Plan (North Anna)—See 6.18.4
   - Decreased availability of private personnel or equipment to prevent or mitigate a worst-case oil release—See 6.27.3.1.

2. **Station** Changes
   - Major changes to radioactive liquid, **gaseous,** or solid waste treatment systems—See 6.10.3
   - Introduction of **an** extremely hazardous substance in **an amount** greater than its threshold planning quantity—See 6.20.8
   - 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 **6.27.3.1.**
   - A substantial increase in the maximum oil storage capacity at the Station—See **6.27.3.1.**

3. Movement of Radioactive Materials
   - Shipment or receipt of SNM—See 6.15.1, 6.15.2, 6.15.3, 6.15.4, and **6.16.3**
   - First **use** of radioactive material packaging—See 6.13.2

4. **NRC Licences, Orders, & Inspections**
   - **Change** in operator or senior operator status—*See* 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 6.5.1.e. and 6.5.B.f.
   - Issuance of **an** NRC shutdown order—See 6.28.6
   - issuance of Dominion Annual Report—See 6.10.8 and **6.14.10**
   - Five years **before** expiration of reactor operating license—See 6.10.5.f.
   - Three years before the predicted date that fracture toughness levels will no longer satisfy 10 CFR 50, App. G, Section V.B.—See 6.10.14.
   - Suspension or revocation of an NRC operating license—See 6.28.6
   - A change of licensee for the Station—See 6.27.3.1.

5. **Permits, Orders, & Evaluations**
   - Proposed changes to the VPDES permit—See 6.26.1.b. (North Anna) or 6.27.3.m. (Surry)
   - Changes or additions to the VPDES permit or State certification —See 6.26.1.b.(North Anna) or 6.27.3.o. (Surry)
   - Stay of a VPDES permit or State certification appeal—See 6.26.1.b. (North Anna)
   - Modifications to Lake Anna Dam or its works—See 6.18.3
   - Suspension from INPO—See 6.28.5
   - Classification as INPO Category 5—See 6.28.5

6. **Insurance & Financial**
   - Material change in proof of financial protection or financial information previously filed—See 6.17.2
   - Expiration, renewal, or replacement of 10 CFR 140 financial protection — See 6.17.3
   - Filing of Chapter 11 petition by or against any component of Dominion Resources—See 6.10.5.g.

c. **Individual Requests or Directives**
   - Worker and former worker radiation exposure data—See 6.5.2 and 6.5.3.
   - Radiation overexposures—See 6.5.4.
   - Terminating employees & workers—See 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 horns.

6.3.1 General Requirements

a. When this subsection (6.3) designates someone other than the Shift Supervisor or a member of Station management to notify a government agency, that person shall ensure the Shift Supervisor or a member of Station management is advised before making the notification. See also 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 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 6.3.3.i. for notification requirements.

b. For events reportable to the NRC Operations Center, the Shift Supervisor shall:

1. Complete NRC Form 361, Event Notification Worksheet.
2. Fax the Event Notification Worksheet to the NRC Operations Center. See 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 ofticer.
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 inoperable, 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 6.1.1. [10 CFR 50.72(a)(ii)(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 10CFR 50.72 and 10CFR 72.75:

- Immediately, the Shift Supervisor 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 Cali shall notify the Site Vice President or a Director
- Within one hour, the STA shall notify the Director Nuclear Station Safety and Licensing or, if the Director Nuclear Station Safety and Licensing is absent, the Director Nuclear Station Operations & Maintenance
- Within one hour, the Director Nuclear Station Safety and Licensing (if absent, the Director Nuclear Station Operations & Maintenance) shall notify the Manager Nuclear Oversight of reactor trips; for other events that are reportable in accordance with 10 CFR 50.72 and 10CFR 72.75, this notification shall be made within 24 hours
- Within 24 hours, the Director Nuclear Station Safety and Licensing (if absent, the Director Nuclear Station Operations & Maintenance) shall notify the NRC Resident Inspector.
- Within 24 hours, the Director Nuclear Station Safety and Licensing (if absent, the Director Nuclear Station Operations & Maintenance) shall notify the Director NL&OS
- Within 24 hours, the Site Vice President, a Director, Manager Nuclear Operations, or Shift Supervisor shall notify the Vice President-Nuclear Operations
- When notified, the Director NL&OS shall promptly notify appropriate corporate organizations, including Public Relations, Medical, Risk Services, and Power Supply, as applicable

6.3.2 Immediate Notifications

MOTE 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 Supervisor 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 6.3.2 or 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 6.3.2 or 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 BO CFR 20.1001-20.2401, Appendix C, wider circumstances in which it appears persons in unrestricted areas could be exposed. See also 6.6.2.b. and 6.6.2.c. [10 CFR 20.2201(a)(i)]

NOTE: The requirements of 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 6.10.11.c. **[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 6.27.2. See also 6.6.3.c.

5. A Technical Specifications safety limit violation. See also 6.23.3, 6.24.3, and *6.3.6.g.* **[10 CFR 50.36(c)(1)(i)(A), & SPS T.S. 6.3.A.2]**

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 liiits of 10 CFR 71.47 **[10 CFR 20.1906(d)(2)]**

1. Radiological Protection shall notify Supervisor Licensing (Station) and the Shift Supervisor.

2. Radiological Protection or Supervisor Licensing (Station) shall notify (see 6.3.1.a.) the final delivering carrier and, by telephone and telegram, mailgram, or facsimile, the NRC Operations Center. See 6.1.1.

3. The notifier in 6.3.2.b.2. shall initiate a Plant Issue (Deviation) as specified in VPAP-1501, including documentation of its notifications on the Plant Issue (Deviation).

c. If an NRC employee 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 6.3.1.a.) (during normal business hours) the NRC Regional Administrator. At other times, notify the NRC Operations Center. See 6.1.1. [10 CFR 26.27(d)]

NOTE: Use Table I, Summary of Reporting Requirements for Non-Radiological Releases To the Environment, to supplement 6.3.2.d. for reporting requirements. The Environmental Compliance Coordinator or Environmental Policy & Compliance 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 [Ref. 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] [Ref. 3.1.8]:

or

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. 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.

1. The individual who observes or suspects **such** an event or condition shall notify the **Shift** Supervisor.
2. The **Shift** Supervisor 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 Environmental Policy & Compliance (see 6.3.k.a.) **(North** Anna) the Shift Supervisor(**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 6.3.4.a.4., 6.20.3, **and 6.27.3.n.**[3]

NOTE: The Environmental Compliance Coordinator or Environmental Policy & Compliance should be consulted when assessing hazardous material release reportability.

e. If a regulated, hazardous material release to the environment' 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 Supervisor. 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** nut reportable **as** a release to the environment **unless** a discharge amount or concentration exceeds **the** permit-authorized limit **OK** 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 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 ami associated reporting requirements will **apply**.
2. Table 1 does not mention PCBs because **no** PCBs are ***in*** use at the Station. The Environmental Compliance Coordinator or Environmental **Policy** & Compliance 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)

2. The Shift Supervisor shalt notify the Manager Nuclear Operations, the Environmental Compliance Coordinator, or Environmental Policy & Compliance, as available.

3. The Environmental Compliance Coordinator or Environmental Policy & Compliance (see 6.3.1.a.) (North Anna) Shift Supervisor **(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** Supervisor shall **also** notify the 91I operator, local and **state** police, and **the** National Response Center. Notifications **shall** be documented on Attachment 1, Oil or Hazardous Substance Release Report. See 6.1.1.a. See also 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 30]

4. Notifications shall include (to the extent known) [CERCLA See. 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
   - Roper precautions **to** take as a result of the release, including evacuation
   - The name and telephone number of the Dominion contact

Table 1
Summary of Reporting Requirements for Non-Radiological Releases To the Environment[a]

Material	Released To		Amount[b]	Report To[c]	See
[illegible]	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	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 71 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.

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 6.3.2.d.; if hazardous materials are involved, go to 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 Environmental Policy & Compliance shall notify (see 6.3.1.a.) the State Department of Environmental Quality (Water) by telephone with the following information [VPDES Permit II.F.2]:

- A description and came 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 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 same 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 6.28.3) [49 CFR 171.15(a)(2)]
- Hazardous materials in which any of the following is a direct result of the hazardous materials: [49 CFR 171.15(a)(1)]
  - A person is killed
  - A person requires hospitalization because of injuries
  - Estimated carrier or other property damage exceeds $50,000
  - 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(a)(4)]

Supervisor Licensing (Station) shall notify (see 6.3.1.a.) DOT by telephone, or confirm carrier notification of DOT by telephone. See also 6.3.2.e. and 6.21.2. The notification shall include the [49 CFR 171.15]:

- 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 Supervisor and provide information necessary to prepare Attachment 2, FERC Public Safety Database Report.

2. The Shift Supervisor shall initiate a Plant Issue (Deviation) in accordance with VPAP-1501.

3. The Shift Supervisor should notify the FERC Regional Engineer of the condition by telephone. See 6.1.1.a. See also 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 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 Supervisor and provide relevant supporting information.

2. The Shift Supervisor shall notify, by telephone, the FERC Regional Engineer of the condition and initiate a Plant Issue (Deviation) in accordance with VPAP-1501. See 6.1.1.a. See also 6.3.4.a.4. and 6.18.1.b. (North Anna) [18 CFR 12.10(a)]

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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 Supervisor, Station Emergency Manager, or Site Vice Resident shall notify the NRC Operations Center of:

a. Deviation from Technical Specifications (permitted by 10CFR 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(c)(1)(ii)(A)]

NOTE: Notifications required by Items 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 7052 (a), 10 CPR 72.74(a), 10 CFR 74.11a]

062GG2.4.31 001

Unit 1 is operating at 100% power when Annunciator F-H5, " 480V OR 4KV EMERG BUS VOLTS HI/LO," actuates. The following conditions exist:

- Switchyard voltage is 530 KV
- Unit 1 main generator is at 200 MVAR IN
- Unit 2 main generator is at 200 MVAR IN
- 1H Emergency Bus voltage is 4415 Volts
- 1J Emergency Bus voltage is 4405 Volts

Per the annunciator response the crew needs to __________

A. adjust the "C" RSST load tap changer
5. reduce 500 KV bus voltage
C. adjust the "A" RSST load tap changer
D. increase 500 KV bus voltage

A. This is the correct answer. Based on the given voltage, the problem is with the "C" RSST. This is to be corrected within two hours.

B. This answer is incorrect. Can't adjust 500 KV bus voltage further since both units are at 200 MVARs in.

C. This answer is incorrect. "A" RSST affects 1J bus.

D. This answer is incorrect. Increasing bus voltage will make 1H bus voltage higher.

AC Electrical Distribution
Knowledge of annunciators alarms and indications and use of the response instructions

References: 1-AR-F-H5

This is a new question.

Level(RO/SRO):	SRO	Tier:	2
Group:	1	Importance Rating:	3.3/3.4
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	Last Exam(Y):	N

480V OR 4KV EMERG BUS VOLTS HI/LO

4160 Buses:
HIGH ≥ 4410 volts
LOW ≤ 3746 volts
(15 sec T.D.)
480 Buses:
HIGH ≥ 515 volts

NOTE: A Deviation Report (DR) is required if the 480V Bus voltage is > 515 volts for a 2 hours. (Reference 3.9)

1.0 Probable Cause

1.1 500KV Bus voltage high or low due to System load

1.2 Failure of Reserve Station Service Transformer tap changer

1.3 Load or excitation improper when Bus is supplied by Emergency Diesel Generator or Station Blackout Diesel Generator

2.0 Operator Action

2.1 Determine whish Emergency Bus is causing alarm (1H or 1J).

NOTE: Automatic LTC operation will occur if 4160V Bus voltage is outside the 4220 - 4410V band, this band corresponds to 500 - 520V on the 480V Bus. If 4160V Bus voltage is within this band, THEN Operator Action to correct the high voltage by additional bus loading or manual tap changer adjustment will be required (no malfunction exists). This action must be commenced within 2 hours of receiving the alarm. (Reference 3.9)

2.2 Determine cause of voltage condition as follows:

2.2.1 IF System voltage is high and System load is light, THEN within 2 hours adjust 500KV Bus voltage, in accordance with System Operator direction. (Reference 3.9)

2.2.2 IF Reserve Station Service Transformer is causing the voltage condition, THEN within 2 hours commence actions to reduce voltage by additional loading of bus or manually adjust the applicable RSST load tap changer using 0-GOP-26.1, Operation *of* Reserve Station Service Transformer Tap Changer.

(Reference 3.9 & 3.10)

1H = RSS C tap changer (normal,alignment)

1J = RSS A tap changer (normal alignment)

2.2.3 IF Emergency Diesel Generator is causing the voltage condition, THEN within 2 hours adjust voltage by use of Exciter Voltage Control switch. (Reference 3.9)

2.2.4 IF the Station Blackout Diesel Generator, during a non-SBO event, is causing the voltage condition, THEN within 2 hours adjust voltage by use of Generator Voltage Control. (Reference 3.9)

2.3 IF 4160V Bus voltage is low. THEN verify that after 56 seconds the Emergency Diesel Generator starts on degraded voltage. (The Bus **will** strip and the EDG output breaker will re-energize the Bus.)

3.0 References

3.1 11715-FE-21T
3.2 11715-FE-21U
3.3 11715-ESK-10BAM
3.4 Memo dated 01-05-94, from D. C. Driver Jr. to J. R. Hayes
3.5 OP-438, Response to DR N-95-0024
3.6 OP-546, Response to DR N-95-0856
3.1 DR N95-1254 480V Bus High
3.8 Memo dated 02-20-96, to Page Kemp from HV Le, Unit 1 480V Emergency Bus Overvoltage Alarms
3.9 Memo dated 09-10-96, to NRC from J. P. O'Hanlon to change response time from 15 minutes to 2 hours.
3.10 0-GOP-26.1, Operation of Reserve Station Service Transformer Tap Changer

4.0 Actuations

4.1 4160V Emergency Bus Overvoltage Relays:
4.1.1 59A/B/C-1H1
4.1.2 59A/B/C-1J1
4.2 480V Bus Overvoltage Relays:
4.2.1 59/1H
4.2.2 59/1H1
4.2.3 59/1J
4.2.4 59/1J1
4.3 4160V Emergency Bus Undervoltage Relays:
4.3.1 27A/B/C-1H1
4.3.2 27A/B/C-1H1-A
4.3.3 27A/B/C-1J1
4.3.4 27A/B/C-1J1-A

PANEL 1H - MAIN CONTROL BOARD

WINDOW NO.	*ALARM*
F-5	4KV EMER BUS 1H NORM SUPPLY BREAKERS AUTO TRIP
F-6	4KV EMER BUS 13 NORM SUPPLY BREAKERS AUTO TRIP
maybe F-7	4KV EMER BUS 1H UV
F-8	4KV EMER BUS 1J **W**
G-1	EMER DG #fH TROUBLE
G-2	EMER DG #1J TROUBLE
G-3	LOSS OF RES SS XFMR A-B-C
G-4	ANNUNCIATOR SYSTEM DC GROUND
G-5	4KV **BUS** 16 NOR **SUP** BKR AUTO TRIP
G-6	4KV RSS BUS FDR BKR AUTO TRIP
G-7	4KV **BUSSES** 1G-2G TIE CLOSED
G-8	4KV BUSSES 16-2G **TIE** AUTO TRIP
H-1	EMER DIESEL GEN #lH DIFFERENTL
H-2	EMER DIESEL GEN #1J DIPFERENTL
H-3	FO STOR TK LEV TROUBLE
H-4	4KV %US 1J EMR SUP BKR AUTO TRIP
H-5	4KV XFER BUS 1D **W**
Good choice H-6	GENERATOR ISO-PHASE BUS DUCT GROUND
H-7	4KV XFER BUS 1F UV
H-8	4KV BUS 1G UV

1-AR-H-IN, Non Controlled - For Reference Only
Panel 1H - MAIN CONTROL BOARD .
REV. 0 REV. N/A

WINDOW NO.	ALARM
A-1	VITAL BUS 1-I INVERT TROUBLE
A-2	VITAL BUS 1-11 INVERT TROUBLE
A-3	VITAL BUS 1-III INVERT TROUBLE
A-4	VITAL BUS 1-IV INVERT TROUBLE
A-5	LOSS REG PT/VOLTS/HERTZ RELAY TRBL
A-6	EMER. DG #1H SWITCH NOT IN AUTO REMOTE
A-7	EMER. DG #1J SWITCH NOT IN AUTO REMOTE
A-8	BUS A OVER 3000 AMP
B-1	BATTERY CHGR 1-I TROUBLE
B-2	BATTERY CHGR 1-11 TROUBLE
E-3	BATTERY CHGR 1-III TROUBLE
B-4	BATTERY CHGR 1-IV TROUBLE
B-5	4KV SUP BKR 15A8 OR 15B10 AUTO TRIP
B-6	AUTO STOP OIL RESET ACTUATED
Nayle B-7	BUS B OVER 3000 AMP
B-8	4KV SUB STA NOR SUP BKR AUTO TRIP
C-1	4KV EMER BUS 1H ALT SUPPLY BREAKERS AUTO TRIP
C-2	4KV EMER BUS 13 ALT SUPPLY BREAKERS AUTO TRIP
C-3	BATTERY CHGR 1C-I TROUBLE X
C-4	BATTERY CHGR 1C-II TROUBLE X
C-5	RSS XFMR 1A LO RELAY TRIPPED
C-6	RSS XFMR 1B LO RELAY TRIPPED

PANEL 1H - MAIN CONTROL BOARD .

WINDOW NO.	ALARM
c-7	RSS XPMR 1C LO RELAY TRIPPED
C-8	SS BUSSES NOR SUP BKR AUTO TRIP
D-l	GENERATOR BREAKER TROUBLE
D-2	4KV-480V SUB STA SUPP BKR BUS 1H AUTO TRIP
D-4	4KV-480V SUB STA SUPP BKR BUS 1J AUTO TRIP
B-4	4KV-480V SUB STA SUPP BKR BUS 1G AUTO TRIP
B-5	RSS XFMR 1A TROUBLE
D-6	RSS XFMR 1B TROUBLE
D-7	RSS XFMR 1C TROUBLE
D-8	4KV XFER BUS BKR AUTO TRIP
Maybe E-1 1-AR-36	MAIN XFMR TROUBLE
E-2	SS XPMR 1A TROUBLE
E-3	SS XPMR 1B TROUBLE
E-4	SS XFMR 1C TROUBLE
E-5	RSS XFMR 1A PW LO RELAY TRIPPED
E-6	RSS XFMR 1B PW LO RELAY TRIPPED
E-7	RSS XFMR 1C PW LO RELAY TRIPPED
E-8	RSS XPMR PW TROUBLE
F-l	MAIN XFMR PILOT WIRE TROUBLE
P-2	MAIN XFMER COOLING CIRCUIT TROUBLE
F-3	4KV BUS 1H EMR SUP BKR AUTO TRIP
F-4	LOSS OF RES STATION POWER

064GG2.4.49 001

Which of the following actions is permissible in accordance with the immediate actions of 1-E-0, " Reactor Trip Response," with no power to either emergency bus?

A✓ Transition to ECA 0.0, "**Loss** Of All AC Power."
B. Emergency start any EDG that did not start.
C. Locally close the EBG output breaker.
D. Reset the overspeed trip lever on the EDG.

A. This is the correct answer. Diagnostics of power problems are handled by ECA 0.0. Step 3 RNO of E-0 sends you there.

5. This answer is incorrect. Examinee may think it **is** permissible to give start signals to equipment that should have auto started. This is done with other components in E-0.

C. This answer is incorrect. Examinee may think closing the EDG output breaker is a simple task that could keep them from transitioning out of E-0.

D. This answer is incorrect. This is a task that is performed in the EOP's without direct procedural guidance on how to do it **but** it is not done until ECA 0.0.

Emergency Diesel Generator
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls

References: 1-ECA-0.0, " **Loss** of All AC Power"
1-EO, " Reactor trip Response "

This is a new question.

Level(RO/SRO):	**SRO**	**Tier:**	*2*
Group:	1	Importance Rating:	4.0/4.0
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	Last Exam(Y):	N

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3]__	VERIFY BOTH AC EMERGENCY BUSSES - ENERGIZED	Do she following: a) IF no AC Emergency Bus is energized. THEN immediately restore power to at least one AC Emergency Bus. IF power cannot be restored. THEN GO TO 1-ECA 0.0, LOSS OF ALL AC POWER, STEP 1. b) Try to restore power to de-energized AC Emergency Bus using 0-AP-10, LOSS OF ELECTRICAL POWER. as time permits. Continue with Step 4.

EMERGENCY DIESEL 1H - LOCAL

LIST OF EFFECTIVE PAGES:

WINDOW NO.	ALARM
A1	STARTING AIR PRESSURE LOW
B1	JACKET WATER OR LUBE OIL TEMPERATURE LOW
C1	LUBE OIL LEVEL LOW
D1	LUBE OIL PRESSURE DIFFERENTIAL AT FILTER
E1	LUBE OIL PRESSURE LOW
A2	LUBE OIL TEMPERATURE HIGH
B2	FUEL OIL LEVEL HIGH
C2	FUEL OIL LEVEL LOW
D2	FUEL OIL PRESSURE LOW
E2	JACKET COOLANT LEVEL LOW
A3	JACKET COOLANT PRESSURE LOW
B3	JACKET COOLANT TEMPERATURE HIGH
c3	START FAILURE
D3	LOSS OF CONTROL POWER
E3	CRANKCASE PRESSURE
A4	ENGINE OVERSPEED
B4	LOSS OF GENERATOR FIELD
C4	GENERATOR OVEREXCITATION
D4	FUEL OIL AUXILIARY PUMP RUNNING
FA	CONTROL ROOM SWITCH IN EMERGENCY POSITION
A5	SHUTDOWN INTERLOCKS NOT RESET
B5	BATTERY CHARGER POWER FAILURE
C5	FUEL OIL AUX PUMP CONT SW IN OFF
D5	STATOR TEMPERATURE HIGH
E5	BLANK

1H-EG-A4

NORTH ANNA POWER STATION
UNIT 1

ENGINE OVERSPEED	> 1050 rpm

NOTE When this and SHUTDOWN INTERLOCKS NOT RESET alarms are present the diesel is to be considered inoperable as per Tech Spec 3.8.1 and 3.8.2 until alarm condition has cleared and the Emergency 1H Diesel Alarm and Shutdown Reset Pushbutton has been reset.

1.0 Probable Cause

1.1 Rapid decrease in generator load.

1.2 Improper governor setting or governor failure.

1.3 Improperly set or faulty overspeed relay.

1.4 Engine firing on iube oil.

2.0 Operator Action

2.1 Verify engine RPM decreasing. IF NOT, THEN depress EDG Emergency Stop RED pushbutton.

2.2 Check governor and overspeed relay setting.

2.3 Submit a Work Request.

2.4 IF engine will NOT stop, THEN inject CO_2 into air intake (engine firing on lube oil).

3.0 References

3.1 11715-LSK-22-12M

3.2 11715-LSK-22-12V

3.3 EM; Tech Manual Colt Industries/Fairbanks-Morse Model 3800TD8-1/8

3.4 NAPS Instrumentation Book (Page -EG-039)

3.5 Tech Spec 3.8.1 and 3.8.2

3.6 11715-BSK-1lC

4.0 Actuation

NOTE: This alarm will give shutdown and lockout under any condition.

4.1 1-EG-SS-602H diesel overspeed relay

ATTACHMENT 2
(Page 3 of 11)
EDG TKOUBLE-SHOOTING

NOTE: WHEN the Annunciator Acknowledge button is pressed, THEN the annunciators listed below wili clear, unless one was the first-out.

______ b. Identify and indicate the status of each annunciator listed below:

ANNUNCIATOR	STATUS (circle one)		
A3, JACKET COOLANT PRESSURE LOW	LIT	FLASHING	NOT LIT
E1, LUBE OIL PRESSURE LOW	LIT	FLASHING	NOT LIT
E3, CRANK CASE PRESSURE	LIT	FLASHING	NOT LIT
C4, GENERATOR OVEREXCITATION	LIT	FLASHING	NOT LIT

______ c. Press Annunciator Acknowledge button.

______ d. Initiate 1-AR-20, Emergency Diesel 1H-Local, for locked-in alarms.

______ 1.4 Locally check EDG for obvious mechanical and electrical abnormalities. Notify the Unit 1 SRO of any abnormalities discovered.

______ 1.5 IF EDG manual trip lever is NOT in the vertical position, THEN reset the lever by placing lever in the vertical position.

NOTE: IF EDG is started from the Control Room, THEN starting will be delayed 2 minutes for prelube.

1.6 Try to start the EDG by any of the foliowing methods:

a. Start EDG from the Control Room as follows:

______ 1. Place Diesel Mode Selector switch in MAN-REMOTE.

QUESTIONS REPORT

for sroquestions

065AG2.4.4 001

Unit 1 is operating at 100% power when the following annunciators come in:

F-F1 STEAM GENERATOR 1A LEVEL ERROR
C-C4 RCP 1A-B-C THERMAL BARRIER CC HI/LO FLOW
C-A1 VCT HI-LO LEVEL L112
C-A4 VCT HI-LO LEVEL L115

The Unit Supervisor surveys the board and notices the following indications:

- All steam generator levels are slowly decreasing
- Feed flow on all steam generators is less than steam flow.
- CC flows to all RCPs is slowly decreasing
- 1-CH-FCV-1122 Charging Flow Control Valve is full open
- 1-CH-LCV-1460A and B Letdown Isolation Valves are closed
- 1-CH-1200A, B, and C Letdown Orifice Isolation Valves are closed.
- VCT level is 13% and decreasing

Based on the above information the crew should enter ________

A✓ 1-AP-28, "Loss of Instrument Air"
B. 1-AP-3, "boss Of Vital Instrumentation"
C. 1-AP- 31, "Loss of Main Feedwater"
D. 1-AP- 33.2, "Loss of RCP Seal Cooling"

A. This is the correct answer. The combination of these indications distinguishes it from any other event.

B. This is incorrect. Examinee may think they have a pressurizer level channel failure because letdown isolated and VCT level is decreasing.

C. This is incorrect. The examinee could choose this answer based on the entry conditions for that AP, however it is only a symptom of a bigger problem. It looks like main feed is going away because all three feed regs are going closed.

D. This is incorrect The examinee may choose this answer based on losing CC to the Thermal Barrier. Seal injection is still present.

Loss of Instrument Air
Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

References: 1-AP 28, " Loss of Instrument Air"
1-AP-3, " Loss of Vital Instrumentation"
1-AP 31, " **boss** of Main Feedwater"
1-WP 33.2, " Loss of RCP Seal Cooling"

This is a new question.

QUESTIONS REPORT
for sroquestions

Level(RO/SRO):	SRO	Tier:	1
Group:	1	Importance Rating:	*414.3*
Type(Bank/Mod/New):	**NEW**	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

<u>NOTE</u>: The following valves in Containment will close on loss of air.

1-CC-TV-106A, B, & C	CC ta RCP A. B. 6 C
1-CC-TV-116A, B. & C	CC from Thermal Barrier A. B, & C
1-NS-LCV-101	CC makeup to NS System
1-CC-TV-102B, D. 6 F	CC from RCP A. B & C
1-CC-TV-101B	CC from Thermal Barriers
1-CC-TV-105A, B, & C	Chilled CC/SW from Air Recirc Coolers A. B, & C
1-SS-TV-108A, B, 6 C	Hot Leg Sample
1-SS-TV-109B 6 C	Cold Leg Sample
1-SS-TV-111A, B, 6 C	S/G Surface Sample
1-DA-TV-100B	Sump Pump Discharge
1-DG-TV-100B	PDTT Pump Discharge
1-VG-TV 100B	PDTT Vent
1-CV-TV-100	Kogger Suction
1-RC-HCV-1556A, B, & C	Loop Fill
1-RC-HCV-1557A, B, & C	Loop Drain
1-RC-HCV-1519B	PG Lo PRT
1-RC-HCV-1550	N_2 to PRT
1-RC-TV-1549	PRT Vent
1-RC-TV-1523	PRT Drain
1-RC-PCV-1455A & B	PRZR Spray
1-RC-PCV-1455C	PORV
1-RC-PCV-1456	PORV
1-RC-TV-1522A, B, & C	PG ta RCP Seal Head Tank A. B, & C
1-RH-FCV-1605	RH Hx Bypass
1-CH-LCV-1460A & B	Letdown Isolation
1-CH-HCV-1311	Auxiliary Spray
1-CH-HCV-1200A, B. & C	Letdown Orifices
1-CH-HCV-1142	RH to Letdown
1-CH-HCV-1201	Excess Letdown Hx Inlet
1-CH-ACV-1137	Excess Letdown Hx Outlet
1-CII-HCV-1307	RCP Seal Bypass
1-SI-PCV-1846	N_2 Regulator
1-SI-HCV-1898	N_2 to PRT
1-SI HCV-18538. B. & C	N_2 to Accumulator A. E. & C
1-SI-HCV-100	N_2 to Accumulator A. B. & C
1-CH-TV-12048	Letdown Isolation
1-RM-TV-100C	Sample to Containment Gas 6 Part Monitor 1-Rt-RMS 159 and 1-RM-RMS-160

NOTE: The following valves in Containment will close on loss of air.

1-SI-HCV-18518. B. & C	Makeup to Accumulator A. B. & C
1-SI-HCV-1850A & B	Accumulator Test A
1-SI-HCV-1850C 6 D	Accumulator Test B
1-SI-HCV 1850E & F	Accumulator Test C
1-SI-HCV-1852A, B. & C	Accumulator Drains A. B, & C
1-SI-73-1842	Accumulator Test Isolations
1-BD-TV-100B & G	Steam Generator "A" Blowdown
1-BD-TV-100D & H	Steam Generator "B" Blowdown
1-BD-TV-100F & J	Steam Generator "C" Blowdown

NOTE: The following valves in Containment will open on loss of air.

1-CC-TV-108A & B	NS TK HX CC Inlet
1-CC-TV-107A & B	NS TK HX CC Outlet
1 RC-HCV-1544	Vessel Flange Leakoff
1-RH HCV-1758	RH Hx Outlet
1 CH-HCV 1310	Charging to Loop "B"
1 CH-HCV 1303A, B. & C	RCP Seal Leakoff A, B. 6 C
1-CN HCV-1389	Excess Letdown 3-Way Valve Fails to VCT (Seal Water Return)

NOTE: The following vi1ves in the AFPH will open on loss of air and depletion of the seismic air flasks.

1-FW-HCV 100A	AFW HCV Header to A SG
1 FW-HCV 1OOB	AFW HCV Header to B SG
1-FW-HCV-100C	AFW HCV Header to C SG
1-FW-PCV 159A	APW Pumps to MOV Kdr Pressure Control Valve
I-FW-PCV-159B	AFW Pumps to HCV Hdr Pressure Control Valve

NOTE: The following valves in the Mechanical Equipment Room will close on loss of air.

1-FW-FCV-1478	A Main Feed Reg Valve
1-FW-FCV-1479	A Main Peed Reg Bypass Valve
1-FW-FCV-1488	B Main Feed Reg Valve
1-FW-FCV-1489	B Main Feed Reg Bypass Valve
1-FW-FCV-1498	C Main Feed Reg Valve
1-FW-FCV-1499	C Main Feed Reg Bypass Valve

NOTE: The following valves in the Auxiliary Building will close on loss of air.

1-CH-FCV-1113B	Boric Acid Blender to VCT Flow Control Valve
1-CH-FCV-1114A	PG to Blender Flow Control Valve
1-CH-FCV-1114B	Boric Acid Blender to VCT Inlet Hdr Flow Gont VV

NOTE: The following valves in the Auxiliary Building will close on loss of air and depletion of the seismic air: flask.

1-RM-TV 100A	Sample to Containment Gas & Part Monitor 1-RM-RMS-159 and 1-RM-RMS-160
1-RM-TV-100B	Sample to Containment Gas & Part.Konitor 1-RM-RMS-159 and 1-RM-RMS-160
1-RM TV-100D	Sample to Containment Gas & Part Monitor 1-RM-RMS-159 and 1-RM-RMS-160

NOTE: The following valves in the Auxiliary Building will open on loss of air.

1-CH-FCV-1122	Charging Flow Control Valve
1-CH-FCV-1113A	BA to Blender Flow Control Valve

NOTE: The following valves in the MSVH will open on loss of air.

1 MS TV-111A	Turbine-Driven AFW Pump Steam Supply Valve
1-MS-TV 111B	Turbine-Driven AFW Pump Steam Supply Valve

MOTE: The following valves in the MSVH will close on loss of air.

1 MS-TV-101A	A Main Steam Trip Valve
1-MS-TV-101B	B Main Steam Trip Valve
1-MS-TV-101C	C Main Steam Trip Valve
1-MS-TV-113A	A MSIV Bypass Valve
1-MS-TV-113B	B MSTV Bypass Valve
1-MS-TV 113C	C MSTV Bypass Valve

NOTE: The following valves in the MSVH will close on loss of air and depletion of the seismic air flasks.

1-MS-PCV-101A	A *SG* Power Operated Relief Valve
1 MS-PCV-101B	B *SG* Power Operated Relief Valve
1-MS PCV 101C	C *SG* Power Operated Relief Valve

NOTE: The Main Control Room Chiller SW outlet header PCVs fail open and the recirc PCVs fail closed on loss of air.
IF Service Water temperature is less than 60°F. THEN Service Water flow from the Main Control Room Chillers should he manually throttled to maintain between 60°F and 85°F Condenser SW Outlet temperature.

CHILLER	THROTTLE VALVE	TEMPERATURE INDICATOR
1-HV-E-4A	1-SW-383	1-HV-TI-1223A
1-HV-E-4B	1-SW-439	1-HV-TI-1223B
1-HV-E 4C (valve used depends on SW header alignment)	1-SW 385 OR OR 405 1-SW	1-HV-TI 1223C

NOTE: The following Auxiliary Building Central Exhaust Dampers will reposition to the filter bypass position on a loss of air and depletion of the Central Exhaust Seismic Air Accumulators.

- 1-HV-AOD 103-1 - Bypass Damper
- 1-HV-AOD-103-2 - Bypass Damper
- 1 HV AOU 103 3 Filter Damper
- 1-HV-AOD-103-4 - Filter Damper

VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-31	LOSS OF MAIN FEEDWATER	2
	(WITH ONE ATTACHMENT)	PAGE 1 of 5

PURPOSE

To provide instructions for recovering from a loss of Feedwater flow in Mode 1 or Mode 2.

ENTRY CONDITIONS

This procedure is entered when any of the following conditions exists:

- Loss of 1 or 2 Main Feed Water Pumps
- Annunciator Panel F-A4, MAIN FD PPS DISCH HDR LO PRESS
- Annunciator Panel F-A5, MAIN PD PP 1A 1B-1C AUTO TRIP
- Annunciator Panel E B5, MAIN FD PPS LO DIFF PRESS
- Inadequate Feed Flow to more than one Steam Generator as indicated by:

  Annunciator Panels F C1/C2/C3, STM GEN 1A/1B/1C LO LEVEL CH I-II

  OR

  Annunciator Panels F D1/D2/D3, STM GEN 1A/1B/1C FW<STM FLOW CH III-IV

  OR

  Annunciator Panels F F1/F2/F3, SG 1A/1B/1C LEVEL ERROR

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL:	DATE	
APPROVAL - ON FILE		

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]__	CHECK **MPW** PUMP STATUS:	
	a) Reactor Power - GREATER THAN 70%	a) <u>IF</u> at least one **MFW** Pump *is* running. **<u>THEN</u>** GO TO Step 2. ~~IF~~ ~~NO~~ **MPW** Pumps are running, THEN trip Reactor and Turbine and GO TO **1-E-0,** REACTOR TRIP **<u>OR</u>** SAFETY INJECTION.
	b) Two **MFW** Pumps - **RUNNING**	b) <u>IF</u> **a** second **MFW** Pump cannot be immediately started. **<u>THEN</u>** trip Reactor and Turbine and GO TO **1-E-0.** REACTOR TRIP OR SAFETY INJECTION.
[2]__	CHECK **MFW SUCTION** PRESSURE AT **LEAST 300 PSIG**	Start **an** additional Condensate Pump.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: • Turbine ramp rates must be limited to 5%/minute or less.

• Control rods should be maintained above insertion limits.

* *

NOTE: Ramp rates close to 5%/minute may cause the Steam Dumps to arm.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.__	EVALUATE REDUCING TURBINE LOAD TO LESS THAN 55% POWER:	
	a) Verify ONLY ONE MFW Pump - RUNNING	a) GO TO Step 4.
	b) Check Reactor Power level - GREATER THAN 55%	b) GO TO Step 4.
	c) Check Turbine load control:	
	1) Verify Turbine valve position - OFF VALVE POSITION LIMITER	1) Take Turbine off Valve Position Limiter.
	2) Verify Turbine Load Control in IMP-IN.	2) Place Turbine Load Control in IMP-IN by depressing the IMP-IN pushbutton.
	d) Reduce Turbine load to 50-55% using OPERATOR AUTO or TURBINE MANUAL	
	e) Insert Control Rods in AUTO or MANUAL as required to maintain Tavy within 5°F of Tref	
	f) Borate as required to maintain final Control Rod position above insertion limits	
	g) Energize additional PRZR Heaters as required to maintain PRZR Pressure above 2205 psig	
	h) Monitor Steam Dumps for praper operation	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 4.__	STABILIZE SG LEVELS:	
	a) Verify steam flow - LESS THAN AVAILABLE FEED FLOW	a) Reduce Turbine load.
	b) Verify SG levels - AT OR TRENDING TO PROGRAM LEVEL	b) Place associated valves in **MANUAL** and control SG levels: • Main Feed Reg Valves • Main Feed Reg Bypass Valves
* 5.__	VERIFY ACCEPTABLE MFW PUMP PERFORUANCE:	
	a) Verify MFW Motor amps - LESS THAN 550 AMPS ON EACH MOTOR	a) Reduce Turbine load.
	b) Verify Annunciator Panel F-B5, MAIN FD PPS LO DIFF PRESS MOT LIT	b) Reduce Turbine load.
6.__	MAINTAIN STABLE PLANT CONDITIONS	
7.__	CHECK IF ISOTOPIC ANALYSIS OF RCS IS REQUIRED:	
	a) Check Reactor Power HAS DECREASED MORE THAN 15% IN ONE HOUR	a) GO TO Step 8.
	b) Have Chemistry perform isotopic analysis of RCS for iodine within 2 to 6 hours	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.__	MAKE NOTIFICATIONS: • STA • Operations Manager On Call • System Operator • Other notifications as required by VPAP-2802. NOTIFICATIONS AND REPORTS	
9.__	INVESTIGATE REASON FOR LOSS OF FEEDWATER:	
	a) Have Operator locally check the following as required: • MFW Pump that auto-started • MFW Pump that tripped • Breakers for tripped MFW Pump • Condensate Pump that was started	
	b) Verify non-isolated Main Feedwater Pump Recirc Valves NOT FAILED • 1-FW-FCV-150A • 1-FW-FCV-150B • 1-FW-FCV-150C	b) Locally isolate failed FCV(s) by closing the following as applicable: • 1-FW-292 (1-FW FCV 150A) • 1 FW-293 (1-FW-FCV-150B) • 1-FW-294 (1-FW FCV 150C)
	c) Check for unexplained increase in FW Heater level	
	d) Walk down Condensate and Feedwater systems as required	
10.__	INITIATE ANY REQUIRED WORK REQUESTS	
11.__	RETURN TO PROCEDURE AND STEP IN EFFECT	

END -

- UPSAR 10.4.3, 15.2.8
- OP 512. Develop new AP €or loss of Main Feedwater
- Unit 1 Tech Spec 3.1.1.1 (ITS 3.1.1)
- Unit 1 Tech Spec 3.1.3.6 (ITS 3.1.6)
- VPAP 2602. Natifications And Reports
- 1-AP 2.1. Turbine Trip Without Reactor Trip Required
- 1 AP 3, Loss of Vital Instrumentation
- 1 E-0. Reactor Trip Or Safety Injection

NUMBER	PROCEDURE TITLE	REVISION
1-AP-33.2	LOSS OF RCP SEAL COOLING (WITH TWO ATTACHMENTS)	8
		PAGE 1 of 5

PURPOSE

To provide instructions for recovering from a loss of RCP seal cooling.

ENTRY CONDITIONS

This procedure is entered by transition from another plant procedure or when a loss of RCP seal cooling occurs to one or more RCPs as indicated by:

1. Loss of seal injection flow as indicated by:
   - Seal injection flow reading zero on 1-CH-FI-1124, 1127. and/or 1130. or
   - Annunciator Panel "C" G-6, RCP 1A-B-C LABYTH SEAL LO FLOW. LIT.

AND

2. **Loss** of Thermal Barrier Component Cooling flow as indicated by:
   - Component Cooling flow reading zero on 1-CC-FI-116A, 116B, and/or 116C, or
   - Annunciator.Panel "C" C-4, RCP 1A-B-C THERM BARR CC HI/LO FLOW. LIT.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL:	DATE	
APPROVAL ON FILE		

103A2.05 001

Unit 1 is at 100% power with a containment entry team preparing to enter containment to isolate a leaking steam generator channel. The channel has already been placed in trip. The following sequence of events takes place:

- 0830 on May 2nd team enters containment through the personnel hatch
- 0847 on May 2nd team isolates steam generator level transmitter
- 0850 on May 2nd team leaves containment via the personnel air lock emergency doors due to a problem with a faulty air lock pushbutton.
- 0800 on May 9th 1-PT-62.4," Personnel Air bock Seal Leakage," was performed. Leakage on the inner door was 2.75 SCFH and 1.85 **SCFH** on the outer door.

At 0900 on May 9th the Unit Supervisor was reviewing all the paperwork associated with the evolution. Based on the above information, the Unit Supervisor would realize 1-PT-62.5," Personnel Air Lock Escape Door Seals Testing," was not performed

____

A✓ within its specified interval and must be performed within 24 hrs unless a risk evaluation is done. **Nu** actions need to be entered at this time

B. but is within its grace period for the surveillance. No actions were entered at this time

C. within its specified interval and must be performed within 24 hrs unless a risk evaluation is done. Declared inner and outer emergency doors inoperable and entered actions for condition C

D. but is within its grace period for the surveillance. Declared inner and outer emergency doors inoperable and entered actions for condition C

A. **This** is the correct answer. Using (1-OP-18.1) Operation of the Personnel Air Lock PT-62.5 needs to be performed within 7 days. TSR 3.0.2 for grace does not apply. TSR 3.0.3 states no actions entered until 24 hrs or specified frequency whichever is greater.

B. This answer is incorrect because the grace period mentioned in 3.0.2 does not apply. Examinee may use several pieces of T.S. 3.6.2 and 1-OP-18.1 correctly but not identify the note that 3.0.2 doesn't apply.

C. This answer is incorrect. Provisions of TSR 3.0.3 clearly states no actions apply until 24 hrs or specified frequency whichever is greater. Examinee could work through the procedure and T.S. 3.6.2 and not correctly apply TSR 3.0.3.

D. This answer is incorrect. Provisions of TSR 3.0.3 clearly states no actions apply until 24 hrs or specified frequency whichever is greater. Examinee could work through the procedure and T.S. 3.6.2 and not correctly apply TSR 3.0.3.

Ability to (a) predict the impacts of the following on the Containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Emergency containment entry

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

References T.S 3.6.2 Containment Leakage Rate
T.S. 3.0. Surveillance Requirement Appicability
T.S 5.5.15 Containment Leakage Testing Program
TRM 3.6.2 Containment Leakage Rate.

This is a new question: References provided.

Level(RO/SRO):	SRO	Tier:	2
Group:	1	Importance Rating:	2.9/3.9
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits. Surveillances may be performed by means of any series of sequential, overlapping, or total steps.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

------------------------------------NOTES------------------------------------

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment." when air lock leakage results in exceeding the overall containment leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	-----------NOTES----------- 1. Required Actions A.1, A.2, and A3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls. ---------------------------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.l	Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND		
	A.2	bock the OPERABLE door closed in the affected air lock.	24 hours
	AND		
	A.3	---------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------------------- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-------------NOTES------------- 1. Required Actions 8.1, 8.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. -------------------------------	
	B.1 Verify an OPERABLE door is closed in the affected air lock.	1 hour
	AND	
	B.2 Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND	
	8.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. .... Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or mare containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	<u>AND</u>		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. --- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

analysis cannot be performed. For the purpose of this program. a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." dated September 1995 as modified by the following exception:

   NEI 94-01-1995, Section 9.2.3: The first Unit 1 Type A test performed after the April 3, 1993 Type A test shall be performed no later than April 2, 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is 45 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Prior to entering a MODE where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   ≤ 0.60 L_a for the Type B and Type C tests on a Maximum Path Basis and ≤ 0.75 L_a for Type A tests.

   During operation where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   5 1.0 L_a for overall containment leakage rate and ≤ 0.60 L_a for the Type B and Type C tests on a Minimum Path Basis.

2. Overall air lock leakage rate testing acceptance criterion is 50.05 L_a when tested at ≥ P_a.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Leakage Rate

TR 3.6.2 Containment leakage rate shall meet the requirements of Technical Specification 5.5.15, "Containment Leakage Rate Testing Program."

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TR not met.	A.1 Enter applicable Condition(s) of Technical Specification (TS) 3.6.1. "Containment," TS 3.6.2, "Containment Air Locks," or TS 3.6.3, "Containment Isolation Valves."	Immediately

TRM SURVEILLANCE REQUIREMENTS

------------------------------ NOTE ------------------------------
TSR 3.0.2 is not applicable.

SURVEILLANCE	FREQUENCY
TSR 3.6.2.1 Verify total Minimum Path Leakage Rate within limit.	Whenever the leakage rate of a penetration or valve is updated or known to change

TRM SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.6.2.2	NOTES 1. Test Frequency is based upon a minimum of two consecutive successful periodic Type A tests where the calculated performance criteria for leakage is less than 1.0 L_a with the elapsed time between the tests being at least 24 months, and with consideration of the acceptable performance factors. 2. For Unit 1, must be performed by **April 2, 2008.** Verify Type A Integrated Containment Leakage Rate within limit.	$\leq$ 10 years
TSR 3.6.2.3	NOTE Test Frequency is based upon the completion of two consecutive periodic as-found Type B tests where results of each test are within the allowable administrative limits and the elapsed time between the first and second tests in a series of consecutive satisfactory tests used to determine performance is 24 months or the nominal test interval for the component. Verify containment penetration leakage rate within limit by performance of Type B tests.	$\geq$ 30 months and $\leq$ 120 months

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.6.2.4	-------------------NOTE------------------- Test Frequency is based upon the completion of two consecutive periodic as-found Type C tests where results of each test are within the allowable administrative limits and the elapsed time between the first and second tests in a series of consecutive satisfactory tests used to determine performance is 24 months or the nominal test interval for the component. --	
	Verify containment isolation valve leakage rate within limit by performance of Type C tests.	≥ 30 months and ≤ 60 months
TSR 3.6.2.5	Verify air lock leakage rate within limit.	30 months
TSR 3.6.2.6	Test air lock door seals.	Prior to entering MODE 4 __AND__ Once within 7 days after each containment entry
TSR 3.6.2.7	-------------------NOTE------------------- For Unit 1, must be performed during one refueling outage between January 1, 2003 and April 2, 2008. --	
	Perform a general visual examination of the accessible interior and exterior surfaces of the containment and components including the liner plate for structural problems which may affect either the containment structure leakage integrity or the performance of the Type A test.	Prior to initiating Type A Test and during two other refueling outages between Type A Tests during 10 year interval

VIRGINIA POWER **NORTH ANNA POWER STATION**	PROCEDURE NO: 1-OP-18.1	
	UNIT NO: 1	REVISION NO: 20
PROCEDURE TYPE: OPERATING	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE:

OPERATION OF THE CONTAINMENT PERSONNEL AIR LOCK

SURV REQ							

REVISION SUMMARY:

NON UPGRADED PROCEDURE

For **Plant** Issue N-2003-0226, added new Reference 2.4.6.

Added P&L 4.12 that When operating Personnel Hatch doors, ensure either both doors are left closed during the entry or both doors are left open (when Containment is atmospheric) to prevent personnel injury.

Added **a** Warning for Step 5.6.16 to Ensure both doors are left open. If the inner door is closed, the next individuals to open the inner door may be unaware that extreme care must be taken when opening the inner door due to possible pressure differences which could exist. Personnel injury or equipnient damage may result

ELECTRONIC DISTRIBUTION — APPROVAL ON FILE

PROBLEMS ENCOUNTERED: ☐ Yes ☐ No **NOTE:** If *yes,* note problems in Remarks.

REMARKS:

(use back for additional space)

SHIFT SUPERVISOR: DATE:

TABLE OF CONTENTS

1.0 PURPOSE

1.1 This procedure provides instructions for:

1.1.1 Entering the Containment through the Personnel Air Lock.

1.1.2 Leaving the Containment through the Personnel Air Lock

1.1.3 Entering Containment by Manual Operation of the Personnel Air Lock.

1.1.4 Leaving the Containment by Manual Operation of the Personnel Air Lock.

1.1.5 Operation of the Personnel Air Lock Emergency Doors.

1.1.6 Opening Both Inner and Outer Personnel Air Lock Doors for Maintenance Access.

1.1.7 Returning the Personnel Air Lock Doors to Normal Status After Being Both Open

2.0 REFERENCES

2.1 Source Documents

2.1.1 UFSAR

2.2 Technical Specifications

2.2.1 Tech Spec 3.6.1

2.2.2 Tech Spec 3.6.2

2.2.3 Tech Spec SR 3.6.2.1

2.2.4 Tech Spec 3.9.4

2.2.5 Tech Spec 3.9.7

2.2.6 Technical Requirements Manual 3.6.2

2.2.7 Technical Requirements Manual SR 3.6.2.6

2.2.8 Technical Requirements Manual 7.2

2.3 Technical References

2.3.1 11715-FB-100A, Personnel Hatch Hydraulic System

2.3.2 Instruction Manual Personnel Air Lock (Chicago Bridge and Iron)

2.4 Commitment Documents

2.4.1 Deviation N-93-0479

2.4.2 Deviation N-95-985. Personnel Hatch Inner Door

2.4.3 Technical Specification amendment 198 (Ut) and 179 (U2), allowing both personnel airlock doors to be open while containment integrity is set, and their bases, defining an operable air lock door

2.4.4 Plant Issue N-2001-0123, Unit 2 Containment Personnel Hatch Inner Door Equalizing Valve Appears to be Approximately 1/8" Open

2.4.5 TRM Revision 52, Addition of Unit 1 and Unit 2 Personnel Hatches to Table 7.2-1

2.4.6 Plant Issue N-2003-0226, Containment Personnel Hatch Door S w i g Open Rapidly

3.0 INITIAL CONDITIONS

WHEN the Unit is in Mode 5 or 6 AND both air lock doors have been opened simultaneously for general Containment access AND it is desired to close both doors for the purpose of establishing Containment Integrity, THEN 1-OP-18.1A, Containment Access System Valve Checkoff, must be performed. (Reference 2.4.1)

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The following Tech Specs apply:

- T.S. 3.6.1, Containment
- T.S. 3.6.2, Containment Air Locks
- T.S. 3.9.4, Containment Penetrations
- T.S. 3.9.7, Refueling Cavity Water Level

4.2 Ensure there are no obstructions in the Air Lock rotating parts.

4.3 If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

4.4 All open and close pushbuttons are momentary contact switches. They must be held (depressed) throughout the opening or closing operation.

4.5 IF the equalization hall valve is jogged open and the full closed limit has NOT cleared, THEN the valve may need to be sufficiently opened to allow clearing the limit, before the valve will operate to the closed position.

4.6 To prevent damage to the Containment Personnel Air Lock door the locking ring must be in the FULL OPEN position before opening or closing the door. (Reference 2.4.2)

4.7 When hydraulic operation of door is complete, the door position light will shift (locked, unlocked) and the sound of the pump will have a step change in pitch due to unloading. (Reference 2.4.4)

4.8 Excessive operation of the hydraulic pump may cause the motor to overheat and thermal out. If the hydraulic pump stops unexpectedly, the Control Room should be notified. There is a thermal overload reset pushbutton located inside the Personnel Hatch in the control cabinet that may be reset with SRO concurrence. (Reference 2.4.4)

4.9 If a malfunction occurs while entering or exiting Containment, the Control Room should be contacted for direction.

4.10 When subatmospheric conditions exist in Containment and multiple bottle entries are required, the Containment Atmosphere must be sampled for O_2.

4.11 The personnel hatches are NOT required to meet Technical Requirements Manual 7.2, Required Action A.2, while the Unit is Defueled OR in Mode 5 OR 6. **(Reference 2.4.5)**

4.12 When operating Personnel Hatch doors, ensure either both doors are left closed during the entry or both doors are left open (when Containment is atmospheric) to prevent personnel injury. **(Reference 2.4.6)**

Init Verif

5.6 INSTRUCTIONS

5.1 Entering The Containment Through The Personnel Air Lock

______ 5.1.1 Review Precautions and Limitations.

______ 5.1.2 IF this entry will be a multiple bottle entry and the Containment is at subatmospheric conditions, THEN the Containment Atmosphere must be sampled for O_2 AND the O_2 must be acceptable prior to swapping breathing air bottles in Containment.

5.1.3 IF the current Containment entry is the first entry within two months, THEN take the following items: **(Reference 2.4.4)**

______ • A hard copy of this procedure

______ • Eight replacement light bulbs for the door position indicators (656AC 155V 6W)

______ • Two replacement light bulbs for the hatch interior (60W long life)

______ • A flat tip screwdriver

______ • A Phillips head screwdriver

______ 5.1.4 Ensure that the Containment lights are turned on.

______ 5.1.5 IF the Personnel Air Lock will be operated during Core Alterations, THEN issue Attachment 4, Personnel Air Lock Operator Duties and Responsibilities Guideline During Core Alterations, to the air lock operator.

* *

WARNING: If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

* *

______ 5.1.6 Verify no one is in the Air Lock

NOTE: Before opening or closing air lock doors, ensure all equipment and personnel are clear of the door swing area.

5.1.7 Verify the "DOOR IN USE", red light is OFF.

NOTE: When hydraulic operation of door is complete, the door position light will shift (locked, unlocked) and the sound of the pump will have a step change in pitch due to unloading.

NOTE. Excessive operation of the hydraulic pump may cause the motor to overheat and thermal out. If the hydraulic pump stops unexpectedly, the Control Room should be notified. There is a thermal overload reset pushbutton located inside the Personnel Hatch in the control cabinet that may be reset with SRO concurrence.

NOTE: If a malfunction occurs while entering Containment, the Control Room should be contacted for direction.

5.1.8 Depress the outside door **OPEN** pushbutton.

5.1.9 Release the OPEN pushbutton when the locking ring is in the unlocked position.

______ 5.1.10 Manually open the outside door and enter the Air Lock.

_______ 5.1.11 Ensure that the Personnel Air Lock hydraulic oil reservoir level is normal.

5.1.12 IF hydraulic oil reservoir level is NOT normal, THEN do one of the following, otherwise mark this Step N/A:

_______ • Add oil

_______ • Abort the entry

_______ 5.1.13 Manually shut the outside door.

_______ 5.1.14 Depress the outside door CLOSE pushbutton. (IF nothing happens, THEN ensure knife edge is against sed and try again.)

_______ 5.1.15 Release the CLOSE pushbutton when the door position light indicates LOCKED AND/OR the pump sound changes pitch.

_______ 5.1.16 Depress the inside door OPEN pushbutton until a comfortable pressure change rate is achieved.

5.1.17 IF depressurization is too rapid OR closing the inside door equalization ball valve is desired, THEN do the following:

_______ a. JOG the inside door CLOSE pushbutton to close die equalization ball valve.

NOTE: IF the equalization ball valve is jogged open and the full closed limit has NOT cleared, THEN the valve may need to be sufficiently opened to allow clearing the limit, before the valve will operate to the closed position.

b. IF the equalization ball valve will NOT close, THEN do the following:

_______ 1. Close 1-CE-2, Personnel Hatch Door 1B Equalizing Line Isol Vv.

_______ 2. JOG the inside door OPEN pushbutton to sufficiently open the equalization ball valve, to clear the full closed limit.

3. Depress inside door **CLOSE** pushbutton to close the equalization ball valve.

_______ 4. Open 1-CE-2, Personnel Hatch Door 1B Equalizing Line Isol Vv.

_______ 5.1.18 When air flow is no longer heard passing through the pressure equalizing valve, <u>THEN</u> depress the inside door OPEN pushbutton.

5.1.19 Release the **OPEN** pushbunon <u>WHEN</u> the inside door has automatically opened <u>OR</u> the sound of the pump has a step change in pitch <u>OR</u> the inner door UNLOCKED light is lit

5.1.20 Ensure the following inner <u>AND</u> outer door pressure equalizing valves manual isolation valves are lek fully open:

- 1-CE-2, Personnel Hatch Door 1B Equalizing L i e Isol Vv
- 1-CE-3, Personnel Hatch Door 1A Equalizing L i e Isol Vv

5.1.21 Enter the Containment and depress the inside door **CLOSE** pushbutton.

_______ 5.1.22 The door will swing closed automatically; when the locking ring is in the locked position, <u>THEN</u> release the **CLOSE** pushbunon.

Completed ______________________________ Date: ____________

5.2 Leaving The Containment Through The Personnel Air Lock

______ 5.2.1 Review Precautions and Limitations.

______ 5.2.2 IF the Personnel Air Lock will be operated during Core Alterations, THEN issue Attachment 4, Personnel Air Lock Operator Duties and Responsibilities Guideline During Core Alterations. to the air lock operator.

* *

WARNING: If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

* *

NOTE: If a malfunction occurs while exiting Containment, the Control Room should be contacted for direction.

NOTE: Before opening or closing air lock doors, ensure all equipment and personnel are clear of the door swing area.

NOTE: When hydraulic operation of door is complete, the door position light will shift (locked, unlocked) and the sound of the pump will have a step change in pitch due to unloading.

NOTE: Excessive operation of the hydraulic pump may cause the motor to overheat and thermal out. If the hydraulic pump stops unexpectedly, the Control Room should be notified. There is a thermal overload reset pushbutton located inside the Personnel Hatch in the control cabinet that may be met with SRO concurrence.

______ 5.2.3 Verify no one is in the Air Lock.

______ 5.2.4 Verify the "DOOR IN USE" red light is OFF.

______ 5.2.5 Depress the inside door, OPEN pushbutton.

______ 5.2.6 Release the OPEN pushbutton when the inside door has automatically opened OR the locking ring is completely disengaged.

______ 5.2.7 Enter the Air Lock and depress the inside door CLOSE pushbutton.

______ 5.2.8 Release the CLOSE pushbutton when the door position light indicates LOCKED and/or the pump sound changes pitch.

NOTE: If pressure change rate is too fast JOG the outside door, CLOSE pushbutton.

______ 5.2.9 Depress the outside door OPEN pushbutton until a comfortable pressure change rate is achieved.

5.2.10 IF the pressurization is too rapid OR closing the outside door equalization ball valve is desired, THEN do the following:

______ a. JOG the outside door CLOSE pushbutton to close the equalization ball valve.

NOTE: IF the equalization ball valve is jogged open and the full closed limit has NOT cleared, THEN the valve may need to be sufficiently opened to allow clearing the limit, before the valve will operate to the closed position.

b. IF the equalization ball valve will NOT close, THEN do the following:

______ 1. Close 1-CE-3, Personnel Hatch Door 1A Equalizing Line Isol Vv.

______ 2. JOG the outside door OPEN pushbutton to sufficiently open the equalization ball valve, to clear the full closed limit.

______ 3. Depress outside door CLOSE pushbunon to close the equalization ball valve.

______ 4. Open 1-CE-3, Personnel Hatch Door 1A Equalizing Line Isol Vv

_____ 5.2.11 When air flow is no longer heard passing **through** the equalizing valve, **THEN** depress **the** outside door OPEN pushbutton.

_____ 5.2.12 Release the OPEN pushbutton **WHEN** the **locking** ring is in the unlocked position and/or a step change in pitch is heard.

5.2.13 Ensure the inner **AND** outer door pressure **equalizing** valves **manual** isolation valves are left **fully** open.

_____ • 1-CE-2, **Personnel** Hatch Door 1B Equalizing **Line** Isol Vv

_____ • 1-CE-3, Personnel Hatch Door 1A Equalizing Line Isol Vv

_____ 5.2.14 Manually open the Air Lock outside door, enter the Auxiliary Building, then manually shut the outside door.

_____ 5.2.15 Depress the outside door, CLOSE pushbutton.

_____ 5.2.16 **Release** the CLOSE pushbutton WHEN ttie **locking ring** is in the locked position.

_____ 5.2.17 **IF** the ***air*** lock is **NOT** being **used** for multiple entries, **THEN turn** off the containment lights.

_____ 5.2.18 **IF** Containment integrity is **required, THEN perform I-FT-62.4** within 7 days.

_____ 5.2.19 **IF Containment** integrity is NOT required, **THEN** enter into die Action Statement **Log** to perform 1-PT-62.4 prior to establishing Containment integrity for Mode 4 entry.

5.2.20 IF either Emergency door or associated equalizing vdve was operated, THEN do one ofthe following as required by Unit 1 operating Mode:

- ______ • IF Containment integrity is required, THEN perform 1-PT-62.5 within 7 days.
- ______ • IF Containment integrity is NOT required, THEN enter into the Action Statement Log to perform 1-PT-62.5 prior *to* establishing Containment integrity for Mode 4 entry.

Completed ______________________ Date: __________

5.3 Entering Containment By Manual Operation Of The Personnel Air Lock

NOTE: This procedure to be used only upon the failure of the electric/hydraulic system when the entry party is outside of the Personnel Hatch.

NOTE: If a malfunction occurs while entering Containment, the Control Room should be contacted for direction.

______ 5.3.1 Review Precautions and Limitations.

______ 5.3.2 IF this entry will be a multiple bottle entry and the Containment is at subatmospheric conditions, THEN the Containment Atmosphere must be sampled for O_2 AND the O_2 must be acceptable prior to swapping breathing air bottles in Containment.

______ 5.3.3 Ensure that the Containment lights are turned on.

* *

WARNING: If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

* *

______ 5.3.4 Verify no one is in the Air Lock.

NOTE: See Attachments 1, 2, and 3 for valve locations.

NOTE: The manual hand pump does NOT operate the pressure equalization valve

______ 5.3.5 Close 1-CE-102A, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE.

______ 5.3.6 Close 1-CE-102B, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE.

______ 5.3.7 Open 1-CE-103A, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.3.8 Open 1-CE-103B, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.3.9 Position Hand Pump valve 1-CE-100, 2A EXTERIOR HAND PUMP SELECTOR VALVE to OPEN.

* *

CAUTION: To preclude inadvertent opening of the emergency door the handle should not be placed in the full open position.

* *

NOTE: The control handle operates both the bleed valve and the emergency door.

5.3.10 Equalize pressure by using the emergency door bleed valve as follows:

a. Slowly rotate the door control handle in the open direction to establish a flow of air.

_______ b. When the flow of air stops, <u>THEN</u> position the control handle to the closed position.

_______ 5.3.11 Hand pump until the locking ring is in the unlocked position.

_______ 5.3.12 Open the door.

_______ 5.3.13 Close 1-CE-103A, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.3.14 Close 1-CE-103B, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.3.15 Open 1-CE-102A, ELECT PP AND 1.4 HAND PP TO DOOR 1A ISOL VALVE.

_______ 5.3.16 Open 1-CE-102B, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE.

_______ 5.3.17 Enter the air lock and close the door.

_______ 5.3.18 Open 1-CE-104A, 1A INTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.3.19 Open 1-CE-104B, 1A INTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.3.20 Close **1-CE-150A**, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

_______ 5.3.21 Close **1-CE-150B,** PERSONNEL HATCH **ELECT** PP TO DOOR 1A ISOL VALVE.

_______ 5.3.22 Position hand pump valve 1-CE-101, 1A INTERIOR HAND PUMP **SELECTOR VALVE** to close.

_______ 5.3.23 **Hand pump** until **locking ring** is in the **locked** position.

_______ 5.3.24 Close **1-CE-104A,** 1A INTERIOR NAND PUMP ISOLATION VALVE.

_______ 5.3.25 **Close** 1-CE-104B, 1A INTERIOR HAND PUMP ISOLATION **VALVE.**

_______ 5.3.26 Open **1-CE-150A**, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL **VALVE.**

_______ 5.3.27 Open 1-CE-150B, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

_______ 5.3.28 Open 1-CE-105A, 1B INTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.3.29 Open 1-CE-105B, 1B INTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.3.30 Close **1-CE-151A,** PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE.

_______ 5.3.31 Close **1-CE-151B,** PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE.

_______ 5.3.32 Position hand pump valve 1-CE-108, 1B INTERIOR HAND PUMP SELECTOR VALVE **to** OPEN.

* *

CAUTION: To preclude inadvertent opening of the emergency door the handle should not be placed in the full open position.

* *

NOTE: The control handle operates both the bleed valve and the emergency door.

5.3.33 Equalize pressure by using the emergency door bleed valve as follows:

______ a. Slowly rotate the door control handle in the open direction to establish a flow of air.

______ b. When the flow of air stops, THEN position the control handle to the closed position.

______ 5.3.34 Hand pump until the locking ring is in the unlocked position.

______ 5.3.35 Open the door.

______ 5.3.36 Close 1-CE-105A, 1B INTERIOR HAND PUMP ISOLATION VALVE.

______ 5.3.37 Close 1-CE-105B, 1B INTERIOR HAND PUMP ISOLATION VALVE.

5.3.38 Ensure the inner and outer door pressure equalizing valves manual isolation valves are left open.

______ • 1-CE-2, Personnel Hatch Door 1B Equalizing Line Isol Vv

______ • 1-CE-3, Personnel Hatch Door 1A Equalizing Line Isol Vv

______ 5.3.39 Open 1-CE-151A, PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE.

______ 5.3.40 Open 1-CE-151B, PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE 1.

_____ **5.3.41** **Enter** the containment and dose the door.

_____ **5.3.42** Close **1-CE-106A, ELECT** PP AND 1B HAND PP TO W O R 1B **ISOL** VALVE.

_____ **5.3.43** Close 1-CE-106B, ELECT PP AND 1B HAND PP TO DOOR 1B **ISOL** VALVE.

_____ **5.3.44** **Open 1-CE-107A,** 2B EXTERIOR **HAND** PUMP ISOLATION VALVE.

_____ **5.3.45** **Open** 1-CE-107B, 2B EXTERIOR HAND **PUMP** ISOLATION VALVE.

_____ 5.3.46 **Position** hand **pump** valve **1-CE-109, 2B EXTERIOR HAND** PUMP SELECTOR VALVE to close.

_____ **5.3.47** **Hand pump until locking ring is in the locked position.**

_____ **5.3.48** **Close 1-CE-107A,** 2R EXTERIOR HAND PUMP ISOLATION VALVE.

_____ **5.3.49** Close **1-CE-107B,** 2B EXTERIOR **HAND** PUMP ISOLATION VALVE.

_____ **5.3.50** **Open** 1-CE-IMA. ELECT PP AND 1B **HAND** PP TO DOOR 1B **ISOL** VALVE.

_____ 5.3.51 Open 1-CE-106B, ELECT PP AND 1B HAND PP TO DOOR 1B ISOL VALVE.

Completed: ______________________________ Date: ____________

5.4 Leaving The Containment By Manual Operation Of The Personnel Air Lock

NOTE: This procedure to be used only upon the failure of the electric/hydraulic system.

NOTE: If a malfunction occurs while exiting Containment, the Control Room should be contacted for direction.

_______ 5.4.1 Review Precautions and Limitations.

* *

WARNING: If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

* *

_______ 5.4.2 Verify no one is in the Air Lock.

NOTE The manual hand pump does NOT operate the pressure equalization valve.

NOTE: See Attachments 1, 2, and 3 for valve locations.

_______ 5.4.3 Close 1-CE-106A, ELECT PP AND 1B HAND PP TO DOOR 1B ISOL VALVE.

_______ 5.4.4 Close 1-CE-106B, ELECT PP AND 1B HAND PP TO DOOR 1B ISOL VALVE.

_______ 5.4.5 Open 1-CE-107A, 2B EXTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.4.6 Open 1-CE-107B, 2B EXTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.4.7 Position Hand Pump valve 1-CE-109, 2B EXTERIOR HAND PUMP SELECTOR VALVE to OPEN.

* *

CAUTION: To preclude inadvertent o p i n g of the emergency door the handle should not be placed in the full open position.

* *

NOTE: The control handle operates both the bleed valve and the emergency door.

5.4.8 Equalize pressure by wing the emergency door bleed valve as follows:

______ a. Slowly rotate the door control handle in the open direction to establish a flow of air.

______ b. When the flow of air stops, <u>THEN</u> position the control handle to the closed position.

5.4.9 Hand pump until the locking ring is in the unlocked position.

______ 5.4.10 Open the door.

______ 5.4.11 Close 1-CE-107A, 2B EXTERIOR HAND PUMP ISOLATION VALVE.

______ 5.4.12 Close 1-CE-107B, 2B EXTERIOR HAND PUMP ISOLATION VALVE.

______ 5.4.13 Open 1-CE-106A, ELECT PP AND 1B HAND PP TO DOOR 1B ISOL VALVE.

______ 5.4.14 Open 1-CE-106B, ELECT PP AND 1B WAND PP TO DOOR 1B ISOL VALVE.

______ 5.4.15 Enter the air lock and close the door.

______ 5.4.16 Open 1-CE-105A, 1B INTERIOR HAND PUMP ISOLATION VALVE.

______ 5.4.17 Open **1-CE-105B,** PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

5.4.18 Close 1-CE-151A, PERSONNEL HATCH ELECT PP TO DOOR 1B **ISOL** VALVE.

5.4.19 Close **1-CE-151B,** PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE

______ 5.4.20 Position **hand** pump valve 1-CE-108, 1B INTERIOR HAND PUMP SELECTOR VALVE **to** close.

5.4.2 1 Hand **pump** until locking **ring** is in **the locked** position.

5.4.22 Close 1-CE-105A, 1B INTERIOR HAND PUMP ISOLATION VALVE.

5.4.23 Close **1-CE-105B,** BB INTERIOR HAND PUMP ISOLATION VALVE.

______ 5.4.24 Open 1-CE-151A, PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL VALVE.

5.4.25 Open **1-CE-151B,** PERSONNEL HATCH ELECT PP TO DOOR 1B ISOL **VALVE.**

______ 5.4.26 Open 1-CE-104A, 1A INTERIOR **HAND** PUMP ISOLATION VALVE.

______ **5.4.27** Open 1-CE-104B, 1A INTERIOR HAND PUMP ISOLATION VALVE.

5.4.28 Close 1-CE-150A, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

5.4.29 Close **1-CE-150B,** PERSONNEL HATCH **ELECT** PP TO DOOR 1A ISOL VALVE.

5.4.30 Position hand pump valve I-CE-101. 1A **INTERIOR** HAND **PUMP** SELECTOR VALVE to open.

* *

CAUTION: To preclude inadvertent opening of the emergency door the handle should not be placed in the full open position.

* *

NOTE: The control handle operates both the bleed valve and the emergency door.

5.4.31 Equalize pressure by using the emergency door bleed valve as follows:

_______ a. Slowly rotate the door control handle in the open direction to establish a flow of air.

_______ b. When the flow of air stops, THEN position the control handle to the closed position.

_______ 5.4.32 Hand pump until the locking ring is in the unlocked position.

_______ 5.4.33 Open the door.

_______ 5.4.34 Close 1-CE-104A, 1A INTERIOR HAND PUMP ISOLATION VALVE.

_______ 5.4.35 Close 1-CE-104B, 1A INTERIOR HAND PUMP ISOLATION VALVE.

5.4.36 Ensure the inner and outer door pressure equalizing valves manual isolation valves are left open.

_______ • 1-CE-2, Personnel Hatch Door 1B Equalizing Line Isol Vv

_______ • 1-CE-3, Personnel Hatch Door 1A Equalizing Line Isol Vv

_______ 5.4.37 Open 1-CE-150A, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

_______ 5.4.38 Open 1-CE-150B, PERSONNEL HATCH ELECT PP TO DOOR 1A ISOL VALVE.

_______ 5.4.39 Leave the air lock and close the door.

_______ 5.4.40 Close 1-CE-102A, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE.

_______ 5.4.41 Close 1-CE-102B, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE

_______ 5.4.42 Open 1-CE-103A, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.4.43 Open 1-CE-103B, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.4.44 Position hand pump valve 1-CE-100, 2A EXTERIOR HAND PUMP SELECTOR VALVE to close.

_______ 5.4.45 Hand pump until locking ring is in the locked position.

_______ 5.4.46 Close 1-CE-103A, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.4.47 Close 1-CE-103B, 2A EXTERIOR HAND PUMP ISOL VALVE.

_______ 5.4.48 Open 1-CE-102A, ELECT PP AM) BA HAND PP TO DOOR 1A ISOL VALVE.

_______ 5.4.49 Open 1-CE-102B, ELECT PP AND 1A HAND PP TO DOOR 1A ISOL VALVE.

_______ 5.4.50 <u>IF</u> the air lock is <u>NOT</u> being used for multiple entries, <u>THEN</u> turn off the Containment lights.

_______ 5.4.51 <u>IF</u> containment integrity is required, <u>THEN</u> perfom 1-PT-62.4 and 1-PT-62.5 within 7 days.

______ 5.4.52 **IF Containment integrity is** NOT **required, THEN enter into the Action Statement Log to** perfom **1-PT-62.4 and 1-PT-62.5 prior to establishing Containment integrity for Mode 4** *entry.*

Completed ______________________________ Date: ____________

5.5 Operation Of The Personnel Air Lock Emergency Doors

* *

CAUTION: These doors and equalizing valves are not interlocked to each other or to the main doors.

* *

NOTE: These doors are to be used only for emergency entrance or exit from the airlock.

_______ 5.5.1 Review Precautions and Limitations.

* *

WARNING: If any personnel are in the air lock, DO NOT open either air lock door without their knowledge. Pressure changes which result could cause injury.

* *

_______ 5.5.2 Verify no one is in the Air Lock.

NOTE: Full handcrank travel, from full closed to full open is 270" (3/4 turn)

_______ 5.5.3 Slowly rotate handcrank UNTIL the locking arms fall into a detent OR until a comfortable pressure change rate is achieved.

5.5.4 When air flow is no longer heard passing through the equalizing valve, THEN rotate the handcrank UNTIL it contacts the open stop and push/or pull the door open.

_______ 5.5.5 Step through, THEN close door AND rotate the handcrank 270° in the opposite direction UNTIL it contacts the closed stop.

5.5.6 IF Containment integrity is required, THEN perform 1-PT-62.5 within 7 days.

_______ 5.5.7 IF Containment integrity is NOT required, THEN enter into the Action Statement Log to perform 1-PT-62.5 prior to establishing Containment integrity for Mode 4 entry.

Completed: ______________________________ Date: ____________

5.6 Opening Both Inner And Outer Personnel Air Lock Doors For Maintenance Access

5.6.1 Review Precautions and Limitations.

NOTE: The TRM 7.2 Action to restore fire doors to operable within 7 days is not applicable to the Containment Personnel Hatches in Modes 5 and 6.

5.6.2 IF core alterations OR movement of irradiated fuel within the Containment are in progress, THEN ensure the requirements of T.S. 3.9.4 and 3.9.7 are met, as follows:

a. One personnel airlock door is operable, as follows:

- At least one door is capable of being closed.
- The door is NOT blocked and no cables or hoses are run through the airlock (The protective cover on the airlock door sealing surface does NOT constitute a closure restriction as this cover may be easily removed by the designated airlock closure individual.)
- A designated individual is continuously available to close the airlock door. This person MUST be in Containment on the 291 ft level.

b. IF fuel movement is in progress, THEN there is at least 23 feet of water above the vessel flange.

5.6.3 Obtain permission from the Shift supervisor.

5.6.4 Depress the outside doors OPEN pushbutton.

5.6.5 Release the OPEN pushbutton when the locking ring is in the unlocked position.

5.6.6 Manually open the outside door.

_______ 5.6.7 Manually depress and hold the outside door limit switch L-3 (see Attachment 2).

_______ 5.6.8 Depress the outside door CLOSE pushbunon to rotate the locking ring to the locked position, then release the outside door limit switch.

* *

CAUTION: Extreme care must be taken when opening the inner door due to possible pressure differences which could exist.

* *

_______ 5.6.9 Ensure all equipment and personnel are clear of the door swing area.

_______ 5.6.10 Depress the inner door OPEN pushbutton until the pressure equalizing vdve is open.

5.6.11 Check for air flow through the pressure equalizing line. IF air flow exists, THEN it is an indication of a pressure difference between the Auxiliary Building and the Containment.

_______ 5.6.12 Coordinate with the Control Room to manipulate the containment and/or Auxiliary Building supply and exhaust fans as necessary to equalize pressure between the Auxiliary Building and the Containment.

_______ 5.6.13 With pressures equalized, depress the inner door OPEN pushbutton to open the inner door.

5.6.14 Notify the Unit SRO to enter into the Action Statement Statu Log that the Containment boundary is breached through the Personnel Match.

_______ 5.6.15 With both doors open, have the Control Rooni shift fans as necessary to establish air flow from the Auxiliary Building into the Containment.

* *

WARNING: **Ensure both doors *are* left open. If the inner door is closed, the next individuals to open the inner door may be unaware that extreme care must be taken when opening the inner door due to possible pressure differences which could exist. Personnel injury or equipment damage may result. (Reference 2.4.6)**

* *

______ 5.6.16 **IF** *the* doors will remain open for an extended period of time, **THEN** if desired, place protective covers over the door seating surfaces.

Completed: ________________________________ Dare: ____________

5.7 Returning The Personnel Air Lock Doors To Normal Status After Being Both Open

_____ 5.7.1 Verify Initial Conditions are satisfied.

_____ 5.7.2 Review Precautions and Limitations.

_____ 5.7.3 Ensure that 1-OP-18.1A, Containment Access System Valve Checkoff, has been performed. (Reference 2.4.1)

_____ 5.7.4 Verify/Remove any covers that were placed over the door seating surfaces, and verify surfaces are clean.

_____ 5.7.5 Coordinate with the Control Room to manipulate the containment and/or Auxiliary Building supply and exhaust fans as necessary to equalize pressure between the Auxiliary Building and the Containment.

_____ 5.7.6 Verify Inner Door Locking Ring indicates in the FULLY OPEN position. (Reference 2.4.2)

_____ 5.7.7 Pull the inner door closed and hold the inner door in the CLOSED position until the Locking Ring engages, when the CLOSED pushbutton is depressed. (Reference 2.4.2)

_____ 5.7.8 Depress the inner door CLOSE pushbutton to close the inner door.

_____ 5.7.9 Manually depress the outside door limit switch L-3 (See Attachment 2).

_____ 5.7.10 Depress the outside door OPEN pushbutton to rotate the locking ring tu the unlocked position, then release the outside door limit switch.

_____ 5.7.11 Pull the outer door closed and hold the outer door in the CLOSED position until the Locking Ring engages, when the Closed pushbutton is depressed. (Reference 2.4.2)

_____ 5.7.12 Depress the outside door CLOSE pushbutton to close the outside door'

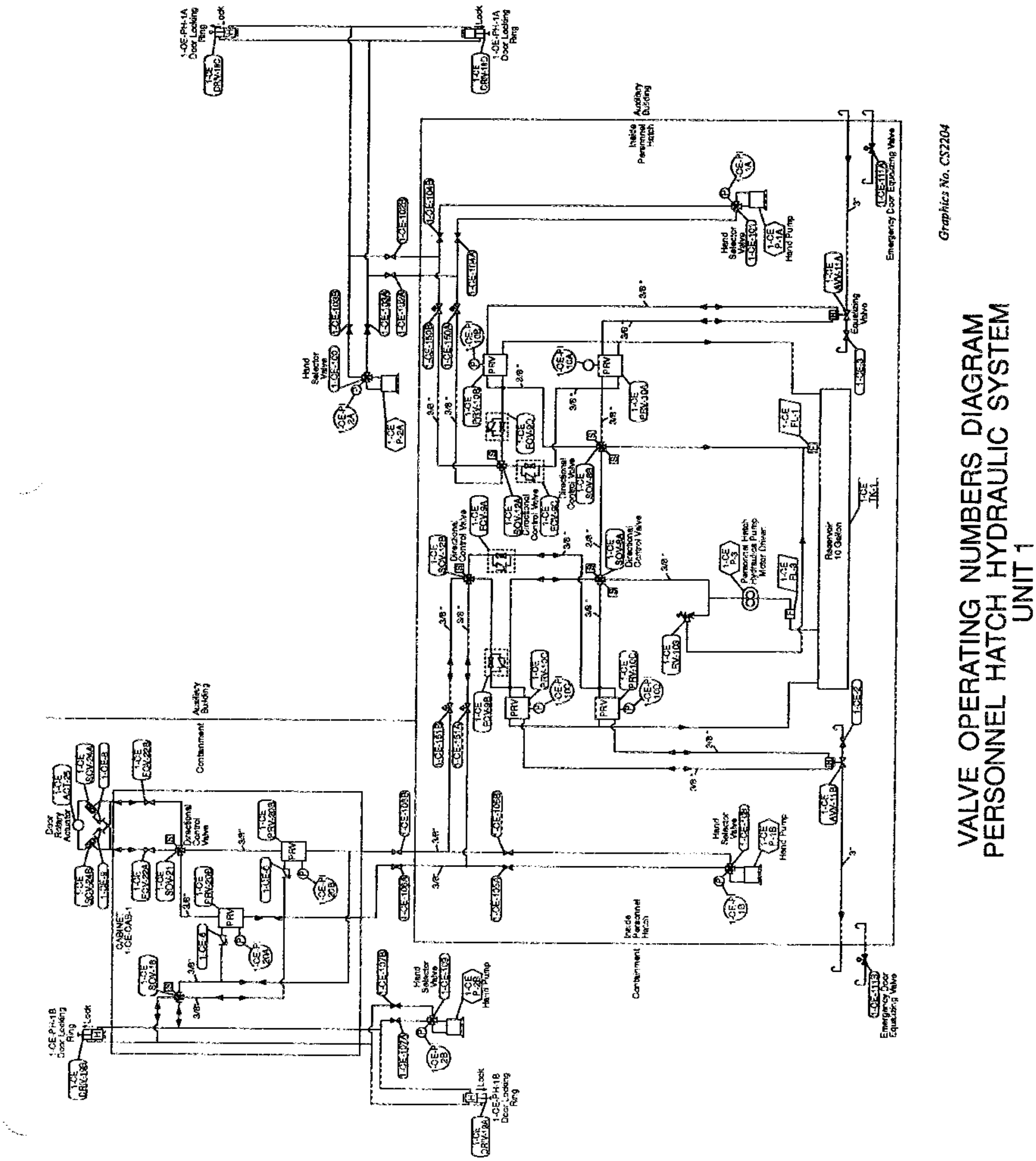
______ 5.7.13 Notify the Unit SRO that the Personnel Hatch doors are closed and the Abnormal Status Log entry for Containment breach may be cleared.

Completed: ______________________________ Date: ______________

ATTACHMENT 1

(Page 1 of 1)

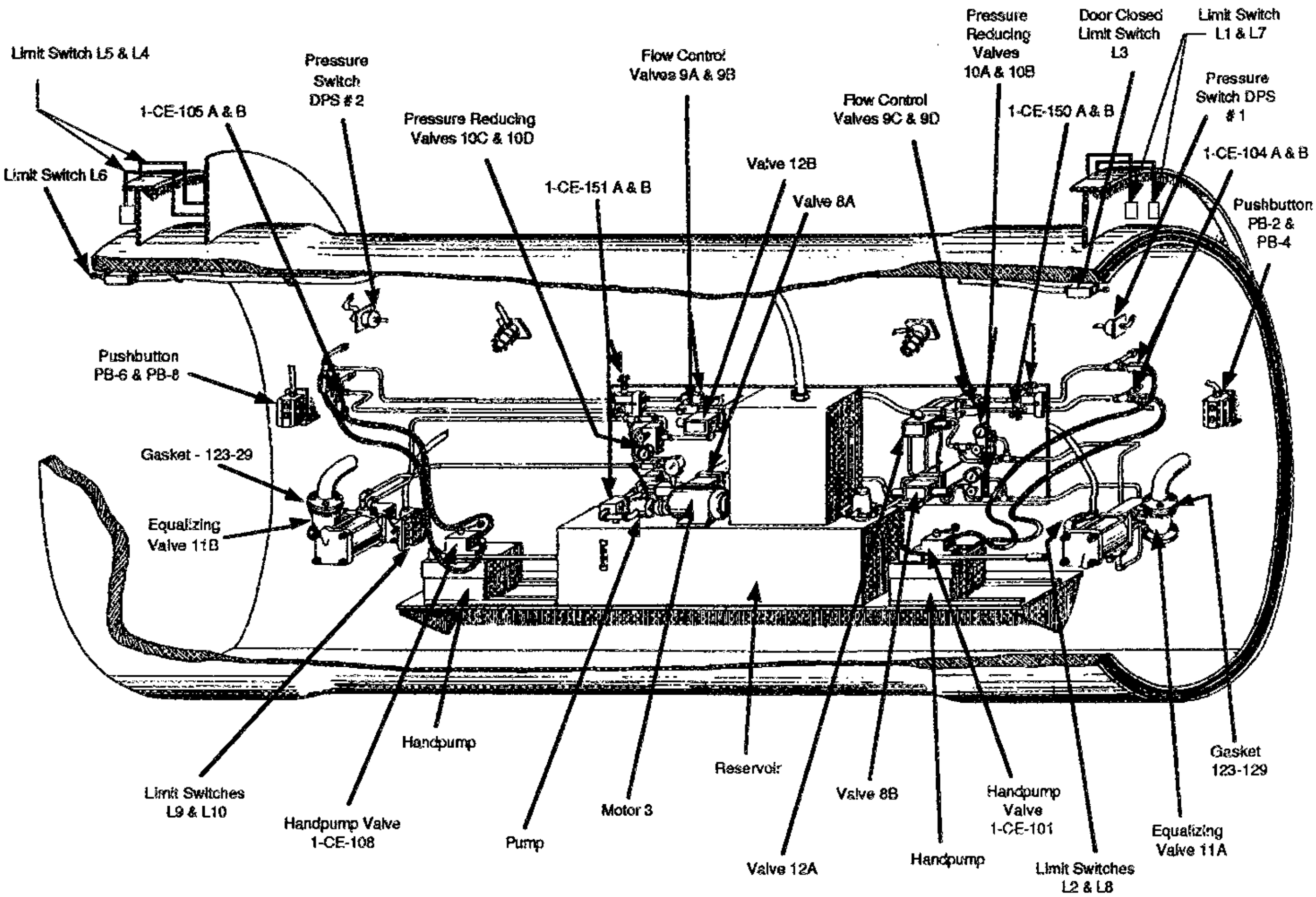
PERSONNEL HATCH HYDRAULIC SYSTEM



ATTACHMENT 2

(Page 1 of 1)

PERSONNEL LOCK INTERIOR VIEW



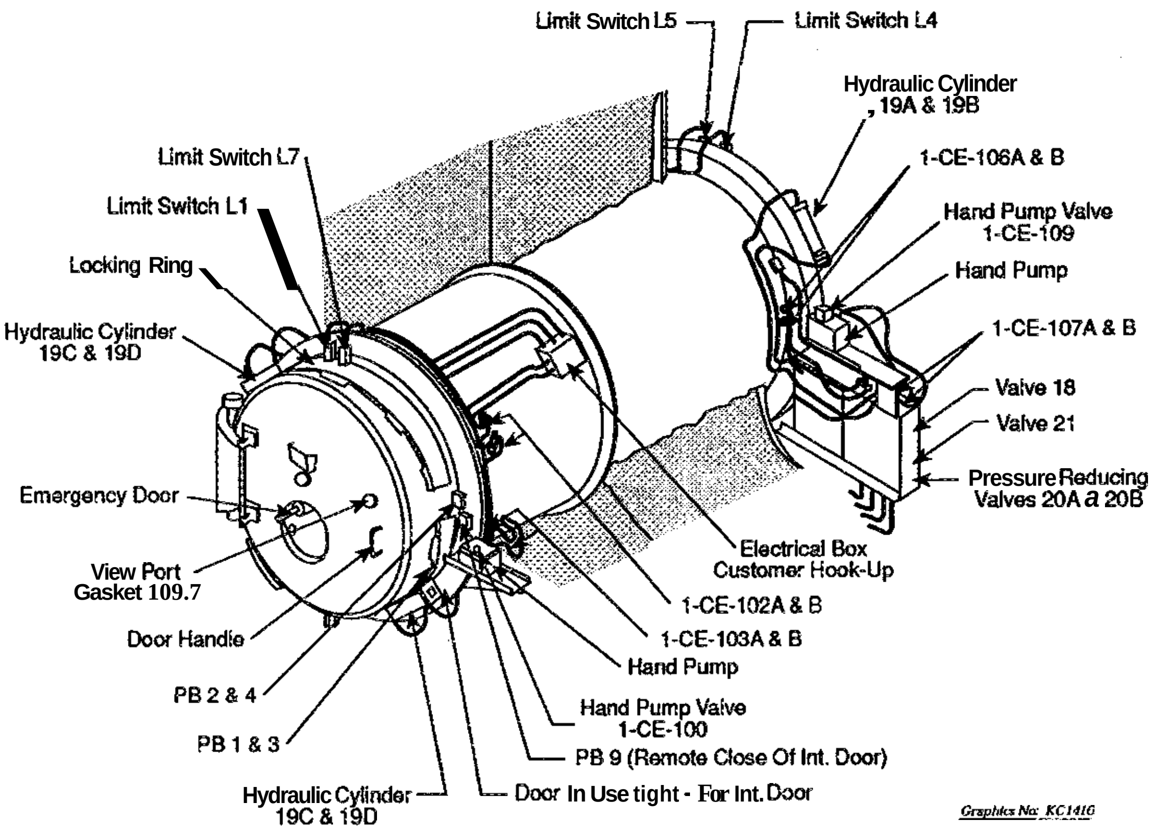
PERSONNEL LOCK INTERIOR VIEW

Graphics No. CB326B

ATTACHMENT 3

(Page 1 of 1)

PERSONNEL LOCK EXTERIOR VIEW



PERSONNEL LOCK EXTERIOR VIEW

ATTACHMENT 4

(Page 1 of 1)

PERSONNEL AIR LOCK OPERATOR DUTIES AND RESPONSIBILITIES GUIDELINE DURING CORE ALTERATIONS

1.0 Maintain at least one Personnel Air Lock door closed at all times.

2.0 Ensure the equalizing valves on each Personnel Air Lock Escape door remain closed.

3.0 Operate the Air Lock, as required to allow Containment Ingress and Egress, ensuring the conditions of Steps 1.0 and 2.0 are met.

4.0 IF equipment deficiencies are noted, THEN submit Work Requests, as required.

5.0 WHEN Core Alterations are complete or suspended, THEN with Refueling SRO permission, open both Personnel Air Lock doors in accordance with Section 5.6.

VIRGINIA POWER

NORTH ANNA POWER STATION

PROCEDURE NO: 1-OP-18.1A		
UNIT NO: 1	REVISION NO: 8	
PROCEDURE TYPE: OPERATING	EFFECTIVE DATE:	EXPIRATION DATE: N/A

PROCEDURE TITLE:

VALVE CHECKOFF-CONTAINMENT ACCESS

1-OP-18.1 D008

REVISION SUMMARY:

- Revised procedure to incorporate OP 96-0097 which deleted the ACTUAL and LBL columns from each valve lineup.

PAR Incorporation Plan

- Incorporated E-PAR P1 which adds Technical Reference 2.1.2 and changes the noun name of 1-CE-4 per Engineerings response to DR 96-2045.

FOR ELECTRONIC DISTRIBUTION - APPROVAL ON FILE

PROBLEMS ENCOUNTERED: ☐ Yes ☐ No NOTE: If yes, note problems in Remarks.

REMARKS:

(use back for additional space)

SHIFT SUPERVISOR: DATE:

1.0 PURPOSE

To provide a normal valve lineup for Containment Access.

2.0 REFERENCES

2.1 Technical References

2.1.1 Instruction Manual Personnel Air Lock, Chicago Bridge and Iron

2.1.2 DR 96-2045, Incorrect Mark Numbers in PT and OP-1A

VALVE SHEET CHECKOFF

SYSTEM: CONTAINMENT ACCESS

VALVE NO.	DESCRIPTION	NORM	DATE	INIT	VER
AUXILIARY BUILDING SIDE					
1-CE-4	Personnel Air Lock Test Connection Isolation	Closed			
1-CE-100	2A Exterior Hand Pump Selector Valve	Off			
1-CE-102A	Elect PP and 1A Hand PP to Door 1A Isol Valve	Open			
1-CE-102B	Elect PP and 1A Hand PP to Door 1A Isol Valve	Open			
1-CE-103A	2A Exterior Hand Pump Isol Valve	Closed			
1-CE-103B	2A Exterior Hand Pump Isol Valve	Closed			
INSIDE PERSONNEL LOCK, OUTER DOOR					
1-CE-3	Personnel Hatch Door 1A Equalizing Line Isol Vv	Open			
1-CE-104A	1A Interior Hand Pump Isolation Valve	Closed			
1-CE-104B	1A Interior Hand Pump Isolation Valve	Closed			
1-CE-150A	Personnel Hatch Elect PP to Door 1A Isol Valve	Open			
1-CE-150B	Personnel Hatch Elect PP to Door 1A Isol Valve	Open			
1-CE-101	1A Interior Hand Pump Selector Valve	Off			
1-CE-111A	Personnel Hatch Emergency Door 1C Equalizing Valve	Closed			

VALVE SHEET CHECKOFF

SYSTEM: CONTAINMENT ACCESS

VALVE NO.	DESCRIPTION	NORM	DATE	INIT	VER
INSIDE PERSONNEL LOCK, INNER DOOR					
1-CE-2	Personnel Hatch Door 1B Equalizing Line Isol Vv	Open			
1-CE-151A	Personnel Hatch Elect PP to Door 1B Isol Valve	Open			
1-CE-151B	Personnel Hatch Elect PP to Door 1B Isol Valve	Open			
1-CE-108	2A Exterior Hand Pump Selector Valve	Off			
1-CE-105A	1B Interior Hand Pump Isolation Valve	Closed			
1-CE-105B	1B Interior Hand Pump Isolation Valve	Closed			
1-CE-111B	Personnel Hatch Emergency Door 1D Equalizing Valve	Closed			
INSIDE CONTAINMENT					
1-CE-106A	Elect PP and 1B Hand PP to Door 15 Isol Valve	Open			
1-CE-106B	Elect PP and 1B Hand PP to Door 1B Isol Valve	Open			
1-CE-107A	2B Exterior Hand Pump Isolation Valve	Closed			
1-CE-107B	2B Exterior Hand Pump Isolation Valve	Closed			
1-CE-109	2B Exterior Hand Pump Selector Valve	Off			

VIRGINIA POWER

NORTH ANNA POWER STATION

PROCEDURE NO: 1-PT-62.4			
UNIT NO: 1	REVISION: 20		
PROCEDURE TYPE: PERIODIC TEST	DEPARTMENT: OPERATIONS	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE:

PERSONNEL AIR LOCK SEAL LEAKAGE

TEST FREQUENCY: Within 7 days after each Containment entry AND Prior to reestablishing Containment integrity	UNIT CONDITIONS REQUIRING TEST: MODES 1, 2, 3, AND4

SPECIAL CONDITIONS: Tech Spec 4.0.2 (ITS SR 3.0.2) and TRM GR 1.0.7 (ITS TRM TSR 3.0.2) are not applicable

SURV REQ	ITS					PMT

REVISION SUMMARY:

- Changed ITS TRM TSR 3.6.2.7 to ITS TRM TSR 3.6.2.6 due to consolidation of TSR 3.6.2.6 and 3.6.2.7 in Purpose and Step 2.2.7.

Writer: J. Goerge / A. Shelton	Reviewer: D. Hawkins

ELECTRONIC DISTRIBUTION — APPROVAL ON FILE

REASON FOR TEST (CHECK APPROPRIATE BOX)

☐ Surveillance ☐ Post-Maintenance Work Order Number (Post-Maintenance Only): ____________

TEST PERFORMED BY (SIGNATURE):	DATE STARTED:	DATE COMPLETED:

TEST RESULT (CHECK APPROPRIATE BOX): ☐ Sat ☐ Unsat ☐ Partial	WORK REQUEST NUMBERS AND DATE:

THE FOLLOWING PROBLEM(S) WERE ENCOUNTERED AND CORRECTIVE ACTIONS TAKEN:

____

____

____

____

____________ (use back for additional space)

COGNIZANT SUPERVISOR OR DESIGNEE:	DATE:
ADDITIONAL REVIEWS: System Engineer:	DATE:

TABLE OF CONTENTS

1.0 PURPOSE

To leak test the personnel air lock door seals in accordance with Tech Specs 4.6.1.3.a (ITS SR 3.6.2.1), 4.6.1.1.b (ITS -None), and TRM SR 5.3.1.6 (ITS TRM TSR3.6.2.6) and SR 5.3.1.7 (ITS TRM TSR 3.6.2.6).

2.0 REFERENCES:

2.1 Source Documents

2.1.1 UFSAR Section 3.1.46, Provisions for Containment Testing and Inspection, Criterion 53

2.1.2 10CFR 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

2.1.3 NRC Regulatory Guide 1.163

2.2 Technical Specifications

2.2.1 Unit 1 Tech Spec 3.6.1.1 (ITS 3.6.1)

2.2.2 Unit 1 Tech Spec 3.6.1.3 (ITS 3.6.2)

2.2.3 Unit 1 Tech Spec 4.6.1.1.b (ITS -None)

2.2.4 Unit 1 Tech Spec 4.6.1.3.a (ITS SR 3.6.2.1)

2.2.5 Technical Requirements Manual TR 5.3.1 (ITS TRM 3.6.2)

2.2.6 Technical Requirements Manual SR 5.3.1.6 (ITS TRM TSR 3.6.2.6)

2.2.7 Technical Requirements Manual SR 5.3.1.7 (ITS TRM TSR 3.6.2.6)

2.3 **Technical References**

2.3.1 11715-FM-82C

2.3.2 Quality Control Inspection Report N-85-174, N-85-709

2.3.3 EWR 85-594 (Assigned mark numbers to the seals)

2.3.4 Calculation Basis - Approved 02-02-89

2.3.5 Merck Index Chemical Encyclopedia, 1976, Section 4319, Page 581

2.3.6 ET CME-98-009, Rev. 1, Response to Teflon Tape. Issues

2.3.7 ET CME-99-0003, Rev. 0, Cleaning Teflon Tape Connections

2.4 Commitment **Documents**

2.4.1 CTS Assignment 02-96-1802, Commitment 001, Implement 10CFR 50 Appendix J, Option B Test Criteria, Tech **Spec** Amendment 196

2.4.2 CTS Assignment 02-96-4000, Commitment 001, Implement (old) TRM 5.3, Criteria for Option B Performance-Based Requirements for **Containment** Leakage Rate **Testing** to 10 CFR 50 Appendix J

2.4.3 NEI-94-01, Industry Guideline for Implementing Perfontlance-Based Option of 10CFR ***50*** Appendix J

Init Verif

3.0 INITIAL CONDITIONS

_____ 3.1 The unit is in modes 1, 2, 3, 4, or 5.

_____ 3.2 Notify the Shift Supervisor on duty of the impending test and coordinate its performance through him.

4.0 PRECAUTIONS AND LIMITATIONS

_____ 4.1 Over pressurizing shall be avoided so as not to endanger personnel and equipment.

_____ 4.2 IF the bubbler test rig will be used in an area where the ambient temperature is less than 35°F, THEN the glycerin in the bubbler should be replaced with a mixture of water and glycerin using the following table.

MINIMUM AMBIENT TEMPERATURE	PERCENT WATER
30°	10
15T	30
-5T	50

_____ 4.3 IF a mixture of water and glycerin is used, THEN replace the glycerin when finished with the test.

_____ 4.4 IF the size of the bubbles observed are unusually small or large, THEN contact Engineering for evaluation.

_______ 4.5 Comply with the following guidelines when marking steps N/A

- IF the conditional requirements of a step do not require the action to be performed, THEN mark the step N/A.
- IF any other step is marked N/A, THEN have the Shift Supervisor (or designee) approve the N/A and submit a Procedure Action Request (PAR).
- IF this test is being performed as a Partial PT for re-testing a seal, THEN mark inappropriate steps N/A.

_______ 4.6 IF teflon tape is used when attaching test equipment to stainless steel piping, THEN the tape removal MUST be independently verified. (Reference 2.3.6 and 2.3.7)

_______ 4.7 The door seals shall be demonstrated operable after each opening, except when the air lock is being used for multiple entries, then at least once per 7 days.

_______ 4.8 Have the Operator and Independent Verifier attend an HP brief.

_______ 4.9 All Teflon tape must be removed from system test connections after use and the test connection must be rinsed. Teflon tape removal must be independently verified.

5.0 SPECIAL TOOLS AND EQUIPMENT

- 0.31 scfh, or less, full scale rotameter calibrated at 14.7 psia

  NQC No.:_________________ Cal Due Date:_____________

- Two (2) adjustable wrenches
- One (1) ≈1' x 3/8" tubing with female compression fittings on both ends

6.0 INSTRUCTIONS

______ **6.1** Leak test the bypass valve on the bubbler using **Attachment** 2.

6.2 Ensure the following valves are closed:

______ • **1-IA-601,** Pets Hatch Outer Boor 0-Ring Test **Ln** IA **Isol** Vv

______ • **1-IA-602,** Pers Hatch 0-Ring Test Line Test Conn **Isol** Valve

______ • 1-LA-609, **Pers** Hatch Inner Door O-Ring Test Ln IA **Isol** Vv

______ • 1-LA-610, Pers Hatch 0-Ring Test Line IA Drain **Vdve**

______ • **1-IA-611, O-Ring** Test **Eqpt** IA Sply 1-IA-FI-100 Out Isol Vv

______ • 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv

______ • 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 **Bypass** Valve

______ • 1-IA-617, O-Ring Test Eqpt IA Sply 1-IA-PI-119 Isol Valve

______ • 1-IA-614, I-IA-PCV-118 Met Isolation Vdve

______ 6.3 IF testing outer **seal** only, THEN N/A Step 6.4 AND GO TO Step 6.24.

NOTE: Steps 6.4 through 6.23 test the inner hatch sed.

______ 6.4 Verify the inner door is CLOSED.

6.5 Open the following valves:

______ 6.5.1 1-IA-617, O-Ring Test **Eqpt IA** Sply 1-IA-PI-119 **Isol** Valve

______ 6.5.2 **1-IA-614, 1-IA-PCV-118** Inlet Isolation Valve

______ *6.6* **Adjust** 1-IA-PCV-118, Pers Hatch **0-Ring** Test Eqpt IA Press Cont **Vdve, to** 44.1 - **45.0 psig as** read on 1-IA-PI-1 19.0-King Test Equipment IA Supply Pressure Indicator.

_______ 6.7 Open 1-IA-609, Pers Match Inner Door O-Ring Test Ln IA Isol Vv.

_______ 6.8 Open 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve and allow pressure to equalize between test rig and door "O" ring seals.

_______ 6.9 Open 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv.

_______ 6.10 Open or verify Open bubbler needle valve to the mid range position.

_______ 6.11 Open 1-IA-611, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Out Isol Vv.

_______ 6.12 Close 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve.

_______ 6.13 Leak test all fittings with Snoop or other approved leak check method.

_______ 6.14 Wait approximately 10 minutes to allow stabilization.

6.15 Observe bubble flow detector for 2 minutes.

_______ 6.15.1 Record the number of bubbles per minute and convert to SCFH:

_______ Bubbles per minute x 9.24×10^{-3} = _______ SCFH
INNER INNER

6.15.2 IF Bubble Frequency is greater than 60 bubbles per minute, THEN use a calibrated Rotameter with range between 0 and 8 SCFH calibrated at 45 psig to record flow below using Attachment 3.

PLOW _______ SCFH.
INNER

_______ 6.16 Crack open E-L4-602, Pers Ifatch O-Ring Test Line Test Conn Isol Valve.

_______ 6.17 Verify flow through 1-IA-602.

_______ 6.18 Close 1-IA-602, Pers Hatch 0-Ring Test Line Test Conn Isol Valve.

6.19 Close the following valves:

_______ 6.19.1 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv

_______ 6.19.2 1-IA-614, 1-IA-PCV-118 Inlet Isolation Valve

6.20 Do the following:

6.20.1 Slowly open one of the following valves to vent the test apparatus and ?he "O" ring. Mark N/A valve not used.

- _______ 1-IA-610, Pers Hatch 0-Ring Test Line IA Drain Valve
- _______ 1-IA-602, Peps Hatch 0-Ring Test Line Test Com Isol Valve

_______ 6.20.2 Open 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve.

6.21 Ensure the following valves are closed:

- _______ 1-IA-610, Peps Hatch O-Ring Test Line IA Drain Valve
- _______ 1-IA-602, Pers Hatch 0-Ring Test Line Test Conn Isol Valve
- _______ 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve

6.22 Close die following valves:

_______ 6.22.1 1-IA-611, 0-Ring Test Eqpt IA Sply 1-IA-FI-100 Out Isol Vv

_______ 6.22.2 1-IA-609, Pers Hatch Inner Door 0-Ring Test Ln IA Isol Vv

_______ 6.22.3 1-IA-617, O-Ring Test Eqpt IA Sply 1-It\-PI-119 Isol Valve

_______ 6.23 IF leakage is greater than 2.86 SCFH, THEN submit WR on 1-CE-PH-1B AND GO TO Step 6.44.

* *

CAUTION: The inner door should be satisfactorily tested prior to testing the outer door.

* *

NOTE: Steps 6.24 through 6.43 test the outer hatch seal.

_______ 6.24 Verify the outer door is CLOSED.

6.25 Open the following valves:

_______ 6.25.1 **1-IA-617, O-Ring** Test **Eqpt** IA Sply 1-IAPI-119 Isol Vdve

_______ 6.25.2 **1-IA-614, 1-IA-PCV-118** Inlet Isolation Valve

_______ 6.26 **Adjust** 1-IA-PCV-118 to 44.1 - 45.0 psig as read on 1-IA-PI-119.

_______ 6.27 Open 1-IA-601, Pers Hatch Outer Door 0-Ring Test Ln IA Isol Vv.

_______ 6.28 Opera **1-IA-613, O-Ring** Test Eqpt JA Sply 1-IA-FI-100 Bypass Valve and allow pressure to equalize between test rig **and** door **"O" ring seals.**

_______ 6.29 Open 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv.

_______ 6.30 Open or verify open bubbler needle valve to the mid range position.

_______ 6.31 Open 1-IA-611 0-Ring Test Eqpt IA Sply 1-IA-FI-100 **Out** Isol Vv.

_______ 6.32 Close 1-IA-613 0-Ring Test **Eqpt** IA Sply 1-IA-FI-100 **Bypass** Valve

_______ 6.33 Leak **test** all fittings with Snoop or other approved leak check method.

_______ 6.34 Wait approximately 10 minutes to allow stabilization.

6.35 Observe bubble flow detector fur 2 minutes.

_____ 6.35.1 Record the number of bubbles per minute and convert to SCFH:

_______ Bubbles per minute x 9.24 x 10^{-3} = _______ SCFH
OUTER OUTER

_____ 6.35.2 IF Bubble Frequency is greater than 60 bubbles per minute, THEN use a calibrated Rotameter with range between 0 and 8 SCFH calibrated at 45 psig to record flow below using Attachment 3.

n o w ______ SCFH
OUTER

_____ 6.36 Crack open 1-IA-602, Pers Hatch 0-Ring Test Line Test Conn Isol Valve.

_____ 6.37 Verify flow through 1-IA-602.

_____ 6.38 Close I-IA-602, Pers Hatch O-Ring Test Line Test Conn Isol Valve.

6.39 Close the following valves:

_____ 6.39.1 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv

_____ 6.39.2 1-IA-614, 1-IA-PCV-118 Inlet Isolation Valve

6.40 Do the following:

6.40.1 Slowly *open* one of the following valves to vent the test apparatus and the "O-ring. Mask N/A valve not used.

_____ • 1-IA-610, Pers Hatch O-Ring Test Line IA Drain Valve

_____ • 1-IA-602, Pers Hatch O-Ring Test Line Test Conn Isol Valve

_____ 6.40.2 Open 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve.

6.41 Ensure the following valves are closed:

_____ • 1-IA-610, P e s Hatch O-Ring Test Line IA Drain Valve

_____ • 1-IA-602, Pers Hatch 0-Ring Test Line Test Conn Isol Valve

_____ • 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve

6.42 Ensure the following valves are closed:

_____ 6.42.1 1-IA-611, O-RING TEST EQPT IA SPLY 1-IA-FI-100 OUT ISOL VV

_____ 6.42.2 1-IA-601, Pers Hatch Outer Door 0-Ring Test Ln IA Isol Vv

6.42.3 1-IA-609, Pers Hatch Inner Door 0-Ring Test Ln IA Isol Vv

_____ 6.42.4 1-IA-617, O-Ring Test Eqpt IA Sply 1-IA-PI-119 Isol Valve

_____ 6.43 IF leakage is greater than 2.86 SCFH, THEN submit WR on 1-CE-PH-1A

_____ 6.44 Turn off containment Bights using switches at door.

_____ _____ 6.45 Verify Teflon tape removed from all system test connections.

6.46 Ensure the following valves are closed:

- ______ ______ 1-IA-601, Pers Hatch Outer Door 0-Ring Test Ln IA Isol Vv
- ______ ______ 1-IA-602, Pers Hatch O-Ring Test Line Test Conn Isol Valve
- ______ ______ 1-IA-609, Pers Hatch Inner Door 0-Ring Test Ln IA Isol Vv
- ______ ______ 1-IA-610, Pers Hatch O-Ring Test Line IA Drain Valve
- ______ ______ 1-IA-611, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Out Isol Vv
- ______ ______ 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv
- ______ ______ 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve
- ______ ______ 1-IA-617, O-Ring Test Eqpt IA Sply I-LA-PI-119 Isol Valve
- ______ ______ 1-IA-614, 1-IA-PCV-118 Inlet Isolation Valve

7.0 FOLLOW-ON

7.1 Acceptance Criteria

______ The results of this test shall be considered acceptable if detectable leakage is $\leq$ 2.86 SCFH when each seal has been pressurized for at least two minutes.

7.2 Follow-On Tasks

______ IF the acceptance criteria cannot be satisfied, THEN refer to the ACTION statements of T.S. 3.6.1.3 (ITS 3.6.2).

7.3 Completion Notification

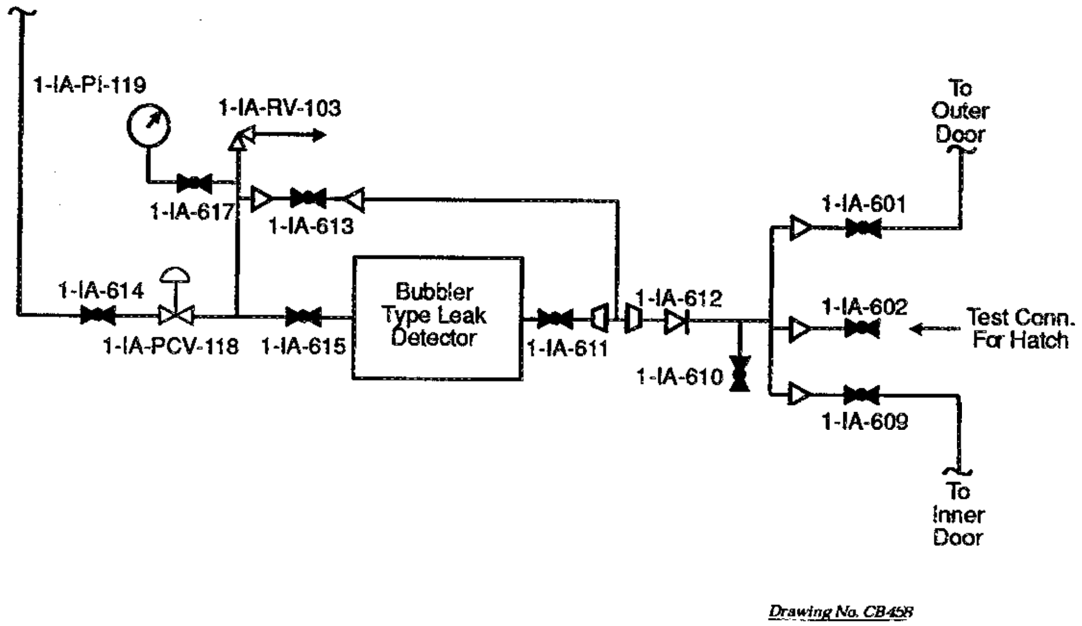
______ Notify the Shift Supervisor that this test is complete.

Completed: ______________________________ Date: ____________

ATTACHMENT I

(page I of 1)

PERSONNEL AIR LOCK SEAL LEAKAGE TEST SYSTEM



PERSONNEL AIR LOCK SEAL LEAKAGE TEST SYSTEM

ATTACHMENT 2

(Page 1 of 2)

LEAK TEST OF BYPASS YALVE ON BUBBLER LEAK DETECTOR

1.0 **Ensure** the **following** valva are **closed:**

_______ • 1-IA-601, Pers Hatch Outer **Door** O-Ring Test Ln IA **Isol** Vv

• 1-IA-602, Pers *Watch* **0-Ring** Test Line Test Conn Isol Valve

• 1-IA-609, Pers Hatch Inner Door **0-Ring** Test Ln IA Isol Vv

_______ • 1-IA-610, Pers Hatch Q-Ring Test Line IA **Drain** Valve

_______ • **1-IA-611, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Out Isol Vv**

_______ • 1-IA-615, O-Ring Test **Eqpt IA Sply 1-IA-FI-100** Inlet **Isol** Vv

_______ • 1-IA-613, O-Ring Test Eqpt **IA Sply 1-IA-FI-100 Bypass** Valve

_______ • **1-IA-617, O-Ring Test** ***Eqpt*** **IA Sply** 1-IA-PI-119 **Isol** Vdve

_______ • **1-IA-614, 1-IA-PCV-118** Inlet **Isolation** Vdve

_______ **2.0** Open I-IA-610, Pers Hatch **O-Ring** Test Line IA Drain Valve.

_______ **3.0** Connect **a law** range rotameter **(0.31** SCFH or less Full Scale calibrated **to** 14.7 **psia)** to 1-JA-610, Pers Hatch **O-Ring** Test Line IA Drain Valve.

NQC number: ____________

4.0 **open** the **following** valves:

_______ 4.1 1-IA-617, O-Ring Test **Eqpt** IA **Sply** 1-IA-PI-119 Isol Valve

4.2 **1-IA-614, 1-IA-PCV-118** Inlet Isolation Vdve

5.0 Adjust 1-IA-PCV-118 to 44.1 to 45.0 psig as read on 1-IA-PI-119.

ATTACHMENT 2

(Page 2 of 2)

LEAK TEST OF BYPASS VALVE ON BUBBLER LEAK DETECTOR

______ 6.0 Record rotameter reading:

__________ SCFH

______ 7.0 Close 1-IA-614, 1-IA-PCV-118 Inlet Isolation Valve.

______ 8.0 Remove the low range rotameter.

______ ______ 9.0 Verify all teflon tape removed from test connections.

______ 10.0 Open 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve, to vent the test rig.

11.0 Close the following valves:

______ 11.1 1-IA-610, Pers Hatch O-Ring Test Line IA Drain Valve

______ 11.2 1-IA-613, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve

______ 11.3 1-IA-617, O-Ring Test Eqpt IA Sply 1-IA-PI-119 Isol Valve

______ ENG 12.0 IF the value recorded in Step 6.8 is more than 0.0 SCFH, THEN contact Engineering for an evaluation.

ATTACHMENT 3
(Page 1 of 6)

SEAL LEAK TEST WITH ROTAMETER

Figure A
Regulator Rig

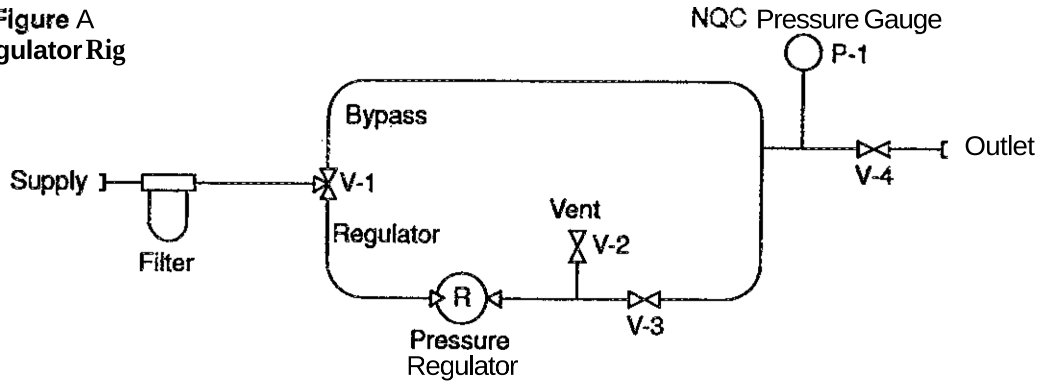
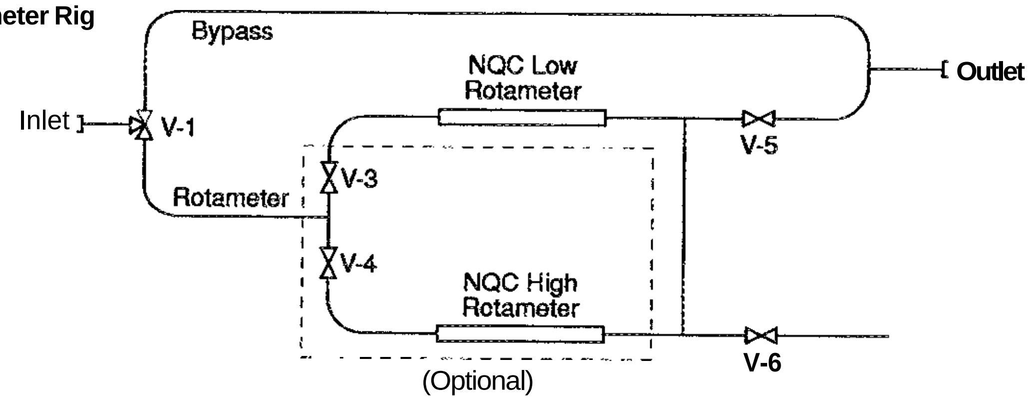


Figure B
Make-Up Rotameter Rig



Graphics No: WF73SD

ATTACHMENT 3
(Page 2 of 6)
SEAL LEAK TEST WITH ROTAMETER

* *

CAUTION: IF direct unregulated supply pressure air is applied to the seal THEN the seal could over pressurize and become damaged.

* *

NOTE: Additional attachments may be attached and used as needed.

1.0 Close the following valves:

_______ • 1-IA-601, Pers Hatch Outer Door 0-Ring Test Ln IA Isol Vv

_______ • 1-IA-602, Pers Hatch 0-Ring Test Line Test Conn Isol Valve

_______ • 1-IA-609, Pers Hatch Inner Door 0-Ring Test Ln IA Isol Vv

_______ • 1-IA-610, Pen Hatch 0-Ring Test Line IA Drain Valve

_______ • 1-IA-611, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Out Isol Vv

_______ • 1-IA-615, O-Ring Test Eqpt IA Sply 1-IA-FI-100 Inlet Isol Vv

_______ • 1-IA-613, Q-Ring Test Eqpt IA Sply 1-IA-FI-100 Bypass Valve

_______ • 1-IA-617, O-Ring Test Eqpt IA Sply 1-IA-PI-119 Isol Valve

_______ • 1-IA-614, 1-IA-PCV-118 Inlet Isolation Valve

ATTACHMENT 3

(Page 3 of 6)

SEAL LEAK TEST WITH ROTAMETER

_______ **2.0** Obtain Regulator Rig and Make-Up Rotameter Rig similar to the ones shown on Figure A and B. Record NQC numbers below:

High Rotameter: NQC# ____________ CALDATE ____________

Low Rotameter: NQC# ____________ CAL DATE ____________

Pressure gauge: NQC# ____________ CAL DATE ____________

3.0 Install Regulator Rig and Make-up Rotameter Rig as follows:

_______ **3.1** Close all valves on the Regulator Rig and place V-1 in OFF.

_______ **3.2** Connect Instrument Air OR Service Air Supply (less than 125 psig) to the supply at V-1.

_______ **3.3** Close all valves on the Make-Up Rotameter Rig and place V-1 in OFF.

_______ **3.4** Connect Regulator Rig outlet hose to inlet at V-1 of Make-up Rotameter Rig.

_______ **3.5** Connect Make-up Rotameter Rig outlet hose to 1-IA-610, Pers Hatch O-Ring Test Line IA Drain Valve.

4.0 Open one of the following valves for the seal to be tested. Mark valve not used N/A.

_______ • 1-IA-601, Pers Hatch Outer Door O-Ring Test Ln IA Isol Vv

_______ • 1-IA-609, Pers Hatch Inner Door O-Ring Test In IA Isol Vv

_______ **5.0** Open 1-IA-610, Pers Hatch O-Ring Test Line IA Drain Valve.

ATTACHMENT 3

(Page 4 of 6)

SEAL LEAK TEST WITH ROTAMETER

6.0 Place Regulator Rig and Make-Up Rotameter Rig in service as follows:

_______ 6.1 Open Air Supply valve.

_______ 6.2 Slowly place V-1 on the Regulator Rig in BYPASS and raise pressure on P-1 to 44.1 to 45.0 psig. Periodically place V-1 in OFF and check the static pressure on P-1.

_______ 6.3 Open V-4 on the Regulator Rig.

_______ 6.4 WHEN the static pressure on P-1 is near 44 psig, THEN place V-1 on the Regulator Rig in REGULATOR position and open V-3. Adjust the regulator to maintain 44.1 to 45.0 psig.

_______ 6.5 Place V-1 on the Make-up Rotameter Rig in BYPASS.

_______ 6.6 Allow system pressure to equalize with the outlet pressure from the Regulator Rig.

_______ 6.7 Open rotameter inlet valve:

- IF High Range Rotameter is installed, THEN open V-4
- IF High Range Rotameter is NOT installed, THEN open V-3.

_______ 6.8 Place V-1 in ROTAMETER position.

_______ 6.9 Open V-5.

_______ 6.10 Allow flow to stabilize and observe reading.

ATTACHMENT 3
(Page 5 of 6)
SEAL LEAK **TEST WITH ROTAMETER**

______ **6.11** IF flowing **through High** Range Rotameter AND reading is < 10% of full scale, THEN *open* V-3 AND close V-4 to **shift** to Low Rxnge Rotameter.

______ 7.0 Record reading for Inner Door in Step 5.15.2, or for Outer Door in Step 5.35.2.

8.0 Disconnect the Makeup **Rotameter** Rig **as** follows:

______ **8.1** Place V-1 on the MakeUp Rotameter Rig in OFF.

______ 8.2 Open V-6 to vent Make-Up Rotameter Rig and Containment Door seals.

8.3 Ensure the following valves are closed.

______ ______ • 1-IA-601, Pers Hatch Outer Door 0-Ring Test Ln IA Isol Vv

______ ______ • 1-IA-609, Pers Hatch Inner Door 0-Ring Test Ln IA Ksol Vv

______ ______ • 1-IA-610, Pers Hatch 0-Ring Test Line IA Drain Valve

______ **8.4** Disconnect Makeup Rotameter Rig outlet hose from 1-IA-610, Pers Hatch 0-Ring Test Line IA **Drain** Valve.

______ ______ **8.5** Verify all teflon **tape** is removed from test connections.

9.0 Test the MakeUp Rotameter Rig bypass valve as follows:

______ 9.1 **Connect** a low range rotameter (0.31 **SCFH** or less Full Scale calibrated to 14.4 psia) to Makeup Rotameter Rig outlet hose.

NQC number: ____________

ATTACHMENT 3

(Page 6 of 6)

SEAL LEAK TEST WITH ROTAMETER

9.2 Close following valves on the MakeUp Rotameter Rig:

- v-3
- v-4
- v-5
- V-6

9.3 Ensure pressure at P-B on the Regulator Rig is between 44.1 and 45.0 psig.

9.4 Place V-1 on Makeup Rotameter Rig in ROTAMETER

9.5 Record low range rotameter reading:

_______ SCFH

ENG 9.6 IF the value recorded in Step 9.5 is more than 0.0 SCFH, THEN contact Engineering for an evaluation.

10.0 Secure the Regulator and Makeup Rotameter Rig as follows:

10.1 Close Air Supply valve.

10.2 Open V-2 on the Regulator Rig to depressurize both Rigs.

10.3 Remove test rotameter.

10.4 Disconnect inlet hose and outlet hose from the Regulator Rig and the Make-up Rotameter Rig.

10.5 Verify all teflon tape is removed from test connections

G2.1.32 001

Unit I is at 100 % power when the Turbine 1 operator comes to the Generator Hydrogen panel to take logs. The current H2 pressure is 68 psig. Purity is 96%. The operator should obtain a copy of ________

A✓ 1-OP-43.3, " Hydrogen Makeup and Bleed and Feed to the Main Generator" and fil the generator to 70-75 psig to prevent Bearing Cooling Water from entering the generator

B. 1-SC-4.3 to verify **H2** pressure is satisfactory for current MWE and MVAR values tc ensure generator windings do not overheat

C. 1-OP-43.3, " Hydrogen Makeup and Bleed and Feed to the Main Generator" and fill the generator to 70-75 psig to prevent the generator from overheating

D. 1-OP-43.3, " Hydrogen Makeup and Bleed and Feed to the Main Generator" to perform bleed and feed of the main generator to prevent Bearing Cooling water from entering the generator

A. This is the correct answer. Precautions and limitations of 1-OP-43.3 states H2 pressure should be between 70-75 psig. This is to keep H2 pressure above Bearing Cooling Water pressure.

B. This answer is incorrect. 1-SC-4.3 is only used if bringing H2 pressure below 60 psig. Feed and bleed is not required. Examinee may choose this answer if they confuse the requirement of feed and bleed and normal **H2** pressure. Examinee may think feed and bleed is required based on purity.

C. **This** answer is incorrect. 70 to 75 psig is the correct H2 pressure but the reason for the pressure is wrong. The generator capability curve will allow operation below 70 psig. The values in this curve are what prevent overheating.

D. This answer is incorrect. H2 purity is not at a value that requires feed and bleed.

Conduct of Operations

Ability tu **explain and apply all system** limits and **precautions**

References:
Objective 453 form study guide on Main Generator and Exciter
1-OP-43.3, " Hydrogen Makeup and Bleed and Feed to the Main Generator."
Turbine Building Logs.
1-AR-T-A1
1-AR-T-A2

QUESTIONS REPORT
for **sroquestions**

Level(RO/SRO):	SRO	Tier:	3
Group:		Importance Rating:	3.4/3.8
Type(Bank/Mod/New):	**NEW**	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	**N**

5: Main Generator Hydrogen Subsystem

Topic 5.1: Hydrogen Subsystem
453

5.1a. Objective

Explain the following concepts associated with the Main Generator Hydrogen Subsystem.

- Why hydrogen **is** used **as** a cooling medium in the main generator
- Why makeup of hydrogen to the main generator is required periodically
- How the control room operator verifies that an explosive mixture does not exist in the main generator
- What determines the minimum and maximum hydrogen pressure allowed in the main generator
- Why carbon dioxide is used **as** an intermediate gas when purging the main generator
- Purpose of the liquid level detectors for the main generator

5.1b. Content

1. The Generator Hydrogen Gas Subsystem removes the heat produced in the generator stator windings, rotor windings, and stator core.
   - **1.1.** Cooling is accomplished by circuiating pressurized hydrogen gas in a closed path through the generator and a set *of* hydrogen coolers.
   - 1.2. Hydrogen is used **as the** cooling medium for the following reasons:
     - 1.2.1. Hydrogen has a low density, which results in less windage and ventilation losses and less windage noise.
     - 1.2.2. Hydrogen has high thermal conductivity and a high heat transfer coefficient. resulting in more heat transfer per unit volume than with air.
     - 1.2.3. Use of hydrogen in a closed recirculating system reduces maintenance expense, due to freedom from moisture and dirt.

2. During normal generator operation, the hydrogen gas inside the generator is maintained at approximately 75 psig pressure and 95 percent or greater purity.
   2.1. This gas volume is sealed inside the generator and recirculated continuously.
   2.2. Makeup hydrogen must be added periodically to the generator for two reasons:
      2.2.1. Hydrogen Leakage out of generator - requires hydrogen addition to maintain pressure; and
      2.2.2. Air leakage into generatof - requires hydrogen addition to maintain purity.

3. The hydrogen purity analyzer monitors the percent purity of generator hydrogen gas to verify that an explosive mixture does not exist in the generator.

4. **The** maximum hydrogen pressure allowed in the main generator is 75 psig based on the main generator physical characteristics, the minimum pressure is 70 psig and is based on the bearing cooling system pressure.

5. When mixed with air, hydrogen gas is explosive over a range of approximately **4 - 74** percent hydrogen.
   5.1. During normal generator operation it is not possible to form such an explosive mixture (due to greater than 95 percent purity); however, when maintenance is performed on the generator. such a condition can arise.
   5.2. Consequently, the hydrogen gas must be purged from the generator prior tu any maintenance.
   5.3. Since air and hydrogen should never be mixed, carbon dioxide (CO_2) is used as an intermediate gas when changing from hydrogen to air or air to hydrogen.

6. Three liquid level detectors are installed in the main generator to indicate the presence of any liquid in the generator due to leakage or condensation.

Topic 5.2: Hydrogen Flowpaths

HYDROGEN **PURITY**
HIGH OR LOW

LO ≤ **90%**
HI ≥ 100%

1.0 Probable Cause

1.1 Low purity
 1.1.1 C02 not fully purged
 1.1.2 Improper operation of the seal oil pressure regulating valves, 1-GM-PDCV-121 and 1-GM-PDCV-122
 1.1.3 Improper operation of the hydrogen side drain regulator
1.2 Hi purity
 1.2.1 Pointer on purity meter is stuck
 1.2.2 Purity meter blower has stopped

2.0 Operator Action

2.1 IF purging generator, **THEN** continue to purge until 100% H2

2.2 Check delta P between H2 Side Seal Oil pressure and Air Side Seal Oil pressure. IF > 2" of water, THEN pressure regulating valves must be adjusted.

2.3 Check operation of the H2 side drain regulator.

2.4 IF the purity meter or purity meter blower is NOT functioning properly, THEN notify Shift Supervisor and Instrument Department.

3.0 References

3.1 Unit 1 Loop Book, pg GM 001
3.2 W instrumentation book 20804 (Generator tech manual)
3.3 11715-FM-104A

4.0 Actuation

4.1 1-GM-H2A-110 (Analyzer Sensor)

HYDROGEN PRESSURE
HIGH or LOW

HI > 80 psi
LO < 65 psi.

1.0 Probable Cause

1.1 H2 pressure high:
- 1.1.1 H2 make-up in progress
- 1.1.2 Loss of BC to H2 cooler

1.2 W2 pressure low:
- 1.2.1 Leakage of H2
- 1.2.2 Venting in progress
- 1.2.3 Insufficient Seal Oil Pressure
- 1.2.4 Purge valve stuck open or leaking on inservice dryer tower
- 1.2.5 Hydrogen Dryer switching failure

2.0 Operator Action

2.1 Check H2 pressure in generator. IF pressure is decreasing rapidly, THEN do one of the following AND purge the generator with C02 using 1-OP-43.1:

a) IF Reactor Power ≥ 30%, THEN GO TO 1-E-0, Reactor Trip Or Safety Injection.

b) IF Reactor Power < 30%, THEN GO TO 1-AP-2.1, Turbine Trip Without Reactor Trip Required.

2.2 H2 Pressure High:
- 2.2.1 Secure H2 Make-up.
- 2.2.2 IF H2 temperature is high, THEN valve in cooling water.

2.3 K2 Pressure Low:
- 2.3.1 IF due to normal operational leakage, THEN make-up as per 1-OP-43.3.
- 2.3.2 IF performing bleed and feed, THEN ensure pressure is being maintained in accordance with the controlling procedure.
- 2.3.3 IF K2 temperature is low, THEN throttle cooling water closed.
- 2.3.4 Dispatch operator to check H2 dryer for the following:
  - a) Any alarm indicated on the H2 dryer skid control panel.
  - b) In-service tower pressure should equal main generator pressure.
  - c) Out of service tower status.
  - d) 1-GM-FI-602, 1B Hydrogen Dryer Hydrogen Leakby, should be approximately zero.
- 2.3.5 IF any problems are noted or suspected with the dryer, THEN do the following:
  - a) Turn off dryer with switch on the side of the dryer panel.
  - b) Close 1-GM-72 and 1-GM-73.
  - c) Close 1-GM-70 and 1-GM-71.
  - d) Submit Work Request.

3.0 References

3.1 Unit 1 Loop Book, pg GM 005

3.2 W instrumentation book 20804 (generator tech manual)
3.3 North Anna Setpoint Document
3.4 1-OP-43.1, Operation of the Generator Gas Systems
3.5 1-OP-43.3, Hydrogen Makeup to the Main Generator
3.6 DCP 88-03
3.7 DCP 95-282, Replacement of Unit 1 Generator Hydrogen Dryer

4.0 Actuation

4.1 1-GM-PS-112

VIRGINIA POWER **NORTH ANNA POWER STATION**	PROCEDURE NO: 1-OP-43.3	
	UNIT NO: 1	REVISION NO: 11
PROCEDURE TYPE: OPERATING	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE

HYDROGEN MAKEUP AND BLEED AND FEED TO THE MAIN GENERATOR

REVISION SUMMARY

- Incorporated OP 02-0417, which added new section 5.3, to simultaneously feed and bleed the generator while in service.
- Added Table of Contents

ELECTRONIC DISTRIBUTION — APPROVAL ON FILE

PROBLEMS ENCOUNTERED: ☐ Yes ☐ No **NOTE:** If yes, note problems in Remarks.

REMARKS:

(use back for additional space)

SHIFT SUPERVISOR:	DATE:

TABLE OF CONTENTS

1.0 PURPOSE

To provide instructions for adding H_2 to the Main Generator during Main Generator operations.

To provide instructions for bleeding and feeding the Main Generator during Main Generator operations.

2.0 REFERENCES

2.1 Source Documents

None

2.2 Technical Specifications

None

2.3 Technical References

2.3.1 Westinghouse Instruction Book, Generator-20804;1,088,600 KVA Hydrogen h e r Cooled Turbine, Unit 1

2.3.2 Flow Diagram 11715-FM-104A, H_2 and CO_2 Supply Lines

2.4 Commitment Documents

2.4.1 CTS Assignment 02-89-1200, Commitment 001, OE 31-20, Hydrogen Contamination of Plant Air Systems

2.4.2 CTS Assignment 02-89-3032, Commitment 012, SOER 82-09, Turbine Generator Exciter Explosion

3.0 INITIAL CONDITIONS

3.1 The Main Generator is in operation with a H_2 atmosphere.

3.2 H_2 main bank or reserve bank pressure is at least 300 psig.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Comply with the following guidelines when marking steps N/A:

- IF the conditional requirements of a step do not require the action to be performed, THEN mark the step N/A.
- IF any other step is marked N/A, THEN have the Shift Supervisor (or designee) appmve the N/A and justify the N/A on the Procedure Cover Sheet.

4.2 Ensure that the Seal Oil pressure is maintained 12 psig above gas pressure during normal system operation.

4.3 H_2 is explosive when mixed in air at concentrations between 4 and 74 percent. Became H_2 has been known to collect in the Exciter housing, the Electricians sample the atmosphere using 0-ECM-2301-01, Main Generator Hydrogen Gas System Leak Test and Repair, when Hydrogen leakage is known or has increased.

4.4 WHEN working on the H_2 Gas System, THEN always use spark-free wrenches.

4.5 Observe the NO SMOKING signs in the area of the H_2 System.

4.6 WHEN H_2 bank pressure drops to 200 psig, THEN isolate the bank in service and place the other bank in service.

4.7 H_2 leakage from the Generator is indicated by decreasing H_2 pressure in the Generator. Investigate AND correct the cause of the leakage before explosive mixtures can accumulate in the exciter enclosure and pockets.

4.8 To maintain Generator gas pressure greater than BC pressure, H_2 pressure should be maintained between 70 and 75 psig.

4.9 H_2 Purity Meter return flow is directed to the Main Generator atmospheric vent. 1-GM-1, Blower Discharge, and 1-GM-2, Blower Discharge, that are on top of cabinet, should remain closed at all times.

4.10 To prevent possible Hydrogen contamination of plant air systems, valves should be independently verified to be in the correct position. (Reference 2.4.1)

4.11 <u>WHEN</u> performing a Bleed and Feed, <u>THEN</u> 1-SC-4.3 shall be used to determine MWe and VAR limits if any of the following conditions occur

- Desired pressure in Step 5.2.4 is less than 60 psig.
- MVARs greater than 210 out at 100 percent power.
- MVARs greater than 160 in at 100 percent power.

Init Verif

5.0 INSTRUCTIONS

5.1 Performing Hydrogen Makeup to the Main Generator

______ 5.1.1 Verify Initial Conditions are satisfied.

______ 5.1.2 Review the Precautions and Limitations.

5.1.3 Record the following initial data:

______ a. The: ____________ Date: ____________

______ b. 1-GM-PI-112-2, Generator Gas Pressure (Control Board): ______ psig

______ c. 1-GM-PI-100, H_2 Main Bank Pressure: ______ psig

d. 1-GM-PI-101, H_2 Reserve Rank Pressure: ______ psig

* *

WARNING: Smoking is <u>NOT</u> allowed in the area of the H_2 System.

* *

5.1.4 Place one of the following H_2 Banks in service as follows. Mark the other bank N/A:

a. For the Main Bank, do the following (valves are located in upper red box by Hydrogen Banks):

______ • Ensure 1-GM-49, 1-GM-PCV-102A Inlet Isolation Valve, is open.

______ • Open 1-GM-50, 1-GM-PCV-102A Outlet Isolation Valve.

G2.1.7 001

Unit 1 is at 100% power. Pressurizer pressure transmitter, 1-RC-PT-1455 has failed low. All bistables for the associated channel have been placed in the trip condition. Subsequently the crew experiences a loss of Vital Bus I-I.

Based on the above conditions, the unit will **automatically** trip on ______

A✓ worsening condenser vacuum
B. pressurizer high pressure
C. over temperature delta T
5. loss of RCP CC flow

A. This is the correct answer. The air ejector discharge valves go shut on a loss of vital bus 1-1. This causes vacuum to erode reaching the vacuum trip setpoint in approximately 20 minutes.

B. This answer is incorrect. The pressure transmitter is powered from vital bus 1-1. It takes two redundant channels to fail to get a trip. Examinee may think they will have 2 channels tripped but 1455 comes from vital bus 1-1

C. This answer is incorrect. If 1455 was powered from another power source this would be correct. The loss of a power source **will** cause indication to fail low. Based on this the examinee may think this is the correct answer rather than high pressure.

D. This answer is incorrect. Examinee may choose this answer because CC is lost to the RCP's in this failure but this does not lead directly to an automatic trip.

Conduct of Operations
Ability **to evaluate plant performance** and **make operational judgments based on operating characteristics.reactor behavior, and instrument interpretation**

References: 1-MOP-55.72, " Pressurizer Pressure Protection Instrumentation."
1-MOP-26.60, " Vital Bus 1-1."
1-AP-3, " Loss of Vital Instrumentation immediate actions."

This is a new question.

Level(RO/SRO):	**SRO**	**Tier:**	**3**
Group:		**Importance Rating:**	3.7/4.4
Type(Bank/Mod/New):	**NEW**	Cog(Knowledge/Comp):	**COMPREHENSIVE**
Reference(Y/N):	**N**	**Last Exam(Y):**	**N**

2. Monitor natural circulation using Attachment 24 of 0-AP-10, Loss of Electrical Power.

b. IF the Unit is Mode 1 or 2, THEN do the following:

1. Verify RCP temperatures are below the high limits as follows:

- Motor bearing temperature is less than 195 degrees F.
- Pump radial bearing temperature is less than 225 degrees F.
- Stator winding temperature is less than 300 degrees F.

2. IF any of the RCP temperatures exceeds the high limit OR any of the RCP temperatures obviously will exceed the high limit before CC flow can be restored, THEN trip the Reactor and RCPs, and GO TO 1-E-0, Reactor Trip or Safety Injection.

NOTE: Changes in Auxiliary Steam usage should be coordinated with Unit 2 to prevent undesired Aux Steam pressure changes.

c. Do the following to compensate for the closure of 1-AS-FCV-100A and 1-AS-FCV-100B, Steam Supply to Air Ejectors:

1. Secure Main Condenser Air Ejectors using 1-OP-36.2, Main Condenser Air Ejector.

2. IF desired, THEN place the Main Condenser Hoggers in service in accordance with 1-OP-36.1, Main Condenser Hogging Ejectors.

NOTE: At power levels less than 30 percent, the Hoggers are NOT capable of maintaining vacuum greater than Turbine Trip setpoint.

3. Refer to 1-AP-14, Low Condenser Vacuum.

5.2 Placing Pressurizer Pressure Protection Channel I (P-1455) in Test

_____ 5.2.1 Verify Initial Conditions are satisfied.

5.2.2 Review Precautions and Limitations.

CAUTION

IF one of the coincident channel annunciators is LIT, THEN the following apply:

- Placing this channel in TEST could cause a unit trip or a Safety Injection.
- IF placing *this* channel in **TEST** would cause a unit trip or a Safety Injection, THEN comply **with** the following:
- The channel must NOT be placed in TEST.
- The unit must be placed in the mode required by Tech Spec 3.0.3.

5.2.3 Verify coincident channels are not tripped by verifying the following annunciators are NOT LIT:

_____ • Panel "M" H-6, PRZR LO PRESS CHNL II

• Panel "M" H-7, PRZR LO PRESS CHNL III

_____ • Panel "M" H-2, PRZR HI PRESS CHNL II

• Panel "M" H-3, PRZR HI PRESS CHNL III

_____ • Panel "M" G-6, PRZR **PRESS LO-LO SI** CHNL II

_____ • Panel "M" G-7, PRZR PRESS LO-LO SI CHNL III

- Panel "M" D-2, RC LOOP 1B OVRTEMP ΔT C-3 INTLK

- Panel "M" D-3, RC LOOP 1C OVRTEMP ΔT C-3 INTLK

- Panel "M" E-2. RC LOOP 1B OVRTEMP ΔT C-3 CHNL III

- Panel "M" E-3, RC LOOP 1C OVRTEMP ΔT C-3 CHNL III

5.2.4 **Ensure** LOOP ΔT/ΔT SET PT RECORDER SELECTOR 1/1 TR-412 **is** in LOOP 2 or **LOOP 3.**

5.2.5 Using Attachment 1, Process **Typical** Channel Test Card, go to the Instrument Rack Room and locate Channel I Protection Cabinet 1.

5.2.6 Unlock and open Channel I Protection Cabmet 1 and verify **that** annunciator Panel "P" G-5, PCC CAB I VIOLATED DOOR OPEN, is LIT.

NOTE: Attachment 2, **Process Typical 2-Bay** Cabinet, and Attachment 3, Process Typical 3-Bay Cabinet, will aid in identifying the correct card **and** Attachment **4,** Process Typical Channel Test Card, will aid in identifying the correct switch on the card.

NOTE: The Channel Test Status light on the top edge of the Channel Test Card comes on only when the **loop** is properly in TEST with master test switch in NORMAL. Attachment **4** will aid in identifying the Channel Test Status light.

5.2.7 Place the following Bistable (BS) trip switches in TEST and verify the associated annunciators are LIT and computer alarm prints out **(P-250)** or actuates (Phase 2 PCS):

a. C1-428, BS-1 **(1-RC-PTS-1455A)**

______ SV

- ______ Panel "B" F-4, PRZ **H PRESS RX TRIP** CH I-II-III
- ______ Panel "M" H-1, PRZR HI PRESS CHNL I
- ______ Computer Point P0480D, PRESSURIZER HI P 1 PART RE (P-250), OR **PRESSURIZER** HI PRESSURE CH I (Phase 2 PCS)

b. C1-428, BS-2 (1-RC-PTS-1455B)

______ ______ SV

- ______ Panel "L" **R-5,** PRZR **PRESS** > 2000 PSI CHNL I (lit above 2000
- ______ Computer Point P0492D, PRESUZER LO P 1 SI TR PART BLOCK (set above 2000 psig) (P-250), OR PRZR P CH I <2000#: PART P11 (Phase 2 PCS)

- Vital AC bus inverter
- Vital AC bus Sola transformers
- Vital bus transfer switch
- Vital 120-volt AC busses

4.1b. Content

Topic 4.2: Terminal Knowledge Objective 12019

4.2a. Objective

Given a set of plant conditions, evaluate Vital and Emergency Electrical Distribution System operations in light of the following issues.

- Effect of a failure, malfunction, or loss of a related system or component on this system
- Effect of a failure, malfunction, or loss of components in this system on related systems
- Expected plant or system response based on vital and emergency electrical distribution component interlocks or design features
- Impact on the technical specifications
- Response if limits or setpoints associated with this system or its components have been exceeded
- Proper operator response to the condition as stated

4.2b. Content

{Write the instructional content for the objective here...]

QUESTIONS REPORT
for sroquestions

G2.2.19 001

Unit 1 is at 100% power with nothing out of service. Which one of the following would be classified as an "urgent work order" in accordance with DNAP-2000, "Dominion Work Management Precess."

A?' Failure of iso-phase bus cooling
B. Failure of a steam generator feed pump
C. Failure of 1-MS-PT-1446, Turbine First Stage Pressure
D. Failure of 1H EDG

A. This is the correct answer. Iso-phase cooling is load threatening. This is classified as an urgent work request per DNAP 2000.

B. This answer is incorrect. This places the unit in a 72 hour action statement. This gives time to write a proper work package. It is not an urgent work request. Examinee may think entry into a action statement may require an urgent work request.

C. This answer is incorrect. A steam generator feed pump is only load threatening if there is not a spare. In this case there is a spare.

D. This answer is incorrect. 1H EDG is maintenance rule. Based on this the examinee may think it requires an urgent work request so it won't accumulate maintenance rule hours.

Equipment Control
Knowledge of **maintenance work** order requirements

References: DNAP-2000 Urgent Work Orders

Level(RO/SRO):	SKO	Tier:	3
Group:		**Importance Rating:**	2.1/3.1
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	Last Exam(Y):	N

Topic 1.31: Work Requests/Work Orders (DNAP-2000) 13113

1.31a. Objective

Explain how to determine if a corrective maintenance action should be classified as an urgent work order task (DNAP-2000).

1.31b. Content

1. Explain how to determine if a corrective maintenance action should be classified as an urgent work order task
   1.1. Urgent work is a deficiency, which if not immediately done, will endanger personnel, cause major damage to equipment, or result in significant loss of generation.
   1.2. Operations (Shift Manager) shall authorize entry of an urgent work request into Me work control system (or it may be hand-written).
   1.3. Operations shall determine, with the appropriate outage and planning personnel, which existing planned jobs should be stopped to support the urgent work.

Topic 1.32: Work Requests/Work Orders (DNAP-2000) 13130

1.32a. Objective

List the conditions that are acknowledged and approved by the shift manager upon approval of a work order (DNAP-2000).

1.32b. Content

1. Approval of a work order states that the Operations Shift Manager acknowledges and approves the following:
   1.1. The equipment is prepared for maintenance.
   1.2. The equipment is tagged, if designated by the work order.
   1.3. PMT can occur

Which of the following repairs would be classified as an "urgent work order' in accordance with DNAP-2000, "Dominion Work Management Process."

1. Equipment failure requiring entry into a 72-hour Technical Specification action statement
2. Equipment failure which could result in the loss of main generator bus duct cooling fan
3. beak of a carcinogenic herbicide in the turbine building basement
4. Component required to be in continuous operation has a severe oil leak

A. 2, 3, and 4 only

B. 1, 2, 3, and 4

C. 3 only

D. 1 and 2 only

Answer: A

QUESTIONS REPORT

for sroquestions

G2.2.9 001

Which of the following activities requires a Safety Review before being performed?

A. Instrument technician lifts a lead for the VCT HI/LO annunciator at the request of the Reactor Operator before calibrating the level channel.
B. Tagging out 1-CH-P-1A, A Charging Pump to repair a casing Beak
C. Installing jumpers in both source range cabinets in accordance with 1-AP-4.2, " Malfunction IR NI."
D. Performing a Temporary Procedure involving multiple departments.

A. This is the correct answer. Temporary Mods require a safety review. This lead was lifted without an approved procedure or work package. This makes it a temporary **mod.**

B. This answer is incorrect. Tag outs are listed as an exclusion in VPAP-1403. The examinee could choose this answer if they think removing safety related equipment could affect the design basis. Tec. specs. has already done an evaluation for this configuration and a separate safety evaluation is not required.

6. This answer is incorrect. This is a temporary mod but it is covered by a procedure. This excludes it from requiring a safety evaluation. A safety evaluation was done when the procedure was issued.

D. This answer is incorrect. Temporary procedures get a safety review before they are issued. It does require a major evolution brief. Examinee may get these two evolutions confused.

Equipment Control
Knowledge of the process for determining if the proposed change, test or experiment increases the probability of occurrence or consequences of an accident during the change, test, or experiment.

References: VPAP 3001 page 38, " Activities That require The Safety and Regulatory review Process."
VPAP-I 403, "Temporary Modifications"

This is a new question.

Level(RO/SRO):	SRO	Tier:	3
Croup:		**Importance Rating:**	2/3.3
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	Last Exam(Y):	N

6.1 Activities That Require The Safety and Regulatory Review Process

A Safety Review and Regulatory Review shall be prepared for:

- Temporary system installations (e.g., electrical or mechanical, jumpers (temporary modifications as defined in VPAP-1403, Temporary Modifications)), hoses, tubing, water chemistry, and other temporary arrangements **[Commitments 3.2.4, 3.2.6,** and **3.2.12]**
- Nonradioactive systems-kept in operation-that have detectable levels of contamination outside their design basis described in the UFSAR (North Anna) or that exceed a limit supported by an existing Regulatory Evaluation (or Safety Evaluation if its approval date is before June 25,2001). [Commitment 3.2.15]
- Any proposed design change, including a design basis limit for a fission product barrier. See also: 4.16, Design Basis Limits for Fission Product Barriers
- Modification or addition to, or removal from, the facility or procedures that adversely affect UFSAR-described design functions, methods used to perfom or control design functions, or evaluations that demonstrate that the intended design functions will be accomplished
- Alteration to tests or experiments described in the UFSAR
- Tests or experiments not described in the UFSAR where any SSC is utilized or controlled in a manner which is either outside the reference bounds of the design bases described in the UFSAR or inconsistent with the analyses or descriptions in the UFSAR
- Deviations that identify discrepancies in Technical Specifications [Commitment 3.2.5]
- Procedures that control activities involving proposed changes, tests, or experiments, as defined in this procedure

6.2 Activity Screening

If a Safety Review and Regulatory Review is required by 6.1, an Activity Screening Checklist is not required. All other applicable activities shall have an Activity Screening Checklist prepared in accordance with this subsection. The Activity Screening Checklist is a high level screen intended to identify if an activity may be implemented without prior regulatory approval.

6.2.1 Procedures that control activities involving changes, tests, or experiments, as defined in this procedure, shall require preparation and disposition of an Activity Screening Checklist, Attachment 2, prior to approval of those activities.

4.5.3 A valve, spool piece, wire, circuit card, or other component that disables or alters the normal system functions or electrical circuit functions.

4.5.4 A temporary connection such as hoses, tubing, or piping which joins systems together, bypasses a component within a system, or removes components within a system, thus altering the system's design or configuration.

4.5.5 A temporary mechanical device, such as a blank flange, to prevent fluid flow in a system

4.5.6 Temporary heating, ventilation, or air conditioning (RVAC) equipment installed in safety related areas or used to maintain the operability of safety related equipment. **[Commitment 3.2.3]**

4.5.7 Temporary changes to ladder logic executed by safety-related programmable logic controllers or control schemes implemented in digital microprocessor based control equipment. **[Commitment 3.2.11]**

4.5.8 **Exclusions**

The following are **not** Temporary Modifications:

- Tank and pipe drains connected to floor drains
- Temporary removal of electrical box covers
- Work performed on inoperable or out of service equipment that is controlled by an active Tag-Out in accordance with VPAP-1402, Control of Equipment, Tag-Outs, and Tags
- Station configuration changes included in **SNSQC** approved procedures. Refer to 6.3
- Hoses that provide air service through **normal** service connections
- Temporary IIVAC equipment installed in non-safety related areas
- NVAC equipment installed in accordance with C-HP-1061.321, Portable Ventilation Systems: Use and Control **[Commitnient 3.2.3]**
- Temporary lead shielding installed and controlled in accordance with VPAP-2105, Temporary Shielding **Program**
- Setpoint or curve changes made in accordance with VPAP-0303, Scaling/Setpoint Change and Curve Control **Program**

(4.5.8 Exclusions continued)

- Temporary power supplies installed and controlled in accordance with Design Change Packages or Engineering Work Requests
- Ladder logic changes to non-safety related programmable controllers.
- Noses connected to vent or drains
- Spool pieces or blind flanges which are part of system design and are installed and removed to place equipment in service or remove equipment from service (e.g., Steam Generator Wet Lay-up System)
- Temporary Repairs/Replacements to ASME Section XI Class 1, 2, or 3 component pressure boundaries made in accordance with VPAP-0307, Repair and Replacement for ASME Section XI Components
- Installation of electrical jumpers, lifting of electrical leads, or removal of electrical fuses which are within the test boundaries established by a SNSQC approved procedure. A test boundary establishes isolation, so actions performed within the boundary should not affect any instrumentation or systems outside the boundary. The jumper, lifted lead, or fuse shall be in the test boundary and shall not be used to establish the test boundary. When jumpers, lifted leads, or removed fuses are used to establish a test boundary, they shall be controlled as procedurally controlled temporary modifications
- Leak seal repairs performed in accordance with SNSOC approved procedures
- Transient extension cords, communication cords, and fluorescent lighting fixtures that nieet separation criteria by not: crossing a fire barrier hoard in a cable tray, routing between two trains of safety related (colored) ladder tray, conduit, or cable tray, or being attached to permanently installed cable tray, conduit, or floor raceway
- Non-safety related relief valves gagged in accordance with OPAP-0012, Valve Operation **[Commitment 3.2.13]**

QUESTIONS REPORT

for sroquestions

G2.3.6 001

Who is responsible for approving **all** radioactive release permits prior to the execution. of the release?

A. HP Technician
B. Shift Manages
C. Chemistry Technician
D. Supervisor RA&MC

A. This is the correct answer. HP Technician ***is*** responsible for both preparing and approving all release permits.

B. This answer is incorrect. Radioactive releases are T.S. and as such it would make sense that operations management would review release permits but they are not required to.

C. This answer is incorrect. Chemistry approves non-radioactive release permits for environmental concerns.

D. This answer is incorrect. Supervisor of RA&MC only gets involved if release limits are greater than percentage of Tech Spec allowed.

Radiation Control

Knowledge of the rcqinrements for reviewing and approving release permits

References: HP-3010.023 Abnormal Liquid Release.
HP-3010.020 Radioactive Liquid Waste Release Permits.
HP- 3010.030 Radioactive Gaseous Waste Release Permits.

This is a new question

Level(RO/SRO):	SRO	Tier:	3
Group:		Importance Rating:	2.1/3.1
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	KNOWLEDGE
Reference(Y/N):	N	Last Exam(Y):	N

VIRGINIA POWER **NORTH ANNA POWER STATION**	PROCEDURE NO: **HP-3010.020**	
	UNIT NO: **1 AND 2**	REVISION NO: **3**
PROCEDURE TYPE: **HEALTH PHYSICS**	EFFECTIVE DATE: **ON FILE**	EXPIRATION DATE: **N/A**

PROCEDURE TITLE.

RADIOACTIVE LIQUID WASTE RELEASE PERMITS

	ITS					

REVISION SUMMARY:

- **Incorporated** ITS Subtask 7214
  - **Added (ITS 5.5.4)** behind TS **6.8.4.e in Step 2.2.1.**

Writer: Nora Nicholson/Alison Campbell	**Reviewer:** J. Edmonds

ELECTRONIC DISTRIBUTION — APPROVAL ON FILE

TABLE OF CONTENTS

1.0 PURPOSE

1.1 This procedure provides instructions €or initiating and preparing radioactive liquid waste release permits for liquid effluent pathways requiring sampling and analysis in accordance with station Technical Specifications and VPAP-2103N, Offsite Dose Calculation Manual (NAPS).

2.0 REFERENCES

2.1 Source Documents

2.1.1 10 CFR 20, Standards for Protection Against Radiation

2.1.2 North Anna UFSAR, Chapter 12, Radiation Protection

2'2 Technical Specifications

2.2.1 6.8.4.e, Radioactive Effluent Controls Program (ITS 5.5.4)

2.3 Technical References

2.3.1 VPAP-1701, Records Management

2.3.2 VPAP-2101, Radiation Protection Plan

2.3.3 VPAP-2103N, Offsite Dose Calculation Manual (ODCM, North Anna)

2.3.4 HP-3010.021, Radioactive Liquid Waste Sampling and Analysis

2.3.5 IF-3010.022, Radioactive Liquid Waste Accountability arid Dose Calculations

2.4 Commitment Documents

2.4.1 NRC Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Eight-Water-cooled Nuclear Power Plants, Revision 1, June 1974

3.0 INITIAL CONDITIONS

Operations has requested or will request HP to initiate a radioactive liquid waste release permit in accordance with VPAP-2103N, Offsite Dose Calculation Manual.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The Supervisor Radiological Analysis and Material Control (RA&MC) is responsible for compliance with this procedure.

4.2 Instruction subsections and steps may be completed in the sequence and frequency appropriate to accomplish required tasks.

4.3 Liquid release authorization requirements are based on the percent Tech Spec concentration limit the release would represent if released at the source's maximum available GPM. If the percent Tech Spec is greater than the current percent Tech Spec allocation, then the liquid must be processed and resampled, or the release authorized by Supervisor RA&MC. The current percent Tech Spec allocation *is* indicated on a Liquid Waste Sampling and Analysis Requirements, completed in accordance with HP-3010.021, Radioactive Liquid Waste Sampling and Analysis.

4.4 A Release Permit is required for either batch or continuous liquid effluent releases containing radioactive materials. Releases which go through the Clarifier are continuous releases.

4.5 **VPAP-2103N,** Offsite Dose Calculation Manual, requires Operations to have a Release Permit from Radiological Protection prior to releasing radioactive liquid effluents.

4.6 For liquid effluents, Technical Specifications allow effluent concentrations 10 times the concentrations provided in 10 CFR *20,* Appendix B. The term Allowable Concentration in Water (ACW) is used in effluent control procedures to indicate values 10 times the Effluent Concentration in Water (ECW) provided in 10 CFR 20, Appendix B. The specific value for noble gas radionuclides in water provided in the ODCM is not subject to the 10 times increase.

4.1 Computer generated Release Permits **and** Records are acceptable.

5.0 SPECIAL TOOLS AND EQUIPMENT

- Computer, if used for release permit preparation

6.0 INSTRUCTIONS

6.1 Liquid Waste Batch Release Authorization

6.1.1 When requested by Operations to initiate a liquid release permit, then obtain from Operations the following information as required to complete Attachment 1, Liquid Waste Batch Release Permit and Record, Part 1.1, Operations Department Release Request:

- Release source,
- Maximum release rate available from source, GPM,
- Expected source release rate, GPM,
- Expected volume of waste to be released, gallons,
- Indication if Clarifier is bypassed by the release pathway (yes or no),
- Number of Circulating Water Pumps (CWP) running,
- Dilution water flow rate, GPM (maximum 218,000 per CWP running), and
- Operations By Printed Name and Signature, and Date.

6.1.2 Analyze, or have analyzed, the liquid waste sample obtained by responsible individuals. Ensure analysis is in accordance with HP-3010.021, Radioactive Liquid Waste Sampling and Analysis.

6.1.3 Prepare Liquid Waste Batch Release Permit and Record form, Part 3, Analyses and Release Data, as applicable.

NOTE: If the Clarifier is in service and at discretion of Supervisor RA&MC, credit may be taken for Clarifier DF for various radionuclides of interest. If Clarifier is bypassed or a DF will not be applied, then Clarifier DF = 1.

a. Record the foilowing:
- Sample μCi/mL (based on sample analyses performed)
- Clarifier DF (if applicable)

b. For each radionuclide i, calculate Nuclide Fraction of Allowable Concentration in Water (ACW). Record results in column Nuclide Fraction of ACW.

$$\text{Nuclide Fraction of ACWi} = \frac{\text{Sample } \mu Ci/mL_i}{\text{ACWi} \times \text{Clarifier } DF_i}$$

Where: $\mu Ci/mL_i$ = concentration of nuclide i in pathway

$ACWi = ECW_i \times AdjF$

ECW_i = Effluent Concentration value for Water (ECW) of nuclide i , from 10 CFR 20, Appendix B, Table 2, Column 2, or

= **VPAP-2103** (ODCM) value for noble gases

AdjF = 10 for nuclides with values provided in IO CFR 20, Appendix B, Table 2, Column 2, or

= 1 for noble gas radionuclides

DF_i = Clarifier DF for nuclide i, if applicable

c. Sum each recorded value for Nuclide Fraction **of** ACW. Record sum at bottom of column Nuclide Fraction of ACW.

d. Calculate maximum allowable release rate, GPM (Calc. Max. RR, GPM). Record results of calculations as indicated on form.

$$\text{Calc. Max. RR, GPM} = \frac{\text{Dilution Water GPM}}{\sum \text{Nuclide Fraction of ACWi}}$$

e. Calculate the maximum percent Tech Spec (Max. % Tech Spec) the release would represent if released at the source's maximum available GPM. Record results.

$$\text{Max. \% Tech Spec} = 100 \times \frac{\text{Maximum Source GPM}}{\text{Calc. Max. RR, GPM}}$$

f. Determine the current percent Tech Spec allocated for the pendiig liquid release source (allocation as determined in HP-3010.021, Radioactive Liquid Waste Sampling and Analysis). Record value.

g. IF Percent Tech Spec calculated in Substep 6.1.3.e. is greater than the current percent Tech Spec allocation, THEN determine requirements for liquid waste release.

1. Inform the Supervisor RA&MC. Determine if the liquid waste may or may not be released.

2. If the waste may be released, then determine a maximum release rate to be authorized and additional controls (if any).

3. If the waste may not be released, then have Supervisor RA&MC provide the reason and method to correct condition (e.g., reduce I-131 concentrations in waste) to be provided to Operations.

6.1.4 Complete Liquid Waste Batch Release Permit and Record form, Part 2, Release Authorization and Requirements, as applicable.

a. IF the waste may not be released, THEN terminate the release process for the batch.

1. Write, in Part 2, Release Authorization, Permit No., "No Release" or equivalent.

2. Advise Operations of the situation as soon as feasible.

3. GO TO Section 7.0.

NOTE: Maximum Release Rate Authorized rate must be less than or equal to the Maximum Source GPM indicated by Operations and the maximum release rate allowed.

b. Determine a value for the Maximum Release Rate Authorized.

1. Calculate the maximum release rate allowed (RRA), GPM, based on Calc. Max. RR, GPM calculated in **Substep** 6.1.3.d, and pathway allocated % Tech Spec.

$$\text{RRA, GPM} = \text{Calc. Max. RR, GPM} \times \frac{\text{Allocated \% Tech Spec}}{100}$$

2. Select a release rate less than or equal to both of the following:
   - Maximum Source GPM indicated by Operations, and
   - Calculated maximum release rate allowed (RRA).

c. Indicate (√) the Release Authorized (source authorized). Verify the source authorized, is the same as requested by Operations in Part 1.1.

d. Record the Maximum Release. Rate Authorized. GPM.

e. Calculate the Percent Tech Spec the Maximum Release Rate Authorized (Auth. Max RR, GPM) represents.

$$\text{Authorized Release Rate \% Tech Spec} = 100 \times \frac{\text{Auth. Max RR, GPM}}{\textbf{Calc. Max. RR, GPM}}$$

f. Record the Authorized Release Rate % Tech Spec (round off to two digits).

g. If applicable, record any conditions or controls pertaining to the release.

h. Record next available release Permit No.

i. Record HP Authorization **By** Printed Name and Signature, and Date,

6.1.5 Forward Liquid Waste Batch Release Permit and Record form to Operations.

NOTE: Subsection 6.2 is applicable when the Liquid Waste Batch Release Permit and Record fomi is returned from Operations.

6.2 Liquid Waste Batch Release Permit and Record Close Out

6.2.1 Review the Liquid Waste Batch Release Permit and Record form. Verify recorded release rate or rates did not exceed authorized release rate. Verify calculations of waste volume released completed in **Part** 1.2 by Operations.

6.2.2 If the release meets one or more of the following conditions:

- Release did not go through the Clarifier (e.g., Turbine Building Sump), or
- Clarifier was bypassed,

Then, complete Liquid Waste Batch Release Pennit and Record form, Part 3, Analyses and Release Data.

a. Record mL Released (volume released, from Part 1.2).

b. Calculate µCi released. Record results.

$$\mu\text{Ci Released} = \text{Sample } \mu\text{Ci/mL} \times \text{mL Released}$$

c. Record Release Data By Printed Name and Initials, and Date.

6.2.3 GO TO Section 7.0.

6.3 Liquid Waste Continuous Release Authorization

NOTE. A Liquid Waste Continuous Release Permit is required at all times. A release permit will be specific based on Clarifier status, Clarifier in service or Clarifier bypassed. A release permit is valid only for the Clarifier status indicated. If Operations requests two release permits and upon Supervisor RA&MC approval, having two active release permits is acceptable, in which case Operations is responsible for ensuring compliance with the applicable permit at any given time based on Clarifier status.

6.3.1 When requested by Operations to initiate a continuous liquid rclease permit, then obtain from Operations the following information as required to complete Attachment 2, Liquid Waste Continuous Release Permit, Part 1.1, Operations Department Requirements:

- Clarifier Status Release Permit to Cover (in service or bypassed, one only),
- Maximum Anticipated SG Blowdown Rate (high capacity and low capacity as applicable), GPM each SG,
- Minimum Expected Dilution Water GPM, and
- Operations By Printed Name and Signature, and Date.

NOTE: If considered valid, analysis data may be obtained from previous release permits. If required, samples should be obtained from pathways and analyzed. Containment Mat Sump (CMS), SW Reservoir, and Storm Drain activity may be estimated based on scaling down typical liquid waste sample activity.

6.3.2 Obtain effluent isotopic analysis data applicable to continuous releases to be authorized.

a. Obtain analysis data for the following potential continuous release pathways:
   - SG (from each of the six SGs)
   - BRTT, **LLWDT,** and CDT (typical expected activity)
   - CMS
   - Service Water Reservoir Blowdown (SWRBD)
   - Component Cooling Water (CCW)
   - Storm Drains (StmDrn)
   - Unit 1 and Unit 2 High Capacity Blowdown System

b. If a continuous release pathway is not prerecorded on the form, then record a description or name of the pathway, obtain analysis data for the pathway.

6.3.3 Prepare Liquid Waste Continuous Release Permit form, Part 3, Pathway Release Rate (RR) and % **Tech** Spec Calculations, as applicable.

NOTE: If the Clarifier is in service **and** at discretion of Supervisor RA&MC, credit may be taken for Clarifier DF for various radionuclides of interest. If Clarifier is bypassed or a DF will not be applied, then Clarifier DF = 1.

a. For each pathway, calculate Pathway Fraction of Allowable Concentration in Water (ACW). Use applicable Clarifier DFs. If calculated Fraction of ACW is less than 1.0, then use a value of 1. Record results on form.

$$\text{Pathway Fraction of ACW} = \sum \frac{\text{Source } \mu Ci/mL_i}{ACW_i \times \text{Clarifier } DF_i}$$

Where: $\mu Ci/mL_i$ = concentration of nuclide i in pathway
ACW_i = $ECW_i \times AdjF$
ECW_i = Effluent Concentration value for Water (ECW) of nuclide i, from 10 CFR 20, Appendix B, Table 2, Column 2, or
= VPAP-2103 (ODCM) value for noble gases
AdjF = 10 for nuclides with values provided in 10CFR 20, Appendix B, Table 2, Column 2, or
= 1 for noble gas radionuclides
DF_i = Clarifier DF for nuclide i, if applicable

b. For each pathway, calculate maximum allowable release rate, GPM (Calc. Max. RR, GPM). Record results on form.

$$\textbf{Calc. Max. RR, GPM} = \frac{\text{Dilution Water GPM}}{\text{Pathway Fraction of ACW}}$$

c. IF both the following conditions exist:
   - Calc. Max. RR, GPM for each SG is at least 10 times anticipated SG BD rate, and
   - Calc. Max. RR, GPM for other pathways is greater than 1000 GPM,

   THEN GO TO Substep 6.3.3.d, IF NOT, THEN determine requirements for the liquid waste (one or more pathways meet the criteria for having Supervisor RA&MC determine release requirements).

   1. Inform the Supervisor RA&MC. Determine which liquid waste pathway may or may not be released or requires treatment.

   2. If the waste may be released, then determine a maximum release rate to be authorized and additional controls (if any).

   3. If the waste may not be released, then determine the reason and suggested corrective actions to be provided to Operations.

NOTE: Maximum authorized release rate must be less than or equal to the Maximum Source GPM indicated by Operations.

d. Determine a release rate to be authorized.

e. Record the authorized maximum release rate (Auth. Max. RR, GPM).

f. Calculate the Percent Tech Spec (% Tech Spec) the authorized maximum release rate represents. Record % Tech Spec (round off to two digits).

$$\% \text{ Tech Spec} = 100 \times \frac{\text{Auth. Max RR, GPM}}{\text{Calc. Max. RR, GPM}}$$

g. Sum % Tech Spec values for SGs. Record total.

h. If clarifier is in service, sum % Tech Spec values for BRTT, LLWDT, CDT, CMS and SWRBD. Record total.

i. If clarifier is bypassed, sum % Tech Spec values for CMS and SWRRD. Record total.

j. IF permit is being prepared for S/G high capacity blowdown, THEN sum % Technical Specification (T.S.) values for each unit separately.

6.3.4 Complete Liquid Waste Continuous Release Permit form, Part 2, Release Authorization and Requirements, as applicable.

a. If a liquid waste pathway may not be released, then terminate the release process for the applicable continuous release pathway.

1. Record "No Release" or equivalent over the space for the applicable pathway release rate.

2. Record, in other Controls or Conditions, a comment that the applicable pathway may not be released.

3. Advise Operations of the situation as soon as feasible.

b. Indicate (√) the release pathways to be authorized.

c. Record the maximum release rates authorized, GPM.

d. If the Clarifier is in service indicate required BRTT, LLWDT and CDT sampling and analysis.

1. If a calculated maximum allowable release rate was less than 1,000 GPM, then sample and analyze prior to release.

2. If a calculated maximum allowable release rate was 1,000 GPM or more, then indicate sampling and analysis as determined by Supervisor RA&MC.

e. Record the Minimum Required Dilution Water Flow Rate, GPM (based on the dilution water flow rate used to calculate % Tech Spec).

f. Record or indicate the total % of Tech Spec limits which the authorized release rates represent.

g. If applicable, record any conditions or controls pertaining to the release.

h. Record next available release Permit No.

i. Record HP Authorization By Printed Name and Signature, and Date..

j. Make a copy of form. File copy in Count Room (for use by Count Room personnel).

6.3.5 Forward original Liquid Waste Continuous Release Permit form to Operations.

6.3.6 At least once per month, review Count Room copies of Continuous Release Permits. Verify permits are current and valid.

NOTE: Subsection 6.4 is applicable when the Liquid Waste Continuous Release Permit form is returned from Operations.

6.4 Liquid Waste Continuous **Release Permit** Close **Out**

6.4.1 Review the Liquid Waste Continuous Release Pennit form. Verify form is properly completed. If applicable, verify value for volume CCW released to Service Water is provided.

6.4.2 Obtain and discard Count Room copy of Continuous Release Pennit.

6.4.3 GO TO Section 7.0.

6.5 **Liquid Waste** Continuous **Release Rate** Record

NOTE. A continuous release rate record is used to document release rates of continuous releases are **within** Tech Specs limits. Accountability of activity released *is* addressed in procedure HP-3010.022.

6.5.1 Have a separate Liquid Waste Continuous Release Rate Record (Attachment 3) in use for each applicable continuous liquid release pathway.

a. Have one form for each Liquid Waste Continuous Release Permit and Record form currently approved for use (one or both for Clarifier in service or Clarifier bypassed).

b. If CCW is leaking radioactivity to SW, then have a Liquid Waste Continious Release Rate Recod Initiated for a CCW Leak to SW.

c. If a pathway other than previously identified exists, then have a record for the applicable pathway.

6.5.2 Complete each Liquid Waste Continuous Release Rate Record form.

a. Record the Period Covered by Form (release period to and from dates) and Applicable Calendar Quarter.

b. Indicate the Release Status and Minimum Analysis Requirements. Check (√) applicable boxes or record data as required. If required, have Supervisor RA&MC provide guidance.

c. Record the following sample analyses results.

- Date and Time sample obtained
- $\Sigma \frac{\mu Ci/mL}{ACW}$ (fraction of ACW)
- Waste Gallons (gallons the sample represents)
- Dilution Gallons (gallons of dilution water available during the period of time represented by the sample)
- % Tech Spec

$$\% \text{ Tech Spec} = 100 \times \frac{\frac{\mu Ci/mL}{ACW} \times \text{Waste Gallons}}{\text{waste Gallons} + \text{Dilution Gallons}}$$

d. When either of the following conditions exists:

- End of applicable calendar quarter, or
- All lines are used,

Then initiate new Liquid Waste Continuous Release Rate Record forms.

6.5.3 IF a calculated % Tech Spec exceeds the percent allocated (allocation as determined in HP-3010.021, Radioactive Liquid Waste Sampling and Analysis), THEN notify Supervisor RA&MC.

7.0 FOLLOW-ON

7.1 Follow-On Tasks

7.1.1 **Forward** the following completed forms to the Supervisor RA&MC for review.

- Liquid Waste Batch Release Permit and Record
- Liquid Waste Continuous Release Permit
- Liquid Waste Continuous Release Rate Record

7.1.2 Place the reviewed foms in the designated location pending completion of release accountability records, in accordance with HP-3010.022, Radioactive Liquid Waste Accountability and Dose Calculations.

7.2 Records Disposition

Forward the following records to Records Management in accordance with VPAP-1701, Records Management.

- Liquid Waste Batch Release Permit and Record
- Liquid Waste Continuous Release Permit
- Liquid Waste Continuous Release Rate Record

ATTACHMENT 1
(Page 1 of 1)

LIQUID WASTE BATCH RELEASE PERMIT AND RECORD

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Prior to release, complete Part 1.1, forward Release Permit to Health Physics for release authorization.

Release Source ☐ BRTT A ☐ LLLW A ☐ CDT A ☐ U1 T. Bl. Sump ☐ T. Bl. Cmn. Sump ☐ Drain SG ___ ☐ 2-BD-TK-2
☐ BRTT B ☐ LLLW B ☐ CDT B ☐ U2 T. Bl. Sump ☐ Drain Hotwell ______ ☐ ______

Maximum Source GPM: ________ Clarifier bypassed? __ Yes __ No Dilution Water Flow Rate: ________ GPM

Expected Source GPM: ________ Expected Volume to Release: ________ gallons. No. Circ. Water Pumps ______

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS DEPARTMENT RELEASE DATA

During and after release, complete Part 1.2, return to Health Physics for completion of release records.

Release Status	Date and Time	Tank Level	Release Rate GPM	Ops By (Initials)	Release Status	Date and Time	Tank Level	Release Rate GPM	Ops By (Initials)
Start					Start				
Stop					Stop				

Vol. Released = Vol. at start, gal ________ - Vol. at stop, gal ________ = ________ gallons. Gallons x 3785 = ________ mL.

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS
(as indicated and marked). A valid LW-111 alarm voids this permit.

PERMIT No.

Release Author. ☐ BRWA ☐ LLLWA ☐ CDT A ☐ U1 T. Bl. Sump ☐ T. Bl. Cmn. Sump ☐ Drain SG ___ ☐ 2-BD-TK-2
☐ BRTT B ☐ LLLW B ☐ CDT B ☐ U2 T. Bl. Sump ☐ Drain Hotwell ______ ☐ ______

Maximum Release Rate Authorized: ________ GPM. The estimated gross activity to be released is ______ Ci.

Authorized Release Rate is ______ % of Tech Spec limit. Required Controls or Conditions ______________________

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - ANALYSES AND RELEASE DATA	Release Data By (Printed Name)	(Initials)	Date

Nuclide	Sample µCi/mL	ACW µCi/mL	Clarifier DF**	Nuclide Fraction of ACW	mL Released	µCi Released	Remarks
H-3		1E-2					
Mn-54		3E-4					
Fe-59		1E-4					
Co-58		2E-4					
Co-60		3E-5					
Nb-95		3E-4					
Sb-124		7E-5					
Sb-125		3E-4					
Ag-110m		6E-5					
I-131		1E-5					
Cs-134		9E-6					
Cs-137		1E-5					
Xe-133		2E-4*					
			Σ Nuclide Fraction of ACW				

* ODCM value. ** Only if applicable.

$$\text{Nuclide Fraction of ACW} = \frac{\text{Sample } \mu\text{Ci/mL}}{\text{ACW} \times \text{Clarifier DF}}$$

$$\text{Max. \% Tech Spec} = \frac{100 \times \text{Max Source GPM}}{\text{Calc. Max. RR, GPM}} = \_\_\_\_\_\_$$

$$\text{Calc. Max. RR, GPM} = \frac{\text{Dilution water GPM}}{\Sigma \text{ Nuclide Fraction of ACW}} = \_\_\_\_\_\_ \text{GPM}$$

% Tech Spec Allocated for Source = ______________

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 2
(page 1 of 2)
LIQUID WASTE CONTINUOUS RELEASE PERMIT

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Complete Release Permit Part 1.1. Forward to Health Physics for authorization.
Clarifier Status Release Permit to Cover (one only): ☐ CLARIFIER IN SERVICE ☐ CLARIFIER BYPASSED
Maximum Anticipated SG Blowdown Rate, GPM: 1A____, 1B____, 1C____, 2A____, 2B____, 2C____
Minimum Expected Dilution Water GPM: 359,700 ☐ Other ______________

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS ARTMENT RELEASE PERMIT TERMINATION

Wh n Release Permit expires (see expiration date below) complete Part 1.2 and return permit to Health Physics.
IfC W leaking to Service W t d t nl l d ol e f CCW l as d to SW: ________ gallons.
Remarks ______________________________

Operations By (Printed Name)	operations By (Signature)	Date

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS
(as indicated and marked) A valid LW-111 alarm voids this permit.

PERMIT No.

This Release **Permit Expires (Date** and Time) ______________

☐ Steam **Generator Blowdown**: Maximum authorized blowdown release rate GPM:
1A________, 1B________, 1C________, 2A________, 2B________, 2C________

☐ **Sampling** and **Analysis Requirements (as** checked):
BRTT: ☐ P* ☐ W* ☐ _______, LLWDT: ☐ P* ☐ W* ☐ _______, CDT: ☐ P* ☐ W* ☐ _______
* P = sampling and analysis required prior to release. W = weekly sampling and analysis required.

☐ **Containment** Mat **Sumps** (when Clarifier bypassed): Maximum Mat Sump GPM ____________
☐ SW **Reservoir** Blowdown (when Clarifier bypassed): Maximum Reservoir Blowdown GPM ____________
☐ **CCW** to **Service Water Leakage:** Maximum authorized CCW Hx to Service Water Leak Rate GPM ____________
☐

☐ **Dilution** Water and Other **Requirements** Minimum Required Dilution Water Flow Rate: __________ GPM.
Authorized release rates are (as checked): ☐ ≤ 5% ☐ < 10% ☐ ________% Tech Spec
Other controls or conditions: ______________________________

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - PATHWAY RELEASE RATE (RR) and % TECH SPEC CALCULATIONS

Description	☐ SG 1A	☐ SG 1B	☐ SG 1C	☐ SG 2A	☐ SG 2B	☐ SG 2C	☐	Totals
Fraction of ACW								N/A
Calc. Max. RR, GPM*								N/A
Auth. Max. RR, GPM								N/A
% Tech. Spec.**								

Description	☐ BRTT	☐ LLWDT	☐ CDT	☐ CMS	☐ SWRBD	☐ CCW	☐ StmDrn	Totals
Fraction of ACW								N/A
Calc. Max. RR, GPM*								N/A
Auth. Max. RR, GPM								N/A
% Tech. Spec.**								

* Calc. Max. RR, GPM = $\frac{\text{Dilution Water GPM}}{\text{Fraction of ACW}}$ ** % Tech Spec = 100 x $\frac{\text{Auth. Max. RW GPM}}{\text{Calc. Max. RR GPM}}$

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 2
(Page 2 of 2)
LIQUID WASTE CONTINUOUS RELEASE PERMIT

PART 4 " OPERATIONS **DEPARTMENT** RELEASE **REQUEST**

Complete Release Permit Part 1 - I Forward to Health Physics for authorization.

Steam Generator High Capacity Blowdown Release Form ☐ CLARIFIER BYPASSED

Maximum Anticipated SG,High Capacity, Blowdown Rate, GPM: 1A_____, 1B_____, 1C_____,
2A_____, 2B_____, 2C_____

Minimum Expected Dilution Water GPM: 359,700 ☐ **Other** _______________

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS | **PERMIT No.**

This Release **Permit Expires** (Date and **Time**)

☐ Steam **Generator** Blowdown: Maximum authorized blowdown release rate GPM:
1A________, 1B________, 1C________, 2A________, 2B________, 2C________

Controls and **Conditions:** A valid RM-SS-125 or RM-SS-225 alarm voids this permit. Notify HP Count Room of any change in blowdown flaw rate.

____
____
____
____

Dilution Water and **Other Requirements** Minimum Required Dilution Water Flow Rate: ____________ GPM
Authorized release rat are (as checked): ☐ < 5% ☐ < 10% ☐ _________% Tech Spec

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - PATHWAY RELEASE RATE (RR) and % TECH SPEC CALCULATIONS

Description	☐ SG 1A	☐ SG 1B	☐ SG 1C	Totals	☐ SG 2A	☐ SG 2B	☐ SG 2C	Totals
Fraction of ACW				N/A				N/A
Calc. Max. RR, GPM*				N/A				N/A
Auth. Max. RR, GPM				N/A				N/A
% Tech. Spec.**								

Comments____
____
____

* Calc. Max. RR, GPM = $\frac{\text{Diltion Water GPM}}{\text{Fraction of ACW}}$ ** % Tech Spec = 100 x $\frac{\text{Auth. Max. RR GPM}}{\text{Calc. Max. RR GPM}}$

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 3
(Page 1 of 1)
LIQUID WASTE CONTINUOUS RELEASE PERMIT RECORD

Period Covered by this **Form** (to end from dates)	Applicable Calendar **Quarter** ☐ 1st ☐ 2nd ☐ 3rd ☐ 4th 19 ____

Release Status and Minimum Analysis Requirements

☐ Clarifier in service:	Analyze Clarifier:	☐ N/A	☐ Once/8 hrs	☐ Once/12 hrs	☐ Once/day	☐ Once/week
☐ Clarifi[illegible]	A[illegible]aly[illegible] SG BD	☐ N/A	☐ O[illegible] /8 [illegible]	☐ [illegible]nc[illegible]/12 hrs	☐ O[illegible] /d[illegible]y	☐ O[illegible]
	Analyze CMSs:	☐ [illegible]A	☐ O[illegible] /8 [illegible]	☐ [illegible]nc[illegible]/12 hrs	☐ Once/day	☐ Once/week
	A[illegible] SWR B	☐ [illegible]A	☐ Once/8 hrs	☐ Once/12 *his*	☐ O[illegible] /[illegible]y	☐ Once/week
O CCW Leak to SW:	A[illegible] CCW	☐ N/A	☐ Once/8 hrs	☐ Once/12 hrs	☐ O[illegible] /d[illegible]y	☐ O[illegible]k
	Analyze SW:	☐ N/A	☐ Once/8 hrs	☐ Once/12 hrs	☐ Once/day	☐ Once/week
☐ ____________	Analyze ________	☐ ____________				

☐ Clarifier Effluent Sample
☐ Steam Generator Blowdown (all SGs being released)

☐ CCW Sample
☐ ____________

Date and Time	$\Sigma \frac{\mu Ci/mL}{ACW}$	Waste Gallons	Dilution Gallons	% Tech Spec	Date and Time	$\Sigma \frac{\mu Ci/mL}{ACW}$	Waste Gallons	Dilution Gallons	% Tech Spec

* % T[illegible]ch [illegible] = 100 x $\frac{\Sigma (\mu Ci/mL / ACW) \times \text{Waste Gallons}}{\text{Waste Gallons} + \text{Dilution Gallons}}$

Data to Release Records By (Printed Name)	(Initials)	Date	Supv. RA&MC (Signature)	Date

Dominion NORTH ANNA POWER STATION	PROCEDURE NO: HP-3010.030	
	UNIT NO: 1 AND 2	REVISION NO: 8
PROCEDURE TYPE: HEALTH PHYSICS	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE

RADIOACTIVE GASEOUS WASTE RELEASE PERMITS

REVISION SUMMARY:

- Updated Corp title on cover sheet and headers
- Step 6.7.4.b, added BRT Vent release flow
- Step Step 6.8.1, Added methodology for BRT Vent release volume
- Step 2.2.1, deleted reference to ITS
- Installed Note prior to Step 6.8.2.b, to instruct on use of BRT Vent isotopic concentrations
- Administrative changes as required (no change bars)

Writer: H. Sinclare	Reviewer: J. Breeden

ELECTRONIC DISTRIBUTION — APPROVAL ON PILE

TABLE OF CONTENTS

1.0 PURPOSE

1.1 This procedure provides instructions for initiating and preparing radioactive gaseous waste release permits for gaseous effluent pathways *requiring* sampling and analysis in accordance with station Technical Specifications and **VPAP-2103N,** Offsite Dose Calculation Manual.

2.0 REFERENCES

2.1 **Source** Documents

2.1.1 IO CFR 20, Standards €or Protection Against Radiation

2.1.2 North Anna **UFSAR,** Chapter 12, Radiation Protection

2.2 Technical Specifications

2.2.1 6.8.4.e, Radioactive Effluent Controls **Program**

2.3 Technical References

2.3.1 VPAP-1701, Records Management

2.3.2 VPAP-2101, Radiation Protection Program

2.3.3 **VPAP-2103N,** Offsite Dose Calculation Manual (ODCM, North Anna)

2.3.4 HP-3010.031, Radioactive Gaseous Waste Sampling and Analysis

2.3.5 HP-3010.032, Radioactive Gaseous Waste Accountability and Dose Calculations

2.4 Commitment **Documents**

2.4.1 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

3.0 INITIAL CONDITIONS

Operations has requested or will request HP to initiate a radioactive gaseous waste release permit in accordance with **VPAP-2103N,** Offsite Dose Calculation Manual.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The Supervisor Radiological Analysis and Material Control (RA&MC) is responsible for compliance with this procedure.

4.2 Instruction subsections and steps may be completed in the sequence and frequency appropriate to accomplish required tasks.

4.3 Gaseous release authorization requirements are based on percent of Tech Spec dose rate limits the release would represent if released at the source's maximum available CFM. If the percent Tech Spec is greater than the current percent Tech Spec allocation, then the pathway must be processed and resampled, or the release authorized by Supervisor RA&MC. The current percent Tech Spec allocation is indicated on a Sampling and Analysis Requirements for Gaseous Waste form, completed in accordance with HP-3010.031, Radioactive Gaseous Waste Sampling and Analysis.

4.4 Release permits are required prior to release of Waste Gas Decay Tanks, Containment Purge or Hogging, and other miscellaneous releases of radioactive materials in gaseous effluents as indicated in this procedure.

4.5 VPAP-2103N, Offsite Dose Calculation Manual, requires Operations to have release permits from Radiological Protection prior to releasing radioactive gaseous effluents.

4.6 Computer generated Release Permits are acceptable.

4.7 Releases via VVA or VVB may result in a release through the other Ventilation stack if damper leakage exists. Leakage shall be evaluated using other radiation monitoring indicators and appropriate sampling shall be initiated.

5.0 SPECIAL TOOLS AND EQUIPMENT

- Computer, if used for release permit preparation

6.0 INSTRUCTIONS

6.1 Waste Gas Decay Tank Release Authorization

6.1.1 When requested by Operations to initiate a Waste Gas Decay Tank release permit, then obtain from Operation the following information as required to complete Attachment 1, Waste Gas Decay Tank Release Pennit Part 1.1, Operations Department Release Request:

- Waste Gas Decay Tank (WGDT) to be released (1-GW-TK-1A or 1-GW-TK-1B),
- Operations By Printed Name and Signature, and Date.

NOTE Requirement for sampling and analyzing WGDT for H-3 is at discretion of Supervisor RA&MC, as indicated by sampling and analysis procedure.

6.1.2 Sample and analyze, or have sampled and analyzed, the WGDT to be released. Ensure analysis is in accordance with HP-3010.031, Radioactive Gaseous Waste Sampling and Analysis.

6.1.3 Prepare Waste Gas Decay Tank Release Permit form, Part 3, Analyses and Release Data, as applicable.

a. Record Sample µCi/mL (by radionuclide based on sample analyses performed).

b. For each radionuclide, multiply the sample µCi/mL by the prerecorded Release Rate Factor, RRF. Record results in column µCi/mL x RRF.

NOTE: Either noble gases or H-3 will be controlling for release rate.

c. Sum noble gas calculated µCi/mL x RRF values. Record sum in column µCi/mL × RRF, in row $\sum$ NG µCi/mL x RRF.

d. Calculate maximum allowable release rate, CFM for noble gas (NG) and for H-3 (Calc. Max. RR, CFM). Record results of calculations as indicated on form.

$$\text{Calc. Max. RR, CFM NG} = \frac{1}{\sum \text{NG } \mu\text{Ci/mL} \times \text{RRF}}$$

$$\text{Calc. Max. RR, CFM H-3} = \frac{1}{\frac{\text{H-3}}{\mu\text{Ci/mL}} \times \text{RRF}}$$

e. Calculate the maximum percent Tech Spec (Max. % Tech Spec) the release would represent if released at 3 CFM (WGDT maximum available CFM). Record results.

$$\text{Max. \% Tech Spec} = 100 \times \frac{3\ \text{CFM}}{\text{Calc. Max. RR, CFM}}$$

Where Calc. Max. RR, CFM is the **least** calculated value for Calc. Max. RR, CFM NG and Cab. Max. RR, CFM H-3.

NOTE WGDT releases are normally allocated 10% of applicable Tech Spec release rate limits as part of control of releases from all pathways. If authorized release rate is greater than 10% of limits or because of other conditions, certain additional controls may be required as indicated on Release Pennit.

f. Determine current percent Tech Spec allocated for WGDT releases. Record value.

g. <u>IF</u> Max. % Tech Spec calculated in Substep 6.1.3.e. is greater than the current percent Tech Spec allocation, <u>THEN</u> determine requirements for WGDT release.

1. Inform the Supervisor RA&MC. Determine if the WGDT may or may not be released.

2. If the WGDT may be released, then determine a maximum **release** rate to be authorized (RRA), CFM, based on Substep 6.1.3.d Calc. Max. RR, CPM, and WGDT allocated % Tech Spec. Determine additional controls (if any).

$$\text{RRA, CFM} = \text{Calc. Max. RR, CFM} \times \frac{\text{Allocated \% Tech Spec}}{100}$$

3. If the WGDT may not be released, then determine the reason **and** suggested corrective actions to be provided to Operations.

6.1.4 Complete Waste Gas Decay Tank Release Permit form, Part 2, Release Authorization and Requirements, as applicable.

a. <u>IF</u> the WGDT may not be released, <u>THEN</u> terminate the WGDT release process.

1. Write, in Part 2, Release Authorization, Permit No., "No Release" or equivalent.

2. Advise Operations of the situation as soon as feasible.

3. GO TO Section 7.0.

NOTE: Maximum authorized release rate must be less than or equal to 3 CFM.

b. Determine a release rate to be authorized.

c. Indicate (√) the WGDT authorized for release. Verify the WGDT authorized, is the same as requested by Operations in Part **1.1.**

d. Record Maximum Authorized Release Rate, **CFM.**

e. Calculate the Percent. Tech Spec the Maximum Release Rate Authorized (Auth. Max RR, CFM) represents.

$$\text{Authorized Release Rate \% Tech Spec} = 100 \times \frac{\text{Auth. Max RR, CFM}}{\text{Calc. Max. RR, CFM}}$$

f. Record the Authorized Release Rate % Tech Spec (round off to two digits).

g. If applicable, record any conditions or controls pertaining to the release.

h. Record next available release Permit No.

i. Record HP Authorization By Printed Name and Signature, and Date.

6.1.5 Forward Waste Gas Decay Tank Release Permit form to Operations.

NOTE: Subsection **6.2** is applicable when the Waste Gas Decay Tank Release **Permit** form is returned from Operations.

6.2 Waste Gas Decay Tank Release Permit Close Out

6.2.1 Review the Waste Gas Decay Tank Release Permit form. Verify recorded release rate or rates did not exceed authorized release rate. Verify calculations of waste volume released completed in Part 1.2 by Operations.

6.2.2 Complete Waste Gas Decay Tank Release Pennit form, Part 3, Analyses and Release Data.

a. Record mL Released (volume released, from Part 1.2)

b. Calculate µCi released. Record results.

µCi Released = Sample **µCi/mL x** mL Released

c. Record Release Data By Printed Name and Initiais, and Date.

6.2.3 GO TO Section 7.0.

6.3 Reactor Containment Purge Authorization

NOTE: Purging is not permitted without a valid Reactor Containment Purge Permit. A Reactor Containment Purge **Permit** is valid **from** start of purging until routine termination ***of*** purge, or unless terminated earlier by Health Physics for cause, or as indicated on Release Permit. If purge **is** secured by a valid Containment Gas Monitor or valid Vent Vent B gas monitor (RMS) action, the Release Permit **is** considered terminated.

NOTE If damper leakage exists some purge release may occur **through** VVA. Appropriate sampling should be established to evaluate this occurrence.

6.3.1 When requested by Operations to initiate a Reactor Containment purge permit, then obtain from Operation the following information as required to complete Attachment 2, Reactor Containment Purge Permit Part I.1, Operations Department Release Request:

- Reactor Containment (Unit 1 or Unit 2) to be purged,
- Operations By Printed Name and Signature, and Date.

6.3.2 Sample and analyze, or have sampled and analyzed, the Reactor Containment atmosphere to be released. Ensure sampling and analyses include noble **gas** radionuclides, H-3, radioiodines, and particulates.

6.3.3 Prepare Reactor Containment Purge Permit form, Part 3, Analyses and Release Data, ***as*** applicable.

a. Record Sample µCi/mL (by radionuclide).

b. For each radionuclide, multiply sample µCi/mL by prerecorded Release Rate Factor, RRF. Record results in column µCi/mL × RRF.

NOTE: Either noble gas radionuclides or tritium-iodme-particulates (TIP) will be controlling for release rate. Calculating release rate. for iodines and particulates will be based on whether the main HEPA-charcoal filter banks will or will not be used during purge and if it is desired to take credit for the filters to modify RRFs.

c. Sum noble gas calculated values of µCi/mL x RRF. Record sum in column µCi/mL x RRF, in space for Σ NG µCi/mL x RRF.

d. Sum TIP radionuclide calculated values of µCi/mL x RRF. Record sum in column µCi/mL x RRF, in space for Σ TIP µCi/mL x RRF.

e. Calculate.maximum allowable release rate, CFM (Calc. Max. RR, CFM) for both noble gas radionuclides (NG) and TIP. Record results of calculations as indicated on form.

$$\text{Calc. Max. RR, CFM} = \frac{1}{\sum \mu\text{Ci/mL} \times \text{RRF}}$$

f. Calcnlate the maximum percent Tech Spec (Max. % Tech Spec) the release would represent if released at 22,000 CFM maximum available purge rate. Record results.

$$\text{Max. \% Tech Spec} = 100 \times \frac{22{,}000 \text{ CFM}}{\text{Calc. Max. RR, CFM}}$$

NOTE: Containment purges are normally allocated 10% of applicable Tech Spec dose rate limits as part of control of releases from all pathways. If authorized release rate is greater than 10% of limits or because of other conditions, certain additional controls may be required as indicated on Release Permit.

g. Determine current percent Tech Spec allocated for containment releases. Record value.

h. IF Max. % Tech Spec calculated in Substep 6.1.3.f is greater than the current percent Tech Spec ailocation, THEN determine requirements for containment release.

1. Inform the Supervisor RA&MC. Determine if the containment may or may not be released.

-

2. If the containment may be released, then determine a maximum release rate to be authorized (RRA), CFM, based on Substep 6.3.3.e Calc. Max. RR, CFM, and containment allocated % Tech Spec. Determine additional controls (if any).

$$\text{RRA, CFM} = \text{Calc. Max. RR, CFM} \times \frac{\text{Allocated \% Tech Spec}}{100}$$

3. If the containment may not be released, then determine the reason and suggested corrective actions to be provided to Operations.

6.3.4 Complete Reactor Containment Purge Permit form, Part 2, Release Authorization and Requirements, as applicable.

a. IF the containment may not be released, THEN terminate the release process.

1. Write, in Part 2, Release Authorization, Permit No., "No Release" or equivalent.
2. Advise Operations of the situation as soon as feasible.
3. GO TO Section 7.0.

NOTE: Maximum authorized release rate must be less than or equal to 22,000 CFM.

b. Determine a release rate to be authorized.

c. Indicate (√) the containment authorized for purge. Verify the containment authorized, is the same as requested by Operations in Part 1.1.

d. Record the Maximum Authorized Purge Rate, CFM.

e. Calculate the percent Tech Spec the Maximum Authorized Release Rate (Auth. Max RR, CFM) represents. Record % Tech Spec (round off to two digits).

$$\text{Authorized Purge Rate \% Tech Spec} = 100 \times \frac{\text{Auth. Max RR, CFM}}{\text{Calc. Max. RR, CFM}}$$

f. Record any conditions or controls pertaining to the release (e.g., require main HEPA-charcoal filter bank filtration, no purge during WGDT release).

1. Operations shall be instructed to monitor both VVA and VVB radiation monitors upon initiation of the release.

2. Operations shall be instructed to observe the start of the purge on MGPI monitors if available.

g. Record next available release **Pennit** No.

h. Record HP Authorization By Printed Name and Signature, and Date.

6.3.5 Forward Reactor Containment Purge **Permit** form to Operations.

6.3.6 Ensure required Reactor Containment sampling and analysis program is initiated as required to determine activity released in accordance with Subsection **6.4.**

NOTE Subsection **6.4** is applicable when the Reactor Containment Purge Permit form is returned from Operations.

6.4 Reactor **Containment Purge** Permit **Close Out**

6.4.1 Review the Reactor Containment Purge Permit **form,** Part 1.2.

a. Verify recorded purge rates did not exceed authorized release rate.

b. Verify required data are recorded.

6.4.2 Complete Reactor Containment Purge Permit form, Part 3, Analyses and Release Data.

NOTE: Particulate and radioiodines activity due to purge is calculated based on the continuous particulate and charcoal sampler in Vent Vent B.

NOTE Purge volume and noble gas and H-3 activity released is determined as instructed by Supervisor RA&MC. The following methods or alternates may be used.

- Vent Vent B sampling and analysis **at** a frequency appropriate to determine activity released during purge,
- Reactor containment volume for purge volume

a. If Vent Vent B (VVB) and Vent Vent A (VVA) sampling and analysis was implemented, then obtain sample and analysis data performed at required frequencies for noble gas and H-3 activity. Caiculate activity released.

1. For each sampling **and** analysis data set, determine VVB and VVA flow rate or VVB and VVA Flow Integrator (FI) start and stop reading, applicable to the sample.

2. Calculate volume released between saniple sets, as applicable.

Volume, mL = Vent **CFM x** Purge Time, minutes x 2.832E+4 mL/ft^3

Volume, mL = 1.42E+9 × (Vent FI stop reading - Vent FI start reading)

Where **1.42E+9 = 50,000** x **2.832E+4 mL/ft^3**

50,000 = Factor to convert difference in FI readings to ft^3

3. Calculate activity released between sample **sets.**

Activity Released, µCi = Volume, mL × Nuclide **µCi/mL**

b. If reactor containment volume is used for purge volume, then use **5.21E+10** mL.

Activity Released, **µCi** = 5.21E+10 mL × Nuclide µCi/mL

c. Record µCi released by nuclide.

d. Record Release Data By Printed Name and Initials, and Bate.

6.4.3 GO TO Section 7.0.

6.5 Reactor **Containment Hogging Authorization**

NOTE: Hogging *is* not permitted without a valid Reactor Containment Hogging Permit. A Reactor Containment Hogging Permit is valid from start of hogging until routine termination of hogging, or unless terminated earlier by Health Physics for cause, or as indicated on Release Permit. If a valid Containment Gas or Particulate Monitor (RMS) alarm occurs, the Release Permit is considered terminated.

6.5.1 When requested by Operations to initiate a containment hogging permit, then obtain from Operations the following information as required to complete Attachment 3, Reactor Containment Hogging Permit, Operations Department Release Request:

- Reactor Containment (Unit I or Unit 2) to be hogged, and
- Operations By Printed Name and Signature, and Date.

6.5.2 Sample and analyze, or have sampled and analyzed, the Reactor Containment atmosphere to be released. Ensure sampling and analyses include noble gas radionuclides, H-3, radioiodines, and particulates.

6.5.3 If hogger steam supply is contaminated (e.g., secondary coolant greater than 1E-5 μCi/mL beta-gamma activity), then prepare Attachment 4, Miscellaneous Gaseous Release Permit as instructed by Supervisor RA&MC.

6.5.4 Prepare Reactor Containment Hogging Permit form, Part 3, Analyses and Release Data, as applicable.

a. Record Sample μCi/mL (by radionuclide).

b. For each radionuclide, multiply the sample μCi/mL by the prerecorded Release Rate Factor, RRF. Record results in column μCi/mL x RRF.

NOTE Either noble gas radionuclides or tritium-iodine-particulates (TIP) will be controlling for release rate.

c. Sum noble gas calculated values of μCi/mL x RRF. Record sum in column μCi/mL x RRF, in space for Σ NG μCi/mL x RRF.

d. Sum TIP radionuclide calculated values of μCi/mL x RRF. Record sum in column μCi/mL x RRF, in space for Σ TIP μCi/mL x RRF.

e. Calculate maximum allowable release rate, CFM (Calc. Max. RR, CFM) €or both noble gas radionuclides (NG) and TIP. Record results of calculations as indicated on form.

$$\text{Calc. Max. RR, CFM} = \frac{1}{\sum \mu\text{Ci/mL} \times \text{RRF}}$$

f. Calculate the maximum percent Tech Spec (Max. % Tech Spec) the release would represent if released at 4500 CFM maximum available hogging rate. Record results.

$$\text{Max. \% Tech Spec} = 100 \times \frac{4500\ \text{CFM}}{\text{Calc. Max. RR, CFM}}$$

NOTE: Containment hogging is normally allocated 10% of applicable Tech Spec dose rate limits as part of control of releases from all pathways. If authorized release rate is greater than 10% of limits or because of other conditions, certain additional controls may be required as indicated on Release Permit.

g. Determine current percent Tech Spec allocated for containment hogging. Record value.

h. IF Percent Tech Spec calculated in Substep 6.1.3.f. is greater than the current percent Tech Spec allocation, THEN determine requirements for Containment hog.

1. Inform the Supervisor RA&MC. Determine if the containment may or may no? be hogged.

2. If the containment may be hogged, then determine a maximum release rate to be authorized (RRA), CFM, based on **Substep 6.5.4.e** Calc. Max. RR, CFM, and containment hog allocated % Tech Spec. Determine additional controls (if **any**).

$$\text{RRA, CFM} = \text{Calc. Max. RR, CFM} \times \frac{\text{Allocated \% Tech Spec}}{100}$$

3. If the containment may not be hogged, then determine the reason and suggested corrective actions to be provided to Operations.

6.5.5 Complete Reactor Containment Hogging Permit form, Part 2, Release Authorization and Requirements, as applicable.

a. IF **the** containment may not be released, THEN terminate the release process.

1. Write, in **Part** 2, Release Authorization, Permit No., "No Release" or equivalent.

2. Advise Operations of the situation as **soon** as feasible.

3. GO TO Section 7.0.

NOTE: Maximum authorized release rate must be less than or equal to 4500 CFM.

b. Determine a release rate to be authorized.

c. Indicate (√) the containment authorized for hog. Verify the containment authorized, is the same as requested by Operations in Part 1.1.

d. Record the Maximum Authorized Hog Rate, CFM.

e. Calculate the percent Tech Spec the Maximum **Authorized** Release Rate (Auth. Max RR, CFM) represents. Record % Tech Spec (round off to **two** digits).

$$\text{Authorized Purge Rate \% Tech Spec} = 100 \times \frac{\text{Auth. Max RR, CFM}}{\text{Calc. Max. RR, CFM}}$$

f. If applicable, record any conditions or controls pertaining to the release (e.g., no hogging during WGDT release).

g. Record next available release Permit No.

h. Record HP Authorization By Printed Name and Signature, and Date.

6.5.6 Forward Reactor Containment Hogging Permit form to Operations.

NOTE Subsection 6.6 is applicable when the Reactor Containment Hogging Permit form is returned from Operations.

6.6 Reactor Containment Hogging Permit Close Out

6.6.1 Review the Reactor Containment Hogging Permit form, Part 1.2.

a. Verify recorded hogging rate did not exceed authorized release rate.

b. Verify required data are recorded.

c. Verify hogging volume and rate calculations, based on data recorded by Operations.

6.6.2 Complete Reactor Containment Hogging Permit form, Part 3, Analyses and Release Data.

NOTE If gamma emitting nuclides (other than those listed on the form) are measured, they must be accounted for in release records.

a. Record mL Released.

b. Calculate µCi released for each nuclide with data. Record results.

µCi Released = Sample µCi/mL x mL Released

c. Record Release Data By Printed Name and Initials, and Date.

6.6.3 *GO* TO Section *7.0*.

6.7 Miscellaneous Gaseous Release Authorization

NOTE: Miscellaneous gaseous releases include batch releases of noble gases which may not be accounted for by routine sampling, or any planned release not being routed through the Process Vent or Ventilation Vents (e.g., steam driven aux feedwater pump testing if primary to secondary leakage exists).

NOTE Miscellaneous releases through either VVA or VVB may result in releases through the other stack if damper leakage exisis.

6.7.1 When requested by Operations to initiate a miscellaneous gaseous release permit, then obtain from Operations the following information as required to complete Attachment **4,** Miscellaneous Gaseous Release Permit, Part 1.1, Operations Department Release Request:

- Source of release,
- Maximum expected release rate, CFM,
- Release medium (gas, steam, other), and
- Operations By Printed Name and Signature, and Date.

6.7.2 Sample and analyze, or have sampled and analyzed, source **to** be released. Ensure analysis is IAW with **Up-3010.031,** Radioactive Gaseous Waste Sampling and Analysis.

6.7.3 Prepare Miscellaneous Gaseous Release Permit form, Part **3,** Analyses and Release Data, as applicable.

a. Record Sample µCi/mL (by radionuclide based on sample analyses performed).

b. Indicate the release mode. Check (√) if the release is considered Ground level (release other than Process Vent) or Mixed Mode (Process Vent release).

c. For each noble ***gas*** radionuclide, multiply sample µCi/mL by prerecorded Release Rate Factor, RRF for the release mode. Record results in column µCi/mL x RRF.

NOTE The Supervisor RA&MC may authorize partition factor use for s t e m release calculations (e.g., steam driven aux feedwater pump). If the release will be filtered, then credit may be taken for the filters used.

d. If partition factors are used, then adjust RRFs as applicable (e.g., if a factor of 0.1 is used, then the recorded RRF is reduced by 0.1). Record, in column Remarks or Partition Factor for applicable nuclide, the factor used.

e. If credit is to be taken for filter use, then adjust RRFs as applicable.

1. Indicate (√) filter used (one or both HEPA and charcoal).

2. If HEPA filter is used, then reduce particulate RRFs by 0.01 or other value as directed by the Supervisor RA&MC.

3. If charcoal filter is used, then reduce I-131 and I-133 RRF by 0.1 (e.g., for ground level release, I-131 RRF is reduced from 4.75E+1 to 4.75E+0).

f. For each particulate or iodine nuclide, multiply sample µCi/mL by RRF (adjusted if applicable) for the release mode. Record results in column µCi/mL x RRF.

NOTE Either noble gas radionuclides or tritium-iodine-particulates (TIP) will be controlling for release rate.

g. Sum noble gas calculated values of µCi/mL x RRF. Record sum in space for Σ NG µCi/mL x RRF (in column µCi/mL x RRF).

h. Sum TIP radionuclide calculated values of µCi/mL x RRF. Record sum in space for Σ TIP µCi/mL × RRF (in column µCi/mL × RRF).

i. Calculate maximum allowable release rate, CFM (Calc. Max. RR, CFM) for both noble gas radionuclides (NG) and TIP. Record results of calculations as indicated on form.

$$\text{Calc. Max. RR, CFM} = \frac{1}{\Sigma\ \mu\text{Ci/mL} \times \text{RRF}}$$

j. Calculate the maximum percent Tech Spec (Max. % Tech Spec) the release would represent if released at maximum expected CFM release rate. Record results.

$$\text{Max. \% Tech Spec} = 100 \times \frac{\text{maximum expected release rate CFM}}{\text{Calc. Max. RR, CFM}}$$

NOTE: Miscellaneous releases are normally allocated 10% of applicable Tech Spec dose rate limits as part of control of releases from all pathways. If authorized release rate is greater than 10% of Limits or because of other conditions, **certain** additional controls may be required as indicated on Release Permit.

k. Determine current percent Tech Spec allocated for miscellaneous releases. Record value.

1 IF Max. % Tech **Spec** calculated in Substep **6.7.3.j** is greater than the current percent Tech Spec allocation, THEN determine requirements for Miscellaneous release.

1. Inform the Supervisor RA&MC. Determine if the Miscellaneous source may or may not be released.

2. If the source may be released, then determine a maximum release rate to be authorized (RRA), CFM, based on Substep 6.7.3.i, Calc. Max. RR, CFM, and miscellaneous releases allocated % Tech Spec. Determine additional controls (if any).

$$\text{RRA, CFM} = \text{Calc. Max. RR, CFM} \times \frac{\text{Allocated \% Tech Spec}}{100}$$

3. If the source may not be released, then determine the reason and suggested corrective actions to be provided to Operations.

6.7.4 Coniplete Miscellaneous Gaseous Release Permit form, Part 2, Release Authorization and Requirements, as applicable.

a. IF the Miscellaneous source may not be released, THEN terminate the Miscellaneous release process.

1. Write, in Part 2, Release Authorization, Permit No., "No Release" or equivalent.
2. Advise Operations of the situation as soon as feasible.
3. GO TO Section 7.0.

NOTE: Maximum authorized release rate must be less than or equal to maximum expected release rate indicated by Operations.

b. Determine a release rate to be authorized. For BRT Vent releases, use 40 scfm unless another value is authorized by Supervisor RA&MC.

c. Indicate source authorized for released. Verify the source authorized, is the same as requested by Operations in Part 1.1.

d. Record the Maximum Release Rate Authorized, CFM.

e. Calculate the Percent Tech Spec the Maximum Reiease Rate Authorized (Auth. Max RR. CFM) represents.

$$\text{Authorized Reiease Rate \% Tech Spec} = 100 \times \frac{\text{Auth. Max RR, CFM}}{\text{Calc. Max. RR, CFM}}$$

f. Record the Authorized Release Rate % Tech Spec (round off to two digits).

g. If applicable, record any conditions or controls pertaining to the release..

1. Releases through either VVA or VVB shall include conditions or controls to monitor both VVA and VVB radiation monitors from onset of release.

2. Controls and conditions for releases via VVA or VVB shall include instructions for Operations to mark start and end of release on MGPI strip chart (if applicable).

h. Record next available release Permit No.

i. Record HP Authorization printed Name and Signature, and Date.

6.7.5 Forward Miscellaneous Gaseous Release Pennit form to Operations.

NOTE: Subsection 6.8 is applicable when the Miscellaneous Gaseous Release Permit form is returned from Operations.

6.8 Miscellaneous Gaseous Release Permit Close Out

6.8.1 Review the Miscellaneous Gaseous Release Permit form. Verify recorded release rate or rates did not exceed authorized release rate. Verify calculations of waste volume released completed in Part 1.2 by Operations. For BRT Vent releases, determine the volume released based on volume of water added to the BRT being filled.

6.8.2 Complete Miscellaneous Gaseous Release Permit, Part 3, Analyses and Release Data.

a. Record mL Released (volume released, from Part 1.2)

NOTE For BRT Vents, unless another method is approved by Supervisor RA&MC, use highest concentration released for each isotope detected during the sample period in all samples taken.

b. Calculate µCi released. Record results.

- Ensure that for iodine samples obtained fron the MGPI iodine samplers, the concentration was corrected for transmission line losses IAW, HP-3010.031, Radioactive Gaseous Waste Sampling and Analysis.
- µCi Released = Sample µCi/mL × mL Released

c. Record Release Data By Printed Name and Initials, and Date.

7.0 FOLLOW-ON

7.1 Follow-On Tasks

7.1.1 Forward completed forms, or equivalent, to Supervisor RA&MC for review.

- Waste Gas Decay Tank Release Permit
- Reactor Containment Purge Permit
- Reactor Containment Hogging Permit
- Miscellaneous Gaseous Release Permit

7.1.2 Place the reviewed forms in the designated location pending completion of release accountability records, in accordance with HP-3010.032, Radioactive Gaseous Waste Accountability and Dose Calculations.

7.2 Records Disposition

Forward the following records to Records Management in accordance with VPAP-1701, Records Management.

- Waste Gas Decay Tank Release Permit
- Reactor Containment Purge Permit
- Reactor Containment Hogging Permit
- Miscellaneous Gaseous Release Permit

ATTACHMENT 1
page 1 of 1)
WASTE GAS DECAY TANK RELEASE PERMIT

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Prior to release, complete Part 1.1. Forward Release Permit to Health Physics for release authorization.
WGDT to be released: ☐ 1-GW-TK-1A ☐ 1-GW-TK-1B

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS DEPARTMENT RELEASE DATA

WGDT released: ☐ 1-GW-TK-1A ☐ 1-GW-TK-1B
At each start and stop of release, record data indicated below. After release, return Permit to Health Physics.

Release Status	Date and Time	WGDT PSIG	Release Rate CFM	Operations By (Printed Name)	(Initials)	Remarks
stop						
Stop						

Vol. Released, mL = 8.9E+5 x (PSIG Start - PSIG Stop) = ____________ mL.

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS
As indicated and marked. A valid Process Vent RMS alarm voids this permit.

PERMIT No.

☐ 1-GW-TK-1A ☐ 1-GW-TK-1B
Maximum Authorized Release Rate: ___________ **CFM.** Authorized Release Rate is: _______ % Tech Spec.
Required controls or conditions ____________________

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - ANALYSES AND RELEASE DATA	Release Data By (Printed Name)	(Initials)	Date

Nuclide	Complete Prior to Release. Sample µCi/mL	Release Rate Factor, RRF	µCi/mL x RRF	mL Released µCi Released	Remarks
Kr-85m		1.32E-3*			
Kr-85		2.57E-4**			
Kr-87		6.70E-3*			
Kr-88		1.66E-2*			
Xe-131m		1.22E-4**			
Xe-133m		2.84E-4*			
Xe-133		3.33E-4*			
Xe-135m		3.53E-3*			
Xe-135		2.05E-3*			
Xe-138		1.00E-2*			
Ar-41		1.00E-2*			
		Σ NG µCi/mL x RRF			
H-3***		4.25E-4			

* Based on total body dose factor. ** Based on skin dose factors. *** If required by Supv R.A.

$$\text{Calc. Max. RR, CFM NG} = \frac{1}{\Sigma \text{ NG } \mu\text{Ci/mL x RRF}} = \_\_\_\_\_\_ \text{CFM} \qquad \text{Max. \% Tech Spec} = \frac{100 \times 3 \text{ CFM}}{\text{Calc. Max. RR, CFM}} = \_\_\_\_\_\_$$

$$\text{Calc. Max. RR, CFM H-3} = \frac{1}{\text{H-3 } \mu\text{Ci/mL x RRF}} = \_\_\_\_\_\_ \text{CFM}$$

%Tech Spec Allocated for WGDT = ____________

Transfer release data to release records? ☐ Yes ☐ No

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 2
page 1 of 1)
REACTOR CONTAINMENT PURGE PERMIT

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Prior tu release, complete Part 1.1. Forward Release Permit to Health Physics for release authorization.
Reactor Containment: ☐ Unit 1 ☐ Unit 2. Release Type: Purge

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS DEPARTMENT RELEASE DATA

At start and stop of purge, record data indicated below. After purge, return Purge Permit to Health Physics.

Release Status	Date and Time	Purge Release Rate CFM	Operations By (Printed Name)	Operations By (Signature)	Remarks
Start					
Stop					

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS
(as indicated and marked) A valid Containment gas monitor high alarm voids this permit.

PERMIT No.

☐ Unit 1 ☐ Unit 2 **Maximum Authorized Purge Rate:** __________ **CFM.** Authorized Burge Rate is: ______ % Tech Spec.
Required controls or conditions (See Attached)________________________________

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - ANALYSES AND RELEASE DATA	Release Data By (Printed Name)	(Initials)	Date

Nuclide	Sample µCi/mL	Rel F Factor, RRF	µCi/mL x Applicable RRF	µCi Released	Remarks or RRF Adjustment Factor
	[illegible] Release			mL Released	
Kr-85m		1.03E-2*			
Kr-85		1.99E-3**			
Kr-87		5.20E-2*			
Kr-88		1.29E-1*			
Xe-131m		9.48E-4**			
Xe-133m		2.20E-3*			
Xe-133		2.58E-3*			
Xe-135m		2.74E-2*			
Xe-135		1.59E-2*			
Xe-138		7.75E-2*			
Ar-41		7.76E-2*			
	Σ NG µCi/mL x RRF			* Based on total body dose factor. ** Based on skin dose factors.	

Take credit for HEPA-Charcoal filtration? ☐ Yes ☐ No. If yes, then reduce iodine and particulate RRF by appropriate factor.

Nuclide	Sample µCi/mL	RRF	µCi/mL x Applicable RRF	µCi Released	Remarks or RRF Adjustment Factor
H-3		3.30E-3			
Cr-51		2.50E-4		N/A	
Te-127m		1.77E-2		N/A	
Te-129m		1.85E-2		N/A	
I-131		4.75E+1		N/A	
I-133		1.13E+1		N/A	
	Σ TIP* µCi/mL x RRF			* TIP = H-3, I-131, I-133, & Particulates.	

Calc. Max. FIR, CFM NG = $\dfrac{1}{\Sigma_{NG}\ \mu Ci/mL \times RRF}$ = __________ CFM

Calc. Max. RR, CFM TIP = $\dfrac{1}{\Sigma_{TIP}\ \mu Ci/mL \times RRF}$ = __________ CFM

Max. % Tech Spec = $\dfrac{100 \times 22{,}000 \text{ CFM}}{\text{Calc. Max. RR, CFM**}}$ = ______

%Tech Spec Allocated for Source = ____________

** Use lowest NG or TIP Calc. Max RR. CFM.

Transfer release data tu release records? ☐ Yes ☐ No

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 3
(Page. 1 of 1)
REACTOR CONTAINMENT HOGGING PERMIT

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Prior to release, complete Part 1.1. Forward Release Permit to Health Physics for release authorization.
Reactor Containment: ☐ Unit 1 ☐ Unit 2. Release Type: Hogging

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS DEPARTMENT RELEASE DATA

During and after release, complete Part 1.2. Return to Health Physics for completion of release records.

Release Status	Date and Time	Rx. Cont. PSIA	Max Hogging Rate CFM	Operations By (Printed Name)	Operations By (Signature)	Remarks
Start						
Stop						

Calculate volume release and hogging release rate. Vol. Released, ft^3 = 1.25E+5 x (PSIA Start - PSIA Stop) = ________ ft^3.

Hogging CFM = $\frac{\text{Volume Released, ft}^3}{\text{Hogging minutes}}$ = $\frac{\_\_\_\_\_\_ \text{ ft}^3}{\_\_\_\_\_\_ \text{ minutes}}$ = ________ CFM mL = ________ ft^3 x 28,320 = ________ mL.

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS
As indicated and marked. A valid Containment RMS alarm voids this permit.

PERMIT No.

☐ Unit 1 ☐ Unit 2 **Max. Authorized** Release Rate: ____________ **CFM.** Rate is: ______% of Tech Spec limit.
Required controls or conditions ______________________________

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - ANALYSES AND RELEASE DATA	Release Data By (Printed Name)	(Initials)	Date

Nuclide	Complete Prior to Release: Sample µCi/mL	Release Rate Factor, RRF	µCi/mL x Applicable RRF	mL Released: µCi Released	Remarks or RRF Adjustment Factor
Kr-85m		[illegible].03E-2*			
Kr-85		1.99E-3**			
Kr-87		5.20E-2*			
Kr-88		1.29E-1*			
Xe-131m		9.48E-4**			
Xe-133m		2.20E-3*			
Xe-133		2.58E-3*			
Xe-135m		2.74E-2*			
Xe-135		1.59E-2*			
Xe-138		7.75E-2*			
Ar-41		7.76E-2*			
Σ Noble Gas Radionuclide µCi/mL x RRF				* Based on total body dose *factor.* ** Based on skin dose factors.	
H-3		3.30E-3			
Cr-51		2.50E-4			
Te-127m		1.77E-2			
Te-129m		1.85E-2			
I-131		4.75E+1			
I-133		1.13E+1			
Σ TIP* µCi/mL x RRF				* TIP = H-3, I-131, I-133, & Particulates	

Calc. Max. RR, CFM NG = $\frac{1}{\Sigma \text{ NG } \mu\text{Ci/mL x RRF}}$ = ____________CFM Max. % Tech Spec = $\frac{100 \text{ x } 4{,}500 \text{ CFM}}{\text{Calc. Max. RR, CFM**}}$ = ______

% Tech Spec Allocated for Source = ______________

Calc. Max. RR, CFM TIP = $\frac{1}{\Sigma \text{ TIP } \mu\text{Ci/mL x RRF}}$ = ____________CFM

** Use lowest NG or TIP Calc. Max RR. CFM.

Supv. RA&MC (Signature)	Date	Release Records By (Printed Name)	Release Records By (Signature)	Date

ATTACHMENT 4
(Page 1 of 1)
MISCELLANEOUS GASEOUS RELEASE PERMIT

PART 1.1 - OPERATIONS DEPARTMENT RELEASE REQUEST

Prior to release, complete Part 1.1, forward Release Permit to Health Physics for release authorization.

Source of release ______

Maximum expected release rate ______ CFM, as ☐ Gas ☐ Steam ☐ ______

Operations By (Printed Name)	Operations By (Signature)	Date

PART 1.2 - OPERATIONS DEPARTMENT RELEASE DATA

During and after release, complete Part 1.2. Return to Health Physics for completion of release records.

Release Status	Date and Time	Release Rate CFM	Operations By (Printed Name)	Operations By (Signature)	Remarks
Start					
stop					

Volume Released, ft^3 = ______ CFM x ______ minutes = ______ ft^3. mL = ______ ft^3 x 28,320 = ______ mL

PART 2 - RELEASE AUTHORIZATION AND REQUIREMENTS

As indicated and marked. A valid associated RMS alarm voids this permit.

PERMIT No.

Source ______ **Maximum Authorized Release Rate:** ______ **CFM.** Authorized Rate is: ______ % Tech Spec.

Required controls or conditions (See Attached) ______

HP Authorization By (Printed Name)	HP Authorization By (Signature)	Date

PART 3 - ANALYSES AND RELEASE DATA	Release Data By (Printed Name)	(Initials)	Date

Nuclide	Complete Prior to Release. Indicate Release Type. Sample µCi/mL	Release Rate Factor, RRF ☐ Ground	Release Rate Factor, RRF ☐ Mixed Mode	µCi/mL x Applicable RRF	mL Released µCi Released	Remarks or Adjustment Factors
Kr-85m		1.03E-2*	1.32E-3*			
Kr-85		1.99E-3**	2.57E-4**			
Kr-87		5.20E-2*	6.70E-3*			
Kr-88		1.29E-1*	1.66E-2*			
Xe-131m		9.48E-4**	1.22E-4**			
Xe-133m		2.20E-3*	2.84E-4*			
Xe-133		2.58E-3*	3.33E-4*			
Xe-135m		2.74E-2*	3.53E-3*			
Xe-135		1.59E-2*	2.05E-3*			
Xe-138		7.75E-2*	1.00E-2*			
Ar-41		7.76E-2*	1.00E-2*			
			Σ NG µCi/mL x RRF		* Based on total body dose factor. ** Based on skin dose factors.	
H-3		3.30E-3	4.25E-4			
Cr-51		2.50E-4	3.21E-5			
Te-127m		1.77E-2	2.29E-3			
Te-129m		1.85E-2	2.39E-3			
I-131		4.75E+1	6.14E+0			
I-133		1.13E+1	1.45E+0			
☐ No filter ☐ HEPA ☐ Charcoal			Σ TIP* µCi/mL x RRF		* TIP = H-3, I-131, I-133, & particulates.	

If HEPA or charcoal filter is or are used, then particulate or iodine RRF may be reduced by an appropriate adjustment factor (e.g., 0.1). If partition factors are applicable, then RRF for applicable nuclides may be reduced by appropriate partition factor.

Calc. Max. **RR**, CFM NG = $\frac{1}{\Sigma \text{ NG } \mu\text{Ci/mL x RRF}}$ = ______ CFM

Calc. Max. RR, CFM TIP = $\frac{1}{\Sigma \text{ TIP } \mu\text{Ci/mL x RRF}}$ = ______ CFM

Max. % Tech Spec = $\frac{100 \text{ x } \_\_\_\_ \text{ CFM}}{\text{Calc. Max. RR, CFM**}}$ = ______

% Tech Spec Allocated for Source = ______

** Use lowest NG or TIP Cab. Max RR. CFM.

Data to Release Records By (Printed Name)	(Initials)	Date	Supv. RA&MC (Signature)	Date

Dominion NORTH ANNA POWER STATION	PROCEDURE NO: HP-3010.023	
	UNIT NO: 1 AND 2	REVISION NO: 10
PROCEDURE TYPE: HEALTH PHYSICS	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE

ABNORMAL LIQUID RELEASE

EP						

REVISION SUMMARY:

- Changed corporate name on cover sheet and header
- Added EP to cover sheet to incorporate concern HP-01-0002
- Removed reference to old T.S throughout procedure (see change bars)
- Added reference 2.3.8, HP-3010.050
- Added reference 2.3.9, NUREG/CR *5569*
- Step 7.1.4, added reference to 2103N
- Attachment 2, added notification of Ops Shift Supervisor

Writer: H. Sinclare	Reviewer: J. Breeden

ELECTRON IC DISTRIBUTION — APPROVAL ON FILE

TABLE OF CONTENTS

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1.0 PURPOSE

1.1 This procedure provides instructions for responding to an abnormal release of radioactive materials in liquid pathways, including sampling, analysis, evaluation, and accountability.

2.0 REFERENCES

2.1 **Source Documents**

2.1.1 10 CFR 20, Standards for Protection Against Radiation

2.1.2 10 CFR 50, Domestic Licensing of Production and Utilization Facilities

2.1.3 40 CFR 302, Designation, Reportable Quantities, and Notification

2.2 **Technical Specifications**

2.2.1 ITS 5.5.4, Radioactive Effluent Controls Program

2.2.2 ITS 5.6.3, Annual Radioactive Effluent Release Report

2.2.3 ITS 5.5.1, Offsite Dose Calculation Manual (ODCM)

2.2.4 SNSOC Responsibilities (UFSAR)

2.3 **Technical References**

2.3.1 VPAP-1701, Records Management

2.3.2 VPAP-2103N, Offsite Dose Calculation Manual (ODCM, NAPS)

2.3.3 VPAP-2802, Notifications and Reports

2.3.4 North Anna Power Station Abnormal Operating Procedures (APs)

2.3.5 North Anna Power Station Emergency Plan Implementing Procedures (EPIPs)

2.3.6 HF-1032.010, Radiological Survey Records

2.3.7 HF-3010.022, Radioactive Liquid Waste Accountability and Dose Calculations

2.3.8 HP-3010.050, Preparing Effluent Records and Reports Using Computer Programs

2.3.9 NUREG/CR 5569, Wealth Physics Position Data Base, HPPOS 254, Definition of Unplanned Release

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2.4 Commitment Documents

None

3.0 INITIAL CONDITIONS

An abnomal release of radioactive materials in a liquid pathway has occurred, typically as indicated by one or more of the following:

- Operations Department determination,
- Radiation Monitoring System (RMS) indication,
- Observation of a release in progress which meets VPAP-2103N, Offsite Dose Calculation Manual, classification criteria for unplanned release, or
- Implementation of an Operation Abnormal Operating Procedure (AP) or Emergency Plan Implementing Procedures (EPIPs).

4.0 PRECAUTIONS AND **LIMITATIONS**

4.1 The Supervisor Radiological Analysis and Material Control (RA&MC) shall ensure qualified personnel are available to perform the instructions provided in this procedure.

4.2 This procedure has 5 subsections for the following objectives.

- Actions to be taken in the event of a *stem* generator high capacity blowdown system, radiation monitor alarm
- Sampling and analyses for determining the radionuclides released
- Evaluating the release, including determining the activity released and an assessment of percent Technical Specifications
- Evaluating liquid activity concentrations in unrestricted area for NRC notification requirements
- Accounting for the activity released in release records

Use of a given subsection is based on the particular needs at the time. For a release to an unrestricted area, sampling and analyses, and evaluating the release are required as a minimum.

4.3 If EPIPs have been initiated due to an abnormal release, then the following apply.

- Implementation of EPIPs takes priority over this procedure.
- Use of this procedure during the emergency phase shall be as directed by responsible emergency response personnel.
- Use of this procedure upon completion of the emergency phase (e.g., for nuclide accountability) shall be as directed by Supervisor RA&MC.
- An EPA RQ screening will likely be required in accordance with this procedure.

4.4 Accountability should be completed in a timely manner such that pontine 31 day dose calculations include the contributions of the abnormal release.

4.5 If ODCM based dose calculations due to the abnormal release are desired, then calculate dose in accordance with HP-3010.022, Radioactive Liquid Waste Accountability and Dose Calculations.

4.6 Computer generated worksheets and records are acceptable.

5.0 SPECIAL TOOLS AND EQUIPMENT

- Computer, if used for records or dose calculations

6.0 INSTRUCTIONS

NOTE The following section provides guidance on actions to be taken in the event of a stem generator high capacity blowdown system radiation monitor alarm. If the abnormal release is <u>not</u> due to such an event, then go to section 6.2.

6.1 Actions to be Taken in the Event of a Steam Generator High Capacity Blowdown System Radiation Monitor Alarm.

6.1.1 Obtain or have obtained a one (1) liter (l) sample from Unit-1 (Unit-2) steam generator high capacity blowdown system effluent. If obtaining the sample, then:

- At sample panel 1-EI-CB-385M2 (Unit-I) or 2-EI-CB-385M2 (Unit-2), Open 1-SS-786 (Unit-1) or 2-SS-1037 (Unit-2)
- Purge approximately six (6) liters to flush line (purge volume is identical for both units)
- Obtain a one (1) liter sample
- Shut 1-SS-786 (Unit-1) or 2-SS-1037 (Unit-2)

NOTE Do not delay gamma analysis while preparing tritium samples.

6.1.2 Analyze the sample for principal gamma emitters and tritium.

6.1.3 If either gamma emitters and/or tritium are detected, then perform the following actions:

- Have Chemistry obtain samples of the affected unit's individual steam generators, if not already obtained. Analyze for gamma emitters and tritium.
- Notify Supervisor RA&MC or designee.
- Obtain from Operations the glow rate of the release. If unable to obtain a flowrate, use a value of 201 gpm and make a notation to that effect in the remarks.

6.1.4 If peaks, other than those due to naturally occurring isotopes are detected, then contact the Supervisor RA&MC or designee for advice/guidance OR identification.

6.1.5 If no peaks, other than those due to naturally occurring isotopes are detected, then inform Operations that no radioactivity was detected and terminate this procedure.

6.2 **Sampling and Analysis for Abnormal Liquid Release**

NOTE: The objective of sampling is to obtain representative sample or samples so that total activity released to unrestricted areas and nuclide concentrations in unrestricted areas may be evaluated as accurately as possible. A sample of the undiluted liquid is preferred; if this is not practical, a diluted sample is the next choice.

NOTE: Sample identification data should be marked on the container. The data should include sample volume, location where sample was obtained, and the sample date and time.

6.2.1 Obtain samples representative of release. As practical, apply the following considerations.

- If the release is ongoing, then obtain at Least one sample of the undiluted liquid.
- If an undiluted sample cannot be obtained or the flow rate can not be determined, then obtain at least one sample from discharge canal which will contain the diluted release.
- If the release has been terminated, then obtain at least one sample of any remaining undiluted liquid.
- If the release is through a pathway with a sampler, then obtain a sample from the sampler's sample collection vessel.
- If liquid was released through a storm drain and an undiluted sample cannot be obtained, then obtain a sample from at least one manhole downstream of the Liquid entry point into the storm drain system.

- Look for any indication which may assist in estimating release rate or volume.
- Request the on-duty Shift Supervisor Health Physics or Supervisor RA&MC or designee to determine the need to obtain off site samples.

6.2.2 Transfer samples to appropriate location for preparation and analyses (normally the Hot Lab and Count Room).

NOTE Initial assessments may be made without tritium analysis results, if deemed appropriate.

6.2.3 Analyze the samples for gamma isotopic and tritium, if feasible.

a. For each nuclide with measured activity, express results as µCi/mL.

b. If a diluted sample was obtained (e.g., from discharge canal) and analyzed, then clearly mark the analysis printout to indicate a diluted sample was counted. Record an amount for the estimated dilution which occurred or other data applicable to the dilution (intent is to be able to late estimate undiluted source activity µCi/mL).

c. Ensure the analysis printout indicates the sample location and source of release (e.g., 1-RS-TK-lA, or discharge canal sample during release of 1-RS-TK-1A).

6.2.4 Save all samples until disposal is authorized by Supervisor RA&MC.

6.2.5 Determine nuclide accountability as instructed by Supervisor RA&MC.

6.3 Evaluating Abnormal Liquid Release

NOTE At least one Abnormal Liquid Release to Unrestricted Area Worksheet shall be completed which includes an evaluation for % Tech Spec.

NOTE: It is not necessary to complete the release description prior to evaluating the release. Each step should be completed as the situation, manpower, and data allow.

6.3.1 Initiate an Abnormal Liquid Release to Unrestricted Area Worksheet (Attachment E). Record available data in Part 1, Release Description.

a. Record a Worksheet ID Number (to uniquely identify the worksheet).

b. Record Release Start Date and Time.

c. If the release has been terminated, then record Release Stop Date and Time. Calculate and record the Release Duration, Minutes.

d. Indicate (√) the Source of Release. If the particular source is not prerecorded, then check (√) the blank box and record a description of source.

e. Obtain from Operations the peak waste stream flow rate, GPM. Record the value and other applicable information.

1. If Operations indicates the release rate is estimated, then check (√) Estimated Flow Rate.

2. If Operations can not provide a release rate, then check (√) 6,000 GPM Default.

f. Record Number of Circulating Water Pumps Running.

g. Calculate and record Dilution Water blow Rate, GPM.

Dilution Water Flow Rate, GPM = No. of Circ. Water Pumps Running x [218,000 x (% throttle/100)]

6.3.2 Complete Abnormal Liquid Release to Unrestricted Area Worksheet. Part 2, % Tech. Spec. and Activity Released Determination.

a. For each nuclide with measured activity, record Source µCi/mL (based on analyses of the liquid released).

1. If analytical results are available for a source sample, then record these values.

2. If only analytical results of diluted sample are available, then perform the required adjustments to estimate source µCi/mL. If applicabie, use the following equation.

$$\text{Source } \mu\text{Ci/mL} = \text{Sample } \mu\text{Ci/mL} \times \frac{\text{Dilution Water Flow Rate, GPM}}{\text{Waste Flow Rate, GPM}}$$

b. If a measured nuclide is not prerecorded, then record the nuclide symbol and a value for nuclide Allowable Concentration in Water (ACW), µCi/mL.

ACW, µCi/mL = 10 x 10CFR 20 Appendix B, Tabie 2, Col. 2 value for nuclides other than noble gas radionuclides, or
= 2E-4 for noble gas radionuclides

c. For each nuclide with recorded activity, calculate and record Source ACW Fraction as indicated on the worksheet Sum the Source ACW Fractions. Record Total.

d. Calculate Source Maximum Allowable Flow Rate, GPM. Record results.

$$\text{Maximum Allowable Flow Rate, GPM} = \frac{\text{Dilution Water Flow Rate, GPM}}{\text{Sum of Source ACW Fractions}}$$

e. Calculate Maximum % Tech. Spec. Record results.

$$\text{Maximum \% Tech. Spec.} = \frac{100 \times \text{Waste Peak Flow Rate, GPM}}{\text{Maximum Allowable Flow Rate, GPM}}$$

f. For each applicable batch or continuous liquid waste release occurring concurrently with the abnomal release, obtain Maximum % Tech. Spec. Add these values to the abnomal release % Tech. Spec. to determine Total % Tech. Spec. Record results.

6.3.3 IF % Tech Spec evaluation is greater than 100%, THEN notify the Shift Supervisor Health Physics and Supervisor RA&MC. Perform an EPA RQ screening and an evaluation for 10 CFR 50.72 notification applicability.

NOTE: The screening worksheet is intended to determine if notification to the National Response Center is required in accordance with 40 CFR 302. If the sum of fractions of activity released to RQ values is less than one, *then* notification *is* not required. If the value equals or exceeds one, then notification is required. The sum of Fractions of activity released includes concurrent releases. A follow-up evaluation will be required during completion of Subsection 7.1, Follow-On Tasks.

a. For each nuclide with measured activity, calculate the µCi Released.
µCi Released = Source µCi/mL X mL Released
Where mL Released *is* volume of waste released and represented by the sample.

b. Ensure values are available for the curies released for each nuclide for a 24 hour time period which includes all liquid releases during the period, including those exceeding and not exceeding Tech Specs.

c. Initiate an EPA RQ Screening Worksheet for Liquid Release (Attachment 2, Form HP-3010.023-2).

d. Record the release data. Calculate and record Release RQ Fraction.

NOTE The Qperations Shift Supervisor is responsible for ensuring required 40 CFR 302 notification is accomplished.

e. IF Total Release RQ Fraction is greater than 1, THEN notify the Shift Supervisor Health Physics. **Ensure** the Operations Shift Supervisor is notified.

f. Evaluate release for 10 CFR 50.9'2 notification applicability in accordance with Subsection 6.4, Evaluating Liquid Concentrations in Unrestricted Areas.

6.3.4 On each worksheet completed, record Prepared By Printed Name and Signature, and Date.

6.3.5 Forward the completed worksheets to the Shift Supervisor Health Physics.

6.3.6 Forward a copy of the following worksheets, if completed and indicated by the Shift Supervisor Health Physics, to the Operations Shift Supervisor for review:

- Abnormal Liquid Release to Unrestricted Area Worksheet (Attachment 1),
- EPA RQ Screening Worksheet for Liquid Release (Attachment 2), and
- Abnormal Liquid Release - Liquid Concentration Worksheet (Attachment 3).

6.4 Evaluating Liquid Concentrations in Unrestricted Areas

NOTE: If an abnormal release has the potential to cause concentrations exceeding 20 times 10 CFR 20 Appendix B Table 2, Column 2 values (other than tritium and dissolved radiogases) when averaged over 1 hour in an unrestricted area, then the release shall be evaluated for notification required by 10 CFR 50.72.

6.4.1 Initiate an Abnormal Liquid Release - Liquid Concentration Worksheet (Attachment 3). Record applicable data as required to describe release.

a. Record a Worksheet ID Number *(to* uniquely identify the worksheet).

b. Record applicable data as required to identify release and any assumptions used to determine liquid radioactivity concentrations.

c. Record the 1 Hour Release Start Date and Time and Stop Date and Time.

d. Record data for diluted waste volume.

- Waste Volume Released in 1 Hour Period is the 1 **hour** period of interest which includes the peak **waste** stream **flow** rate.
- **IF** release duration is < 1 hour, **THEN** waste volume is peak waste stream flow rate **(gpm)** x release duration (min).
- **IF** release duration is ≥ 1 hour, **THEN** waste volume is peak waste stream flow rate (gpm) x 60 minutes.
- Dilution Water Volume Available for Waste in 1 Hour Period is volume of dilution water in the **1** hour period of interest.
- Total Diluted Waste Volume in 1 Hour Period, Gallons is the volume of waste plus volume of dilution water in 1 hour period of interest.
- Total Diluted Waste Volume in 1 Hour Period, mL is volume expressed in mL.

e. For each nuclide known or suspected to have been released, record the Activity Released in 1 Hour (μCi, release **during** the 1 hour period of interest).

f. For each nuclide **with** recorded activity released, calculate liquid concentration, **μCi/mL,** in unrestricted area (Unrestricted μCi/mL) as indicated **on** worksheet.

g. For each nuclide **with** recorded **data,** calculate fraction of effluent concentration in water Limit (ECW Fraction). Record results.

$$\text{ECW Fraction} = \frac{\text{Unrestricted } \mu\text{Ci/mL}}{\text{ECW } \mu\text{Ci/mL}}$$

h. Calculate and record Total ECW Fraction.

i. Indicate (√) the applicable box for the NRC notification requirement based on Total ECW Fractions and advise HP **Shift** Supervisor.

j. Record any remarks or conclusions considered appropriate.

k. Record Prepared By Printed Name and Signature, **and** Date.

6.5 Accounting for Abnormal Liquid Release

NOTE During the evaluation of an abnormal release, several Abnormal Liquid Release to Unrestricted Area Worksheets may be completed. The worksheet or worksheets selected to be used for accountability should be designated by the Supervisor RA&MC as appropriate for this purpose.

6.5.1 Obtain the appropriate Abnormal Liquid Release to Unrestricted Area Worksheets to be used to account for the activity released during the abnormal release.

a. If required, calculate and record μCi released for each measured nuclide.

NOTE Abnormal liquid release data is accounted for similar to accounting for routine planned releases. The following steps refer to forms provided in procedure HP-3010.022, Radioactive Liquid Waste Accountability and Dose Calculations. Equivalent forms or alternate accountability techniques are acceptable if activity released is addressed.

NOTE The abnormal release may be accounted for as either a batch release or as a continuous release as considered appropriate. The choice is not critical since dose calculations do not depend on the release type,

6.5.2 Obtain the analysis summary worksheet form which covers the period when the abnormal release occurred. Either of the following forms may be applicable:

- Batch Liquid Waste Analysis Summary Worksheet, or
- Continuous Liquid Waste Analysis Summary Worksheet

NOTE: Normally, data from several release permits may be summarized and recorded in a Batch or Continuous Liquid Waste Analysis Summary Worksheet column However, data applicable to an abnormal release should be recorded in a separate column.

6.5.3 In a separate column of Batch or Continuous Liquid Waste Analysis Summary Worksheet, record the following:

- Release date or dates
- Worksheet ID Number for Abnormal Liquid Release to Unrestricted Area Worksheet containing data being used,
- μCi released per nuclide,
- Total μCi released, and
- Volume (liters) of liquid released.

NOTE: The Supervisor RA&MC or designee will determine if tritium, alpha radioactivity, Sr-89, Sr-90 and Fe-55 released in the abnormal release are to be accounted for by including a portion of sample into the monthly and/or quarterly composites, or if separate analysis data on the abnormal release sample are to be used.

6.5.4 If a separate alpha activity, Sr-89, and Sr-90 release determination will be used for accountability, then obtain the separate saniple analysis. Include the results on the worksheet used to summarize composite data for transferring to Liquid Waste Quarterly Release Summary.

6.5.5 Ensure copies of the applicable Abnormal Liquid Release to Unrestricted Area Worksheets are available (e.g., attached to Summary Worksheet) for later reference, if needed.

7.0 FOLLOW-ON

7.1 Follow-On Tasks

7.1.1 If either of the following was prepared and is to become a record for filing:

- Abnormal Liquid Release to Unrestricted Area Worksheet,
- EPA RQ Screening Worksheet for Liquid Release, or
- Abnormal Liquid Release - Liquid Concentration Worksheet,

Then as required, have an individual in Operations (e.g., Shift Supervisor) review the worksheet and record Station Operations Signature and Date.

7.1.2 If an EPA R Q Screening Worksheet for Liquid Release, was completed, then perfom or ensure performed, a follow-up evaluation of the release with respect to 40 CFR 302 under the direction of the Supervisor HP Technical Services. Forward results to the Superintendent Radiological Protection for any further action.

7.1.3 As applicable, forward the following completed forms, or equivalent, to the Supervisor RA&MC €or review and determination of suitability for inclusion in Annual Effluent Release Report.

- Abnormal Liquid Release to Unrestricted Area Worksheet
- EPA RQ Screening Worksheet for Liquid Release
- Abnormal Liquid Release - Liquid Concentration Worksheet

7.1.4 If the release is to be included in the Annual Effluent Release Report as a unplanned release, as defied in VPAP-2103N, the Supervisor RA&MC shall forward a copy of the release report to SNSOC with appropriate cover sheet. (Ref. 2.2.4)

7.1.5 Evaluate the release in accordance with HP-1032.010, Radiologicai Survey Records, for applicability of specific records being classified as decommissioning records.

7.2 Records Disposition

As applicable, forward the following records to Records Management in accordance with VPAP-1701, Records Management.

- Abnonnal Liquid Release to Unrestricted Area Worksheet
- EPA RQ Screening Worksheet for Liquid Release
- Abnormal Liquid Release - Liquid Concentration Worksheet

ATTACHMENT 1
(Page 1 of 1)
ABNORMAL LIQUID RELEASE TO UNRESTRICTED AREA WORKSHEET

Part 1 - RELEASE DESCRIPTION
(release data for systems to be obtained from and/or verified by Operations)

Worksheet ID Number

Release Start Date and Time	Release Stop Date and Time	Release Duration, Minutes

Source of Release:
☐ Unit 1 RWST ☐ Unit 1 CCT ☐ Unit 1 CST ☐ Unit 1 SGBDV ☐ PGWST A ☐ Clarifier
☐ Unit 2 RWST ☐ Unit 2 CCT ☐ Unit 2 CST ☐ Unit 2 SGBDV ☐ PGWST B ☐ ____________

Peak Waste Stream Flow Rate, GPM	Volume of Waste Released, Gallons	Volume of Waste Released, mL
________ GPM ☐ Estimated ☐ 6,000 GPM Default		

Number of Circ. Water Pumps Running	Dilution Water Flow Rate, GPM (No. of Circ. Water Pumps Running x 218,000)

Part 2 - %TECH. SPEC. AND ACTIVITY RELEASED DETERMINATION

Nuclide	A Source µCi/mL	B ACW µCi/mL*	C = A / B Source ACW Fraction	D mL Released	E = A x D µCi Released	Remarks
H-3		1E-2				
Na-24		5E-4				
Cr-51		5E-3				
Mn-54		3E-4				
Fe-59		1E-4				
Co-58		2E-4				
Co-60		3E-5				
Ag-110m		6E-5				
I-131		1E-5				
I-132		1E-3				
I-133		7E-5				
I-135		3E-4				
Cs-134		9E-6				
Cs-137		1E-5				
Kr-85m		2E-4				
Xe-133		2E-4				
Xe-133m		2E-4				
Xe-135		2E-4				
	Total Source ACW Fraction					

* Non-noble gas ACWs are 10 X ECWs from 10 CFR 20 Appendix B, Table 2, Col. 2. Noble gas ACWs are from ODCM.

Maximum Allowable Flow Rate, GPM = $\frac{\text{Dilution Water Flow Rate, GPM}}{\text{Sum of Source ACW Fractions}}$ = $\frac{\text{\_\_\_\_\_\_\_\_ GPM}}{\text{\_\_\_\_\_\_\_\_}}$ = ____________ GPM

Source Maximum % Tech. Spec. = $\frac{\cancel{100\ x}\ \text{Waste Peak Flow Rate, GPM}}{\text{Maximum Allowable Flow Rate, GPM}}$ = $\frac{100\ x\ \text{\_\_\_\_\_\_\_\_ GPM}}{\text{\_\_\_\_\_\_\_\_ GPM}}$ = ____________ % Tech Spec.

Total % Tech. Spec. = __________ % Tech. Spec. other releases + Source Maximum % Tech. Spec. = ________ Total % Tech Spec.

Prepared By (Printed Name)	Prepared By (Signature)	Date	Supervisor RA&MC (Signature)	Date

n Operations Review (Signature)	Date	Data to Release Records By (Printed Name and Initials)	Date

ATTACHMENT 2
(Page 1 of 1)
EPA RQ SCREENING WORKSHEET FOR LIQUID RELEASE

RELEASE DESCRIPTION AND NUCLIDE RELEASE DATA (data normally obtained from associated Abnormal Liquid Release to Unrestricted Area Worksheet)		Worksheet ID Number
Release Start Date and Time	Release Stop Date and Time	Duration, Hours and Minutes
Data obtained from Abnormal Liquid Release to Unrestricted Area Worksheet. Worksheet ID Number ____________		

COMPARISON OF ACTUAL RELEASES OVER 24 HOUR TO EPA RQ VALUES

Nuclide	A Total Activity Released Ci	B EPA RQ Value Ci	C = A / B Release RQ Fraction*	Remarks
H-3		1.0E+2		
Na-24		1.0E+1		
Cr-51		1.0E+3		
Mn-54		1.0E+1		
Fe-59		1.0E+1		
Co-58		1.0E+1		
Co-60		1.0E+1		
[illegible]-124		1.0E+1		
[illegible]125		1.0E+1		
1-131		1.0E-2		
I-133		1.0E-1		
Cs-134		1.0E+0		
Cs-137		1.0E+0		
Kr-85m		1.0E+2		
Xe-133m		1.0E+3		
Xe-133		1.0E+3		
Xe-135		1.0E+2		
Totals		N/A		

*IF Release RQ Fraction is greater than 1.0, THEN the HPSS shall notify the Operations Shift **Supervisor** that a **40 CFR 302** notification is required.

Remarks ____

Prepared By (Printed Name)	Prepared By (Signature)	Date and Time	
Shift Supervisor HP Review (Signature.)	Date and Time	Supervisor RA&MC (Signature)	Date
ɔn Operations (Signature)		Bate and Time	

ATTACHMENT 3
(Page 1 of 1)

ABNORMAL LIQUID RELEASE - TION WORKSHEET

Worksheet Use: Evaluate Concentration In en Unrestricted Area Averaged Over Hour	Worksheet ID Number

Release Description

Remarks/Assumptions

1 Hour Release Start Date & Time	1 Hour Release Stop Date & Time
Waste Volume Released in 1 Hour Period, Gallons	Total Diluted Waste Volume in 1 Hour Period, Gallons
Dilution Water Volume Available for Waste in 1 Hr. Period, Gallons	Total Diluted Waste Volume in 1 Hour Period, mL (Gallons x 3785)

Nuclide	Activity Released In 1 Hour, µCi	Unrestricted µCi/mL*	ECW µCi/mL**	ECW Fraction	Remarks
Na-24			5E-5		
Cr-51			5E-4		
Mn-54			3E-5		
Fe-59			1E-5		
Co-58			2E-5		
Co-60			3E-6		
Ag-110m			6E-6		
I-131			1E-6		
I-132			1E-4		
I-133			7E-6		
I-135			3E-5		
Cs-134			9E-7		
Cs-137			1E-6		
Totals	N/A	N/A	N/A		

* Unrestricted µCi/mL = $\frac{\text{Ci Released in 1 hour}}{\text{Total Diluted Waste Volume in 1 Hour Period, mL}}$

** ECWs (Effluent Concentration - Water) are from 10 CFR 20 Appendix B, Table 2, Col. 2.

NRC Notification Requirements (see 10 CFR 50.72) (based on 1 hour averag tration i tricted)

Total ECW Fraction(√)

☐ > 20 - Immediate NRC Notification required by 10 CFR 50.72(b)(2)(iv)(B)

☐ ≤ 20 - NRC Notification not required by 10 CFR 50.72(b)(2)(iv)(B)

Prepared By (Printed Name)	Prepared By (Signature)	Date

Supervisor RA&MC (Signature)	Date	Station Operations Review (Signature)	Date

QUESTIONS REPORT
for sroquestions

G2.4.27 001

Unit 1 has tripped. The foilowing conditions apply:

- Main feedwater has been lost.
- There is a fire burning in the Motor- Driven Auxiliary Feedwater Pump Room.
- The crew has entered 1-FCA-6, " Motor-Driven Auxiliary Feedwater Pump Room Fire. "

This procedure will direct the crew to feed ____.

A✓ the "A" Steam generator by locally throttling the Turbine-Driven AFW Pump manual handwheel on the governor
B. all three steam generators by locally throttling the Turbine-Driven AFW Pump manual handwheel on the governor
C. the " A Steam Generator by locally throttling 1-FW-MOV-100D
D. all three steam generators by locally throttling 1-FW-MOV-100D

A. This is the correct answer. Feed the " A Steam Generator by locally throttling the handwheel on the governor valve.

B. This answer is incorrect. All three steam generators are not to be fed until the fire is out. If examinee doesn't make this distinction, they may pick this answer. Normally feeding all three generators is desired. This *is* one of the few exceptions because the valves are inaccessable.

C. This answer is incorrect. Feed is usually controlled by 1-FW-MOV-100D but procedurally it directs you to use the governor valve due to the fire.

D. This answer is incorrect. All three generators are not to be fed until the fire is out because the valves are inaccessable due to the fire.

Emergency Procedures/Plan
Knowledge of fire in the plant procedure

References: 1-FCA-6 Fire In The Motor-Driven Auxiliary Feed Pump Room

Level(RO/SRO):	SRO	Tier:	3
Group:		Importance Rating:	3/3.5
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

Assume the following conditions.

- A fire exists in the unit-1 motor-driven auxiliary feedwater pump house
- The operating crew is responding per the applicable fire contingency action procedure
- An operator has been dispatched to control auxiliary feedwater locally via the Terry turbine

To control feedwater flow locally, the operator __________ and monitors the emergency storage tank level by __________

A. adjusts the governor valve; observing the suction pressure indicator for 1-FW-P-2

B. throttles 1-FW-PCV-159A and 1598; observing the auxiliary shutdown panel indicator

C. adjusts the governor valve; observing the auxiliary shutdown panel indicator

D. throttles 1-FW-PCV-159A and 159B; observing the suction pressure indicator for 1-FW-P-2

Answer: A

VIRGINIA POWER
NORTH ANNA BOWER STATION
FIRE CONTINGENCY ACTION

NUMBER	PROCEDURE TITLE	REVISION
1-FCA-6	MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP ROOM FIRE (WITH TWO ATTACHMENTS)	2
		PAGE 1 of 6

PURPOSE

Provide the necessary guidelines for recovery of Unit I in the event of a fire in the Motor-Driven Auxiliary Feedwater Pump House.

ENTRY CONDITIONS

- 0-FCA-0, FIRE PROTECTION - OPERATIONS RESPONSE

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL: APPROVAL - ON PILE	DATE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	MONITOR PLANT CONDITIONS	
	a) Verify Feedwater Flow - ADEQUATE • Reactor - MODE 5, MODE 6. OR DEFUELED OR • SG levels maintained by Main Feedwater	a) Perform the following: 1) Place Turbine Driven AFW Pump Steam Supply Valve controllers to OPEN: • 1-MS TV-111A • 1-MS-TV 111B • 1-FW-P-2, Turbine-Driven AFW Pump, is running 2) Ensure 1-FW-P-2. Turbine-Driven AFW Pump. is running 3) Open 1-FW-MOV 100D. 4) Verify at least one of the following conditions exists: • Verify AFW flow to "A" SG indicated on 1-FW FI-100A, SG A AFW Flow is GREATER THAN 400 gpm. OR • "A" SG Narrow Range Level - BETWEEN 23% AND 50% 5) Locally control AFW flow in Turbine-Driven AFW Pump Room Attachment 2, LOCAL CONTROL OF 1-FW P 2.
	b) Main Feedwater - IN SERVICE	b) Ensure one of the following. as applicable: • RHR system is in service OR • i-E 0, REACTOR TRIP OR SAFETY INJECTION. or another EOP branched from 1 E-0 is in progress OR • Reactor is defueled and decay heat removal is NOT required.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2.___	MONITOR ECST LEVEL:	
	a) ECST Level - UNAFFECTED BY FIRE: • CN-LI-100A, ECST LEVEL • CN-LI-100B-1, ECST LEVEL	a) Ensure an Operator does the following: 1) Qbtain radic and emergency lantern 2) Locally monitor ECST Level by 1-FW PI 156A, Turbine Driven AFW Pump Suction Press Indicator (in Turbine Driven AFW Pump Room)
	b) Check ECST Level - ADEQUATE: • ECST Level - GREATER THAN 40% OR • AFW Pump Suction pressure GREATER THAN 4 PSIG	b) Initiate 1-AP-22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1
3.___	INITIATE EPIP-1.01. EMERGENCY MANAGER CONTROLLING PROCEDURE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4.__ RACK OUT BOTK MOTOR-DRIVEN AFW PUMPS

a) Place both Motor-Driven AFW Pump selector switches - PULL TO LOCK

- 1-FW-P-3A
- 1-FW-P-3B

b) Locally rack out both Motor-Driven AFW Feed Pump breakers using 0 OP-26.9. 4160 VOLT BREAKER OPERATION. (Unit 1 Emergency Switchgear Room)

AFW PUMP	BREAKER
1-FW-P-3A	1-EP-BKR-15H3
1-FW-P-3B	1-EP-BKR-1553

5.__ LOCALLY OPEN THE FOLLOWING BREAKERS TO DE-ENERGIZE MOTOR-DRIVEN AFW EQUIPMENT

BREAKER	LOCATION	EQUIPMENT
1-EP-MCC-1J1-2N P4	Cable Vault:	1-FW-MOV 100A, AFW to "A" SG
1 EP MCC 1J1 2N G4	Cable Vault	1-FW MOV 100B, AFW to "B" SG
1-EP-MCC-1J1-2N H4	Cable Vault	1-FW-MOV-100C. AFW to "C" SG
1 EP MCC 1J1-2N 54	Cable Vault	-HV-F-70B, Exhaust Fan
1 EP MCC 1J1 2S E4	Cable Vault	-FW-MOV-100D, AFW TO "A" SG
1-EP-MCC-1H1-2N H1	Cable Vault	HV-F-70A, Exhaust Fan
1 EP CB-16B Bkr 5	Control Room	Power to 1-EP-CB-98K (MOV Beaters)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. __	VERIFY FEEDWATER PLOW - ADEQUATE • Reactor MODE 5. MODE 6, OR DEFUELED OR • SG levels maintained by Main Feedwater	Perform the following: a) Place Turbine-Driven AFW Pump Steam Supply Valve controllers to OPEN: • 1-MS-TV-111A • 1-MS-TV-111B • 1-FW-P-2, Turbine-Driven AFW Pump. is running b) Ensure 1-FW-P-2. Turbine-Driven AFW Pump. is running c) Open 1-FW-MOV-100D. d) Verify at least one of the following conditions exists: • Verify AFW flow to "A" SG indicated on 1-FW-FI 100A, SG A AFW Flow is GREATER THAN 400 gpm. OR • "A" SG Narrow Range level BETWEEN 23% AND 50% e) Locally control AFW flow in Turbine-Driven AFW Pump Room Attachment 2. LOCAL CONTROL OF 1 FW-P-2.
7. __	VERIFY FIRE - EXTINGUISHED	RETURN TO Step 6.
8. __	ALIGN 1-FW-P-2 TO ALL SGS USING 1 AP 22.4, LOSS OF BOTH MOTOR DRIVEN AFW PUMPS	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9.___ INITIATE CORRECTIVE ACTIONS TO RESTORE EQUIPMENT IN MOTOR DRIVEN AFW PUMP ROOM TO AVAILABLE CONDITION

- 1-FW-P-3A
- 1 FW-P-3B

- 1-FW-MOV-IOOA
- 1-FW-MOV-100B
- 1 FW MOV 100C
- 1-FW-MOV-100D

- 1-FW-HCV-100A
- 1-FW-IICV lOOB
- 1-FW-HCV 100C

- 1-FW-PCV-159A
- 1-FW-PCV-159B

- 1-HV-F-SOA
- 1-HV-F-70B

10.___ CONSULT WITH TSC OR PLANT STAFF TO DETERMINE LONG TERM PLANT STATUS

END -

NUMBER 1-FCA-6	ATTACHMENT TITLE	REVISION 2
ATTACHMENT 1	REFERENCES	PAGE 1 of 1

- 0-FCA-0, FIRE PROTECTION - OPERATIONS RESPONSE
- i E 0, REACTOR TRIP OR SAFETY INJECTION
- 1-AP-22.4. LOSS OF BOTH MOTOR-DRIVEN AFW PUMPS
- 1 AP-22.5, LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK 1
- i OP 26.9. 4160 VOLT BREAKER OPERATION
- EPIP 1.01, EMERGENCY MANAGER CONTROLLING PROCEDURE
- OF-317. Incorporate monitoring ECST level via AFW suction pressure gages (see E.T. CEE 94 047, Rev. 1) in FCA-6 and FCA-11

NUMBER 1-FCA-6	ATTACHMENT TITLE	REVISION 2
ATTACHMENT 2	LOCAL CONTROL OF 1-FW-P-2	PAGE 1 of 1

NOTE: Personnel involved in performance of this Attachment should remain in the Turbine-Driven Auxiliary Feedwater Bump House until released by the Control Room Operator.

1. ___Obtain the following items:

- Radio
- Emergency Lantern
- Vital Area Key For Unit 1 SFGD. Main Stm. AFWPII

2. ___Proceed immediately to Turbine-Driven Auxiliary Feedwater Pump House.

3. ___Establish communication with Control. Room.

4. ___Monitor ECST Level by monitoring 1-FW-PI-156A, Turbine Driven AFW Pump Suction Press 'Indicator.

5. ___Locally control 1-FW-P-2 output, as directed by Control Room Operator. by adjusting manual handwheel on governor valve.

- END -

1.4. If safe shutdown equipment is affected or an immediate shutdown and cooldown is warranted then the shift supervisor can authorize the continuance of the procedure.

Topic 3.13: FCA-6 Information
13909

3.13a. Objective

Explain the following concepts associated with responding to a fire **in** the motor-driven auxiliary feedwater pump house in accordance with **1-FCA-6.**

- How auxiliary feedwater flow is controlled
- How level in the emergency condensate storage tank can be monitored

3.13b. Content

1. Auxiliary feedwater flow to the " A steam generator is accomplished **by** dispatching an operator to manually adjust the turbine driven **AFW** pump's governor valve; lowering turbine speed to reduce flow, raising turbine speed to increase flow.
   1.1. Once the fire is extinguished the turbine driven AFW pump is aligned to feed "B" and "C" generators.

2. Level in the emergency condensate storage tank can be monitored from:
   2.1. Control room level indication
   2.2. Auxiliary shutdown panel indication
   2.3. Suction pressure indication for **I-FW-P-2.**

Topic 3.14: FCA-9 Information
13910

3.14a. Objective

WE02EG2.2.25 001

Unit 2 is in Mode 4 with the RCS cold leg temperatures at 265°F. 2-CH-PCV-2145, Letdown Pressure Control Valve fails shut. With no operator action, the WCS will ____ within 110% of the RCS pressure/temperature limits based on ________

A✓ stay; PORV actuation
B. not stay; PORV actuation
C. stay; safeties opening
D. not stay; safeties opening

A. This *is* the correct answer. At this temperature LTOPS **is** required to be in service. T.S. Bases states this will use the PORV to prevent the RCS **from** reaching 110% of pressure/temperature limits.

B. This answer is incorrect. The question **is** within the design **limits** of the system based on T.S. Examinee may know LTOPS needs to be in service but not be clear as **to** bases with no operator action. Operator normally responds quickly to this transient.

C. This answer is incorrect. The safeties are credited with keeping the RCS from reaching design limits when not in LTOPS. This also is the reason **this** distractor is plausible.

D. This answer is incorrect. The RCS will stay within design pressure. **It is** not based on safeties at this pressure but on PORV. This answer is plausible because safeties are in the T.S. Bases and they will not limit the RCS to design specifications but the PORV wilt.

SI Termination
Knowledge of bases in technical specifications for **limiting conditions for operations and safety limits.**

References: T.S. Bases 3.4.12

This is a new question.

Level(RO/SRO):	SRO	**Tier:**	I
Group:	*2*	**Importance Rating:**	2.5/3.7
Type(Bank/Mod/New):	NEW	**Cog(Knowledge/Comp):**	COMPREHENSIVE
Reference(Y/N):	N	*Last* Exam(Y):	N

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one charging pump and one low head safety injection (LHSI) pump capable of injecting into the RCS and the accumulators isolated. with power removed from the isolation valve operators, and one of the following pressure relief capabilities:

a. Two power operated relief valves (PORVs) with lift settings of:

1. ≤ 500 psig (Unit 1), 415 psig (Unit 2) when any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2); and

2. ≤ 395 psig (Unit 1), 375 psig (Unit 2) when any RCS cold leg temperature ≤ 150°F (Unit 1), 130°F (Unit 2).

b. The RCS depressurized and an RCS vent of ≥ 2.07 square inches.

---------- NOTES ----------

1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swapping operations.

2. Accumulator isolation with power removed from the isolation valve operators is only required when accumulator pressure is greater than the PORV lift setting.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Two LHSI pumps capable of injecting into the RCS.	A.l	Initiate action to verify a maximum of one LHSI pump is capable of injecting into the RCS.	Immediately

B 4.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the LTOP System design basis pressure and temperature (P/T) limit curve (i.e., 110% of the isothermal P/T limit curve determined to satisfy the requirements of 10 CFR 50, Appendix G, Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. This specification provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference I requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure. therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by limiting coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one low head safety injection (LHSI) pump and one charging pump incapable of injection into the RCS and isolating the accumulators when accumulator pressure is greater than the PORV lift setting. The pressure relief capacity requires either two redundant RCS PORVs or a depressurized RCS and an

(continued)

BACKGROUND (continued)

RCS vent of sufficient size. One RCS PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With limited coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one LHSI and charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or a depressurized RCS and an RCS vent of sufficient size. Two RCS PORVs are required for redundancy. One RCS PORV has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure exceeds a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition is not acceptable. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is passed to a comparator circuit which determines the pressure limit for that temperature. The pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, the PORVs are signaled to open.

The PORV setpoints are staggered so only one valve opens to stop a low temperature overpressure transient. If the opening of the first valve does not prevent a further increase in pressure. a second valve will open at its higher pressure setpoint to stop the transient. Having the setpoints of both valves within the limits in the LCO ensures that the LTOP System design basis P/T limit curve will not be exceeded in any analyzed event.

(continued)

BACKGROUND

PORV Requirements (continued)

When a PORV is opened in an increasing pressure transient. the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS within the LTOP design basis P/T limit curve in an RCS overpressure transient. if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP ma55 or heat input transient. and maintaining pressure below the LTOP System design basis P/T limit curve. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, blocking open a PORV and its block valve, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the LTOP System design basis P/T limit curve (i.e., 110% of the isothermal P/T limit curve determined to satisfy the requirements of 10 CFR 50, Appendix G, Ref. 1). In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding 235°F (Unit 1), 270°F (Unit 2), the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At 235°F (Unit 1). 270°F (Unit 2) and below, overpressure prevention falls to two OPERABLE RCS PORVs or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The RCS cold Beg temperature below which LTOP protection must be provided increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T curves are revised, the LTOP System must be

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition.

The LCO contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 3 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Reactor coolant pump (RCP) startup with temperature asymmetry between the RCS and steam generators,

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur. which either of the LTOP overpressure protection means cannot handle:

a. Rendering **all** but one LHSI pump and one charging pump incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions when accumulator pressure is greater than the BQRV lift setting; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCC 3.4.6, "RCS Loops—MODE 4," and LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection.

The Reference 3 analyses demonstrate that either one PCRV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one LHSI pump and one charging pump are actuated. Thus, the LCO allows only one LHSI pump and one charging pump OPERABLE during the LTCP MODES. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

Reference 3 analyses do not explicitly model actuation of the LHSI pump, since the RCS pressurization resulting from inadvertent safety injection by a single charging pump against a water-solid RCS would not be made more severe by such actuation. Since the LTOP analyses assume that the accumulators do not cause a mass addition transient, when RCS temperature is low, the LCO also requires the accumulators to be isolated when accumulator pressure is greater than the PORV lift setting. The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at 235°F (Unit 1), 270°F (Unit 2).

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 (Ref. 4), requirements by having a maximum of one LHSI. pump and one charging pump OPERABLE.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits shown in the LCO. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one charging pump injecting into the RCS. These analyses consider pressure overshoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the RCS pressure at the reactor vessel beltline will not exceed the LTOP design P/T limit curve.

The PORV setpoints are evaluated when the P/T limits are modified. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating the allowed LTOP overpressure transient. (A vent size of 2.07 square inches is the equivalent relief capacity of one PORV.) The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration. one LHSI pump and one charging pump OPERABLE. maintaining RCS pressure less than the LTOP design basis P/T limit curve.

The RCS vent size is re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the LTOP System design basis P/T limit curve (i.e., 110% of the isothermal P/T limit curve determined to satisfy the requirements of 10 CFR 50, Appendix G, Ref. 1) as a result of an operational transient.

To limit the coolant input capability, the LCO requires a maximum of one LHSI pump and one charging pump capable of injecting into the RCS and all accumulator discharge isolation valves closed with power removed from the isolation valve operator, when accumulator pressure is greater than the PORV lift setting.

The LCO is modified by two Notes. Note 1 allows two charging pumps to be made capable of injection for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection.

(continued)

BASES

LCO (continued)

Note 2 states that accumulator isolation is only required when the accumulator pressure is more than the PORV lift setting. This Note permits the accumulator discharge isolation valves to be open if the accumulator cannot challenge the LTOP limits.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs; or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limits provided in the LCO and testing proves its ability to open at this setpoint, and backup nitrogen motive power is available to the PORVs and their control circuits.

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 2.07 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 235°F (Unit 1), 270°F (Unit 2), in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 235°F (Unit 1), 270°F (Unit 2). When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 235°F (Unit 1), 270°F (Unit 2).

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A.1 and B.1

With more than one LHSI pump and one charging pump capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

C.1, C.2, D.1, and D.2

An unisolated accumulator requires isolation immediately. Power available to an accumulator isolation valve operator must be removed in one hour. These ACTIONS are modified by a Note which states the Condition only applies if the accumulator pressure is more than the PORV lift setting.

If isolation is needed and cannot be accomplished, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 235°F (Unit 1), 270°F (Unit 2), the LCO is no longer Applicable. Depressurizing the accumulators below the PORV lift setting also exits the Condition.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering judgement indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ 235°F (Unit 1), 270°F (Unit 2), with one RCS PORV inoperable, the RCS PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

ACTIONS (continued)

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 5). Thus, with one of the two RCS PORVs inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair PORV failures without exposure to a lengthy period with only one OPERABLE RCS PORV to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.07 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the unit in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of one LHSI pump and a maximum of one charging pump are verified

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed with power removed from the isolation valve operator.

SR 3.4.12.3 is modified by a Note stating that the verification is only required when accumulator pressure is greater than the PORV lift setting. With accumulator pressure less than the PORV lift setting. the accumulator cannot challenge the LTOP limits and the isolation valves are allowed to be open.

The LHSI pumps and charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.4

The RCS vent of ≥ 2.07 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked.

b. Once every 31 days far a valve that is locked, sealed. or secured in position. A removed pressurizer safety valve or blocked open PORV with its block valve disabled in the open position fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

WE08EA2.2 001

A large break LOCA has occurred on unit I. The crew has just transitioned to 1-E-1, "Loss of Reactor or Secondary Coolant," the RO notes the following:

- RCS temperatures have decreased several hundred degrees in a few minutes.

- The STA indicates a RED path on integrity

The operators transition to 1-FR-P.1, " Response To Imminent Pressurized Thermal Shock Condition."

Which ONE of the following describes the correct operator response and the reason fo this response?

A✓ Complete the first step of FR-P.1 and transition to 1-E-1, " Loss of Reactor or Secondary Coolant." PTS is not a concern.
B. Complete FR-P.1 and then transition to 1-E-1, " Loss of Reactor or Secondary Coolant." FRP's are higher priority procedures.
C. Complete steps in FR-P.l until cold leg recirc criteria is met. Transition to 1-ES-1.3, "Transfer to Cold leg Recirculation." When complete, return to FR-P.l. Transfer to cold leg recirc wilt prevent entry into a higher FRP.
D. Complete the first step of FR-P.l and transition to 1-E-1, " Loss of Reactor or Secondary Coolant." Performing FR-P.1 will result in a delay in transitioning to ES-1.3, "Transfer to Cold Leg Recirculation," when RWST low level is reached.

A. This is the correct answer. The first step of P-1 checks for a large break LOCA. During a large break LOCA you cannot repressurize the RCS. This makes PTS not a concern during this event

B. This answer is incorrect because during a large break LOCA the crew will transition out of P-1 at step 1. Examinee may choose this because the majority of FRP's are implemented in their entirety before transitioning out of them. There are instances where you delay FRP implementation but very few you transition out of without completing.

C. This answer is incorrect because the crew only performs step 1 of P-1 and does not ever come back to it. Examinee may choose this answer because switching to cold leg recirc has a caution that FR's may not be implemented until completion of step 8. This activity if not completed, will cause numerous challenges to fission product barriers.

D. This answer is incorrect. Crew does transition out of P-1 at the first step but this is because it is not a concern during a large break LOCA. The examinee may pick this answer because switching over to cold leg recirc is imperative to prevent further challenges to other fission product barriers.

Ability to determine and interpret the following as they apply to RCS Overcooling - PTS: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

(CFR: 41.10 / 43.5 / 45.13)

References:
Objective 12656 from study guide on Functional Restoration Procedures
P-1, "Response to Imminent Pressurized Thermal Shock" step 1
ES-1.3, " Transfer to Cold keg Recirculation" note prior to step 1

Question from INPO Exam Bank.

Level(RO/SRO):	SRO	Tier:	1
Group:	2	Importance Rating:	3.5/4.1
Type(Bank/Mod/New):	BANK	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

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Turkey Point 3 Exam Level S

Mark Question | Print Recor | New Sear

Question

Following a large break LOCA, the RCO notes the following:

- RCS temperatures have decreased several hundred degrees in a few minutes.

- ERDADS display indicates a RED path for an imminent PTS condition.

The operators transition to FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIC transition out based on step 1 guidance.

Which ONE of the following describes the basis for the early transition out of FR-P.1?

FR-P.1 actions should not be performed because PTS is not a concern.

FR-P.1 actions will result in a reduction of cooling water flow to the reactor.

FR-P.1 actions try to maximize RCS subcooling which is not possible

FR-P.1 actions will result in a delay in transitioning to ES-1.3 when RWST low level is reached.

Distracter 2:

Distracter 3:

Topic 5.1.3: FR-P.1 Concepts
12656

5.1.3a. Objective

Explain the following concepts associated with **1-FR-P.1,** "Response to Imminent Pressurized Thermal Shock Condition."

- Why, if a large-break LOCA has occurred, I-FR-P.I is not used
- Why terminating cooldown of the Reactor Coolant System is desirable
- Why a reactor coolant pump is started if none is running and the safety injection termination criteria are not met
- Why, if hot-leg temperatures are increasing following termination of safety injection, actions may be necessary in order to stabilize temperature
- Why a one-hour **soak** of Reactor Coolant System temperature may be required
- How to determine which actions from other guidelines may **be** performed during the Reactor Coolant System temperature soak

5.1.3b. Content

1. A check is made for a large break **LOCA,** if criteria are met than an immediate exit is permitted to the procedure and step in effect.

   1.1. This is allowed since there **is** no PTS concern after a large break LOCA (re-pressurization of the RCS cannot occur following a large break LOCA).

   1.2. This prevents the delay of needed recovery action caused by the unnecessary performance of **FR-P.l.**

2. It is important to terminate any **RCS** cooldown in order to limit the thermal stress imposed on the reactor vessel.

   2.1. This reduces the possibility of vessel failure due to pressurized thermal shock

3. In order to mix the cold incoming **SI** water and the warm reactor coolant, an RCP restart should be attempted.

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Topic 5.1.3: FR-P.1 Concepts
12656

5.1.3a. Objective

Explain the following concepts associated with 1-FR-P.1, "Response to Imminent Pressurized Thermal Shock Condiion."

- Why. if a large-break LOCA has occurred, 1-FR-P.1 is not used
- Why terminating cooldown of the Reactor Coolant System *is* desirable
- Why a reactor coolant pump is started if none **is** running and the safety injection termination criteria are not met
- Why, if hot-leg temperatures are increasing following termination of safety injection, actions may be necessary in order to stabilize temperature
- Why **a** one-hour **soak** of Reactor Coolant System temperature may be required
- **How** to determine which actions from other guidelines may be performed during the Reactor Coolant System temperature soak

5.1.3b. Content

1. A check is made for a large break LQCA. if criteria are met than an immediate exit is permitted to the procedure and step in effect.
   1.1. This is allowed since there is no PTS concern after a large break LOCA (re-pressurization of the RCS cannot occur following a large break LBCA).
   1.2. This prevents the delay of needed recovery action caused by the unnecessary performance of FR-P.1.

2. It is important to terminate **any** RCS cooldown in order to limit the thermal **stress** imposed on the reactor vessel.
   2.1. This reduces the possibility of vessel failure due to pressurized thermal shock.

3. In order to mix the cold incoming **SI** water and the warm reactor coolant, an RCP restart should **be** attempted.

3.1. Following SI actuation. RCS conditions may be restored to within acceptable limits for SI termination to be allowed.

3.2. The combination of a minimum subcooling and sufficient liquid levei in the vessel to cover the core represents less restrictive SI termination criteria than those present in the Emergency Procedures.

3.2.1. For an imminent PTS condition, SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

3.3. The subcooling criterion will ensure sub-cooled conditions and the RVLIS indication ensures the existence of an adequate vessel inventory such that core cooling is ensured.

3.4. If either of the termination criteria are not satisfied. then SI is required to ensure core cooling and should NOT be terminated.

3.5. Most likely the cold leg/downcomer low temperature condition is due to cold incoming SI water.

4. If a secondary break had occurred eariier, the RCS could heat up (depending upon decay heat level and AFW flow rates) after the S/G dries out if ne additional operator action is taken.

4.1. Any heatup will result in an increase in PRZR level that will re-pressurize the RCS.

4.2. Stabilizing RCS temperature by controlling feed flow and steam dump, will prevent this re-pressurization.

5. If RCS cold leg temperature has decreased more than 100°F in any one-hour period, then a "soak" period is required to allow the thermal stresses imposed on the reactor vessel wall to decrease before further cooldown is allowed.

5.1. The "soak" is a period of steady state operation during which any temperature decrease or pressure increase are to be avoided.

5.2. This time period allows thermal gradients in the reactor vessel wall tu be reduced, thus reducing corresponding stresses.

6. Any actions that will not cause either a RCS cooldown or RCS pressure increase and are specified by any other procedure in effect are permitted during this "soak" period.

Cooldown Without CRDM Fans." RVLIS upper head indication **is** 87%. Continued cooldown and depressurization is required. The Shift Manager decides to cooldown **at** 30°/hr. The crew will __________

A! transition to 1-ES-0.3, "Natural Circulation Cooldown With Steam Void In Vessel" and continue to cooldown and depressurize simultaneously
B. remain in 1-ES-0.2B, "Natural Circulation Cooldown Without CRDM Fans" and increase charging to hold RCS pressure constant. This will prevent further void growth
C. transition to ***1-ES-0.3,*** "Natural Circulation Cooldown With Steam Void In Vessel." Cooldown and depressurization will be in a step wise fashion to monitor void growth
D. remain in 1-ES-0.2B, "Natural Circulation Cooldown Without CRDM Fans" and depressurize allowing the head region to refill.

A. This is the correct answer. The RNO for step 11 of ES 0.2B sends you to ES 0.3 or **0.4** if cooldown and depressurization must continue. Caution also states anytime a cooldown must occur at a rate that will form a steam bubble, transition to ES 0.3 **or 0.4** is required.

B. This answer is incorrect. Step 11 of ES 0.28 does give you the option of staying in this procedure if further depressurization is not required; however it instructs the operator to increase pressure **to** collapse voids. It does not give guidance to hold pressure constant to limit void growth.

C. Phis answer is incorrect. Transition to ES 0.3 is correct. In this procedure RVLIS is available. This allows for simultaneous cooldown and depressurization. The examinee may get this confused because ES 0.4 does call for a step wise fashion of reducing pressure and temperature to monitor void growth.

D. Phis answer is incorrect. The crew cannot remain in ES-0.2. It does give guidance to raise pressure to collapse voids. The examinee may choose this answer because numerous times throughout the **EOPs** the operator depressurizes the RCS to refill the Pressurizer. Examinee may mistakenly use this method to refill the Head.

Natural Circ. With Steam Void in Vessel
Ability *to* recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

References: 1-ES-0.2B Caution at beginning of procedure.
1-ES-0.2B Step 11
1-ES-0.3 Cooldown and depressurization sequence.
1-ES-0.4 Cooldown and depressurization sequence

Phis is a new question.

Level(RO/SRO):	SRO	**Tier:**	**1**
Group:	**2**	**Importance Rating:**	**414.3**
Type(Bank/Mod/New):	**NEW**	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8.__ MONITOR RCS COOLDOWN:

a) Core Exit TCs - DECREASING

b) Hot Leg temperatures - DECREASING

c) RCS subcooling based on Core Exit TCs - INCREASING

NOTE: If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that would form a steam void in the Upper Head. then **1-ES-0.3.** NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), or 1-ES 0.4, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS). should **be** used.

9.__ INITIATE! RCS DEPRESSURIZATION:

a) Verify RCS subcooling based on Core Exit TCs GREATER THAN 175°F

a) RETURN TO Step 8.

b) Maintain RCS subcooling based on Core Exit TCs - GREATER THAN 175°F

c) Check letdown - IN SERVICE

c) Try to establish normal letdown.

IF letdown cannot be established. <u>THEN</u> do the following:

1) Depressurize the RCS using one PRZR PORV.

2) GO TO Step IO.

d) Use Auxiliary Spray

e) Cycle PRZR Heaters **as** required

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. ___	CONTINUE RCS COOLDOWN AND DEPRESSURIZATION:	
	a) Maintain cooldown rate in Cold Legs - LESS THAN 15°F/HR	
	b) Kaintain RCS subcooling based on Core Exit TCs - GREATER TIIAN 175°F	b) Stop depressurization and reestablish subcooling.
	c) Maintain RCS temperature and pressure - WITIIIN LIMITS OF ATTACHMENT 1	
	d) Maintain seal injection flow to each RCP between 6 gpm and 8 gpm	

NOTE: If RVLIS indication is not available. then PRLR level response is used to detect voids in the RCS.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. —	CHECK FOR FULL REACTOR VESSEL:	Repressurize RCS within limits of Attachment 1 to collapse potential voids in RCS and continue cooldown.
	• Check PRLR level response AS EXPECTED BASED ON THE FOLLOWING:	IF RCS depressurization must continue. THEN GO TO one of the following:
	• Charging flow	• 1-ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS). STEP 1
	• Letdown flow	**OR**
	• RCS temperature	• 1-ES-0.4. NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), STEP I
	• RVLIS upper range indication GREATER THAN 95%	

VIRGINIA POWER
NORTH ANNA POWER STATION
EMERGENCY PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-ES-0.3	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)	11
		PAGE
	(WITH FIVE ATTACHMENTS)	1 of 10

PURPOSE

To provide instructions to continue plant cooldown and depressurization to Cold Shutdown with no accident in progress. Instructions are provided for conditions that allow for the potential formation of a void in the Upper Head region.

ENTRY CONDITIONS

This procedure is entered from:

- 1-ES-0.2A, NATURAL CIRCULATION COOLDOWN WITH CRDM FANS, or
- 1-ES-0.2B, NATURAL CIRCULATION COOLDOWN WITHOUT CRDM FANS.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL: APPROVAL ON FILE	DATE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Either the first 12 Steps of 1-ES-0.2A NATURAL CIRCULATION COOLDOWN WITH CRDM PANS. or the first *8* Steps of 1-ES-0.2B, NATURAL CIRCULATION COOLDOWN WITHOUT CRDM FANS. should be performed before continuing with this procedure.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.__	TRY TO START ONE RCP:	
	a) Establish conditions for starting one RCP using 1-OP-5.2. REACTOR COOLANT PUMP STARTUP AND SHUTDOWN	a) **WHEN** conditions are established for starting one RCP. **THEN** do Steps 1b through 1d. Continue with Step 2.
	b) Check RVLIS upper range indication - GREATER THAN *95%*	b) Do the following: 1) Raise PRZR level to *69%* using Charging and Letdown.

(STEP 1 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.	TRY TO **START** ONE KCP (Continued):	
		2) Establish subcooling based on Core Exit TCs greater than 50°F using either of the following: • Condenser Steam Dumps OR • SG PORVs OR • Decay Heat Release Valve: a. Locally open 1-MS-20, Decay Heat Release Valve Upstream Isolation Valve. b. Manually open 1-MS-HCV-104. Decay Heat Release Valve.
		3) Energize PRZR Heaters as required to saturate the Pressurizer.
	c) Start one RCP using 1-OF-5.2. REACTOR COOLANT PUMP STARTUP AND SHUTDOWN	c) Continue with Step 2.
	d) GO TO 1-OP-3.2. UNIT SHUTDOWN FROM MODE 3 TO MODE 4	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Saturated conditions in the PRZR should be established before PRZR level is reduced.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. ___	ESTABLISH PRZR LEVEL TO ACCOMMODATE VOID GROWTH:	
	a) Check PRZR level - BETWEEN 36% AND 46%	a) Adjust Charging and Letdown as required.
	b) Put controller for 1-CH-FCV-1122. Normal Charging Flow Control Valve. in MANUAL	
	c) Maintain stable PRZR level using Charging and Letdown	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	CONTINUE RCS COOLDOWN AND INITIATE DEPRESSURIZATION:	
	a) Maintain cooldown rate in Cold Legs - LESS THAN 100°F/HR	
	b) Maintain subcooling based on Core Exit TCs - GREATER THAN 45°F	
	c) Maintain RCS temperature and pressure - WITHIN LIMITS OF ATTACHMENT 1	
	d) Check Letdown - IN SERVICE	d) Depressurize the RCS using one PRZR PQRV. GO TO Step 4.
	e) Depressurize RCS using Auxiliary Spray as follows:	
	1) Verify both PRZR Spray Valves - CLOSED: • 1-RC-PCV-1455A • 1-RC-PCV-1455B	1) Close both PRZR Spray Valves.
	2) Fully open one PRZR Spray valve: • 1-RC-PCV-1455A • 1-RC-PCV-1455B	
	3) Open 1-CH-HCV-1311, Auxiliary Spray Isolation Valve	
	4) Close 1-CH-HCV-1310, Charging Line Isolation Valve	
	5) Throttle the PRZR Spray Valve opened in Step 3e2 to control Auxiliary Spray flow	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.__	CONTROL PRZR LEVEL:	
	a) PRZR level - GREATER THAN 36%	a) Adjust Charging and Letdown, as required, to increase PRZR level to greater than 36%.
	b) PRZR level - LESS THAN 90%	b) Do the following: 1) Energize PRZR Heaters to maintain PRZR pressure stable. 2) Reduce PRZR level to less than 90% using either of the following: • Control Charging and Letdown. as required. OR • Continue cooldown to shrink RCS.
5.__	CHECK RVLIS FULL RANGE INDICATION GREATER THAN 82%	Repressurize RCS to maintain RVLIS Full range greater than 82%. RETURN TO Step 3.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	CHECK IF SI ACCUMULATORS SHOULD BE ISOLATED:	
	a) Check RCS pressure - LESS THAN 1000 PSIG	a) **WHEN** RCS pressure is less than 1000 psig, **THEN** do steps 6b. 7 and 8. Continuo with Step 9.
	b) Isolate SI Accumulators: 1) Initiate Attachment 4 to locally restore power to SI Accumulator Discharge Isolation Valves 2) Close all SI Accumulator Discharge Isolation Valves: • 1-SI MOV-1865A • 1-SI-MOV-1865B • 1-SI-MOV-1865C	b) Vent any SI Accumulator that cannot be isolated using 1-OB-7.3, FILLING. SLUICING. DRAINING, PRESSURIZING. AND VENTING SI ACCUMULATORS. IF any SI Accumulator cannot be vented. **THEN** do the following: 1) Maintain intact SG pressure greater than 120 psig. 2) Consult TSC or Plant Staff.
* 7.___	VERIFY PRESSURE FOR ANY ISOLATED SI ACCUMULATOR REMAINS STABLE AS RCS PRESSURE DECREASES BELOW ACCUMULATOR PRESSURE	Vent auy SI Accumulator that does not maintain stable pressure using 1-OP-7.3. FILLING. SLUICING. DRAINING, PRESSURIZING, AND VENTING SI ACCUMULATORS. **IF** any SI Accumulator cannot be vented. **THEN** do the following: a) Maintain intact SG pressure greater than 120 psig. b) Consult TSC or Plant Staff.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8.__	CHECK IF POWER TO SI ACCUMULATOR MOVS SHOULD DE REMOVED:	
	a) Check RCS mode · IN MODE 4	a) **WHEN** RCS is in Mode 4. **THEN** do Step 8b. Continue with Step 9.
	b) Initiate Attachment 3, Removing Power to SI Accumulator MOVs in Mode 4	
9.__	MAINTAIN Letdown PLOW:	
	a) Open additional Letdown Orifice Isolation Valves. as required	
	b) Adjust 1-CH-PCV 1145. as required	
10.__	MAINTAIN SEAL INJECTION FLOW TO EACH RCP BETWEEN 6 GPH AND 8 GPM	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.__	CHECK IF RHR SYSTEM CAN BE PUT IN SERVICE:	
	a) Check the following RCS conditions: • Hot Leg temperatures LESS THAN 350°F • RCS pressure LESS THAN 400 PSIG	a) RETURN TO Step 3
	b) Consult TSC or Plant Staff to determine required wait period before cooling down on RHR System	
	c) Put RHR System in service using 1-OP-14.1, RESIDUAL HEAT REMOVAL SYSTEM	
12.__	CHECK IF SYSTEMS SHOULD BE ALIGNED FOR NDT PROTECTION:	
	a) Any Cold Leg temperature - WILL BE LESS THAN 235°F WITHIN 1 HOUR	a) WHEN any Cold Leg temperature will be less than 235°F within 1 hour, THEN do Steps 12b, 12c and 12d. Continue with Step 13.
	b) Put one Low-Head SI Pump in PTL	
	c) Put all but one Charging Pump in PTL	
	d) Initiate Attachment 2. NDT Protection to put NDT Protection System in service	
13.__	CONTINUE RCS COOLDOWN TO COLD SHUTDOWN	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

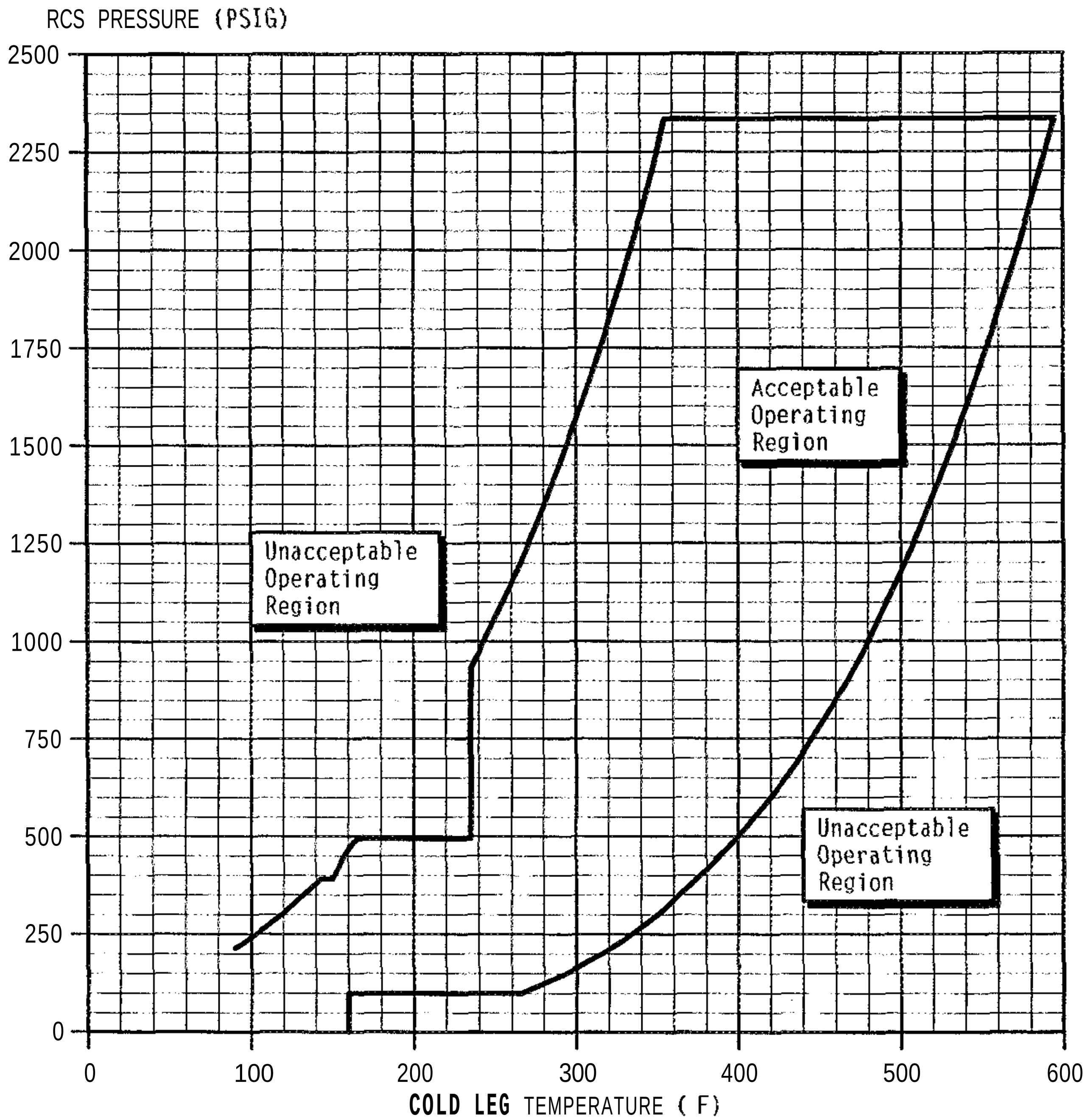
* *

CAUTION: Depressurizing the RCS before the entire system is less than 200°F may result in additional void formation in the RCS.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.___	CONTINUE COOLDOWN OF INACTIVE PORTION OF RCS:	RETURN TO Step 13.
	a) Cool Upper Head Region using CRDM Fans	
	b) Cool SG U-tubes by dumping steam from all SGs	
	c) RVLIS Upper Range - GREATER THAN 95%	
15.___	DETERMINE IF RCS DEPRESSURIZATION IS PERMITTED:	
	a) Check RCS - LESS THAN 200°F: • Core Exit TCs • Hot Legs • Cold legs	a) RETURN TO Step 13.
	b) SGs - ALL COMPLETELY DEPRESSURIZED	b) RETURN TO Step 13.
	c) GO TO 1-OP-3.4, UNIT SHUTDOWN PROM COLD SHUTDOWN (MODE 5) AT 200°F OR LESS TO COLD SHUTDOWN (MODE 5) AT 140°F OR LESS WITH KEFF OF 0.95 OR LESS	

END



RCS PRESSURE (PSIG)
2500
2250
2000
1750
1500
1250
1000
750
500
250
0
0
100
200
300
400
500
600
COLD LEG TEMPERATURE (F)
Acceptable Operating Region
Unacceptable Operating Region
Unacceptable Operating Region

NOTE: The following RCS PORV NDT setpoints apply:

	Tc 150°F to 235°F	Tc < 150°F
1-RC-PCV-1455C	500 psig	395 psig
1-RC-PCV-1456	495 psig	390 psig

1. Before the lowest RCS Cold Leg temperature reaches 235°F, do the following:

___ a. Reduce RCS pressure to 480 psig or less.

___ b. Close 1-RC-MOV-1535, PRZR PORV Block Valve.

___ c. Using the key switch. open 1-RC-PCV-1456.

___ d. Close 1-RC-PCV-1456. PRZR PORV.

___ e. Open 1-RC-MOV-1535, PRZR PORV Block Valve.

___ f. Close 1-RC-MOV-1536, PRZR PORV Block Valve.

___ g. Using the key switch, open 1-RC-PCV-1455C.

___ h. Close 1-RC-PCV-1455C, PRZR POKV.

___ i. Open 1-RC-MOV-1536, PRZR PORV Block Valve.

2. Put key switches for the following valves in AUTO:

___ • 1-RC-PCV-1455C. PRZR PORV

___ • 1-RC-PCV-1456, PRZR PORV

3. Depressurize SI Accumulators to less than 360 psig using 1 OP-7.3. **FILLING.** SLUICING. DRAINING. PRESSURIZING. AND VENTING SI ACCUMULATORS. while continuing with this procedure:

___ • 1-SI TK-1A, A Accumulator

___ • 1-SI TK-1B, B Accumulator

___ • 1 SI-TK-1C, C Accumulator

1. ___ Verify with the Control Room that the Unit is in Mode 4.

2. Do the following to the breakers for the SI Accumulator Discharge Isolation Valves:

   a. Verify with the Control Room that each SI Accumulator is either isolated or vented:

      ___ • 1-SI-TK-1A, "A" SI Accumulator

      ___ • 1-SI-TK-1B, "B" SI Accumulator

      ___ • 1-SI-TK-1C, "C" SI Accumulator

   b. De-energize and lock open the following breakers:

      ___ • 1-EE-BKR-1H1-2N-L3 (1-SI-MOV-1865A) "A" SI Accumulator Discharge.

      ___ • 1-EE-BKR-1H1-2N-N3 (1-SI-MOV-1865B) "B" SI Accumulator Discharge.

      ___ • 1-EE-BKR-1J1-2N-K4 (1-SI-MOV-1865C) "C" SI Accumulator Discharge.

   c. Put Electrical Danger Tags, assigned to the Shift Supervisor, on the following breakers:

      ___ • 1-EE-BKR-1H1-2N-L3 (1-SI-MOV-1865A) "A" SI Accumulator Discharge.

      ___ • 1-EE-BKR-1H1-2N-N3 (1-SI-MOV-1865B) "B" SI Accumulator Discharge.

      ___ • 1-EE-BKR-1J1-2N-K4 (1-SI-MOV-1865C) "C" SI Accumulator Discharge.

1.___ Get an Admin key.

2. Unlock and close the following breakers to restore power to the SI Accumulator Discharge Isolation Valves:

___ • 1-EE-BKR-1H1-2N-L3 (1-SI-MOV-1865A) "A" SI Accumulator Discharge.

___ • 1-EE-BKR-1H1-2N-N3 (1-SI-MOV-1865B) "B" SI Accumulator Discharge.

___ • 1-EE-BKR-1J1-2N-K4 (1-SI-MOV-1865C) "C" SI Accumulator Discharge.

3.___ Notify the Control Room that the SI Accumulator Discharge Isolation Valve breakers have been closed.

Continuous Action Page Steps are listed on the back of this page.

Continuous Action Page Steps are listed on the back of this page.

Continuous Action Page Steps are listed on the back of this page.

CONTINUOUS ACTION PAGE FOR 1-ES-0.3

1. SI ACTUATION CRITERIA
IF either condition listed below occurs OR an SI Actuation occurs, THEN manually initiate SI AND GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, STEP 1:

- RCS subcooling based on Core Exit TCs - LESS THAN 25°F, OR

- PRZR level - CANNOT BE MAINTAINED GREATER THAN 11%.

2. RCP START CRITERIA
Following a loss of all seal cooling. affected RCPs should NOT be started without prior status evaluation.

RCPs should be run in the following order of priority to provide PRZR spray: C, A.

WHEN both conditions listed below exist. THEN start one RCP AND GO TO 1-OP-3.2. UNIT SHUTDOWN FROM MODE 3 TO MODE 4:

- Conditions required by 1-OP-5.2, REACTOR COOLANT PUMP STARTUP AND SHUTDOWN are established. AND either

- RVLIS upper range indicates - GREATER THAN 95%. OR

- PRZR level - GREATER THAN 69% AND RCS subcooling - GREATER THAN 50°F

3. ECST LEVEL CRITERIA
WHEN the ECST level decreases to 40%, THEN initiate 1 AP 22.5. LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1.

4. PRZR SPRAY ISOLATION CRITERIA
WHEN an RCP is stopped. THEN isolate PRZR spray from the stopped RCP.

WE11EA2.2 001

Unit 1 has experienced a small break LOCA. Neither low head safety injection pump i available. The crew has transitioned to I-ECA-1.1, " Loss of Emergency Coolant Recirculation." Safety injection cannot be terminated based on a lack of subcooling. Until recirculation capability can be restored, the crew will conserve inventory by:

A. cooling down and depressurizing to meet subcooling requirements. This wiil allow crew to terminate SI.
B. throttling the BIT outlet isolation valves 1-SI-MOV-1867C/D to the minimum flow required per Attachment 2, " Minimum SI Flow Rate Versus Time After Trip."
C. establishing normal charging and then securing the second HHSI pump.
D. The crew doesn't need to conserve inventory. Makeup to the **RWST** will ensure NPSH of the HHSI pumps.

A. This is the correct answer. To conserve inventory SI needs to be terminated allowing normal charging to be restored. To do this, break flow must be reduced by depressurizing the RCS.

B. This answer is incorrect. Attachment 2 in the back of the procedure is used to verify minimum flow is met. This reference determines the ability to only run one charging pump. It is not intended to be used to reduce SI flow.

C. This answer is incorrect. This would be correct but the order is reversed. The crew should stop one HHSI pump and then evaluate SI termination.

D. This answer is incorrect. Makeup to the **WWST** is a major action category for this procedure but is not of sufficient quantity to keep up with WWST depletion.

Ability to determine and interpret the following as they apply to Loss of Emergency Coolant Recirc: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

(CFR: 41.10 / 43.5 145.13)

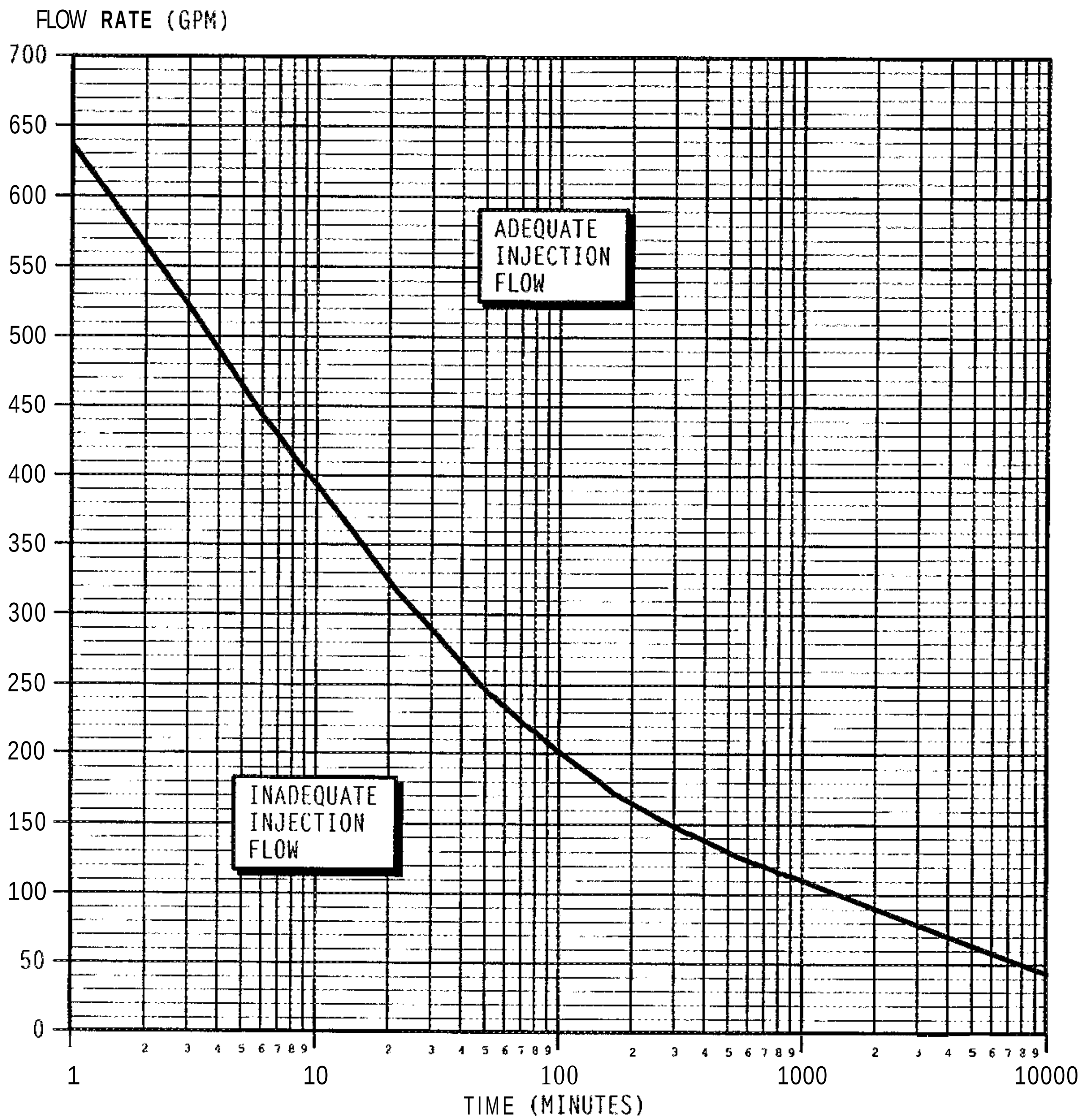
References: 1-ECA-1.1, " Loss of Emergency Coolant Recirculation."

This is modified from INPO bank.

Level(RO/SRO):	**SKO**	**Tier:**	**1**
Group:	1	Importance **Rating:**	3.4/4.2
Type(Bank/Mod/New):	MOD	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	Last Exam(Y):	N

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*15.__	CHECK IF SI CAN BE TERMINATED:	
	a) Check **RVLIS** indication: • No RCPs running FULL RANGE GREATER THAN 67% **OR** • One RCP running - DYNAMIC HEAD GREATER THAN 36%	a) GO TO Step 21
	b) RCS subcooling based on Core Exit TCs GREATER THAN 75°F [125°F]	b) Establish minimum SI flow to remove decay heat. Perform the following: 1) Determine minimum SI flow required using Attachment 2. 2) <u>IF</u> SI flow is less than minimum SI flow. <u>THEN</u> start one additional Charging Pump. 3) GO TO Step 21.
16.__	RESET ISOLATION SIGNALS:	
	a) Reset both Trains of Phase A Isolation	
	b) Reset both Trains of Phase B Isolation. if actuated	

NUMBER 1-ECA-1.1	ATTACHMENT TITLE	REVISION 11
ATTACHMENT 2	**MINIMUM SI FLOW RATE VERSUS TIKE AFTER TRIP**	PAGE 1 of 1



Questions Marked for Collection

..00WE11.A2.2 **12/9/2002**

WEC Cook 1

A LOCA is in progress and both recirculation sump suction valves (ICM 305 and ICM 306) failed to open while transferring to cold leg recirculation. The crew is currently at step 12.b. RNO **of OHP-4023-ECA-1-1, Loss** of Emergency Coolant Recirculation.

This step directs the crew to establish minimum ECCS flow to remove decay heat per Figure 1. This is to he accomplished by manually aligning ECCS pumps and throttling BIT discharge to cold leg valves **as** necessary.

Given the following:

-RWST level is **18%** and lowering.
-East CCP, South SI & West RHR pumps are running.
-RCS Pressure is 340 psig.
-Minimum ECCS Flow Required per Figure 1 is 280 gpm.

Which ONE of the following describes how **this** flow **will he** established?

Shutdown **SI** and RHR Pump aid throttle BIT to **280** gpm of CCP flow.

Shutdown RHR Pumps and throttle BIT to 280 gpm of combined CCP and **SI** pump flow.

Shutdown CCP and **SI** Pumps. RHR pump flow should **be about 280** gpm **at this** pressure

Shutdown SI and RHR Pumps. CCP flow should be about **280** gpm at **this** pressure without throttling the BIT.

Use & modify this question.

Questions Marked for Collection

..00WE11.K2.01 5/5/2003

WEC Salem Unit 1

Which of the following actions are directed by 2-EOP-LOCA-5, LOSS OF EMERGENCY COOLANT RECIRCULATION?

1. Provide guidance on aligning the Safety Injection Pump suction directly to the Containment Sump.
2. Terminate Cold Leg Recirculation and restore Charging and Letdown.
3. Cooldown and depressurize the Reactor Coolant **System to** allow Residual Heat Removal to be put into service.
4. Provide methods to make-up tu the Refueling Water Storage Tank.

2, 3 and 4.

1, 2 and 3.

2 and 4 ONLY

3 and **4** ONLY.

D is correct because it contains all items that LOCA-5 addresses. B and c are incorrect since **they** do not include all actions taken. A is incorrect because the **SI** pump suction cannot he directly aligned to the Containment Sump.

..WE11EA.11.1 3/14/2003

WEC Surry 1

- **A** LOCA has occurred on Unit 1
-The 1H 480 Emergency bus **has** hipped due to a fault on the bus, and cannot be re-energized.
-The Crew has **progressed through** E-I, "Loss of Reactor or Secondary Coolant," the crew is currently at Step 17, INITIATE EVALUATION OF PLAN? STATUS.
-RCS pressure is 1000 psig and slowly Lowering.
-CTMT Radiation monitors are elevated.
-Annunciator 1A-H4 "LHSI PP 1B LOCKOUT OR OL TRIP **has** just illuminated.

Which one of the following describes the correct operator action for the listed cunditiotis?

Transition to ECA-1.1, "**Loss** Of Emergency Coolant Recirculation"

Transition to ES-1.2, "Post LOCA Cooldown and Depressurization"

Transition to ES-1.3, "Transfer Tu Cold Leg Kecirculatiun"

Transition to ECA-1.2, "**LOCA** Outside Containment"

C. **This** is the correct transition with no LHSI pumps available.

A. Incorrect, if cold leg recirc. capability **was** available **this** would he the correct transition

B. Incorrect, if this had been a large break LOCA and RWST level was <16 feet, this would be the correct transition.

D. Incorrect, This would he the correct transition if the radiation alarms were in the aux. bldg.

..00WE11.K2.01 | 5/5/2003

Salem Unit 1 | **Exam Level** R

Mark Question | Print Recor | New Sear

Question

Which of the following actions are directed by 2-EOP-LOCA-5, LOSS OF EMERGENCY COOLANT RECIRCUL

1. Provide guidance on aligning the Safely Injection Pump suction directly to the Containment Sump.
2. Terminate Cold Leg Recirculation and restore Charging and Letdown.
3. Cooldown and depressurize h e Reactor Coolant System to allow Residual Heat Removal to be put into servi
4. Provide methods to make-up to the Refueling Water Storage Tank.

Answer: 2, 3 and 4.

Distracter 1 1, 2 and 3.

Distracter 2 2 and 4 ONLY.

Distracter 3 3 and 4 ONLY.

Distracter Analysis:

Answer: D is correct because it contains all items that LOCA-5 addresses. B and c are incorrect since they do not inclu is incorrect because the SI pump suction cannot be directly aligned to the Containment Sump.

Distracter 1:

Distracter 2:

Distracter 3:

Recirculation." Safety injection cannot be terminated based on a lack of subcooling. Until recirculation capability can be restored, the crew will conserve inventory by:

A. This is the correct answer. Po conserve inventory SI needs to be terminated allowing normal charging to be restored. To do this break flow must be reduced by depressurizing the RCS.

5. Attachment 2 in the back of the procedure is used to verify minimum flow is met. This reference determines the ability to only run one charging pump. It is not intended to be used to reduce SI flow.

C. This would be correct but the order is reversed. The crew should stop one HHSI pump and then evaluate SI termination.

D. Makeup to the RWST is a major action category for this procedure but is not of sufficient quantity to keep up with RWST depletion.

Ability to determine and interpret the following as they apply to Loss of Emergency Coolant Reeirc: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

(CFR: 41.10 143.5 145.13)

References: 1-ECA-1.1, " Loss of Emergency Coolant Recirculation."

This is modified from INPO bank.

Level(RO/SRO):	SRO	Tier:	I
Group:	1	**Importance Rating:**	3.4/4.2
Type(Bank/Mod/New):	MOD	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	N	**Last Exam(Y):**	N

Supervisor declares the valve inoperable and the crew performs the following:

.. . . provided steam dumps are available.

A. This is the correct answer. Per ITS 3.7.1 and its bases, the operator needs to reduce power to less than 52% and reduce the trip setpoint. This is because MTC is positive and a heatup could raise power leading to a power level that the remaining secondary safeties can't handle.

B. This answer is incorrect. It would be correct if not for the positive MTC. This answer takes credit for a negative MTC to turn power during a heatup event. The examinee could pick this if he doesn't take MTC into account when reading ITS.

C. This answer is incorrect. The actions taken in this answer are correct **but** the bases behind them is wrong. MTC is positive. There won't be a restart on a cooldown. The examinee may choose this answer if their understanding on MTC is backwards. Restart accidents on cooldowns are mentioned numerous times throughout training materials. This answer may be picked based on familiarity with the material.

D. This answer is incorrect. The actions taken are not complete without reducing the Hi Flux Trip setpoint. The remaining safeties do have enough capacity to relieve RCS energy with Steam Dumps because they would prevent the RCS from heating up but the accident analysis doesn't take credit for steam dumps. The examinee may pick this answer because steam dumps will prevent RCS temperature from heating up taking MTC out of the equation.

Steam Generator Overpressure
Knowledge of bases in technical specifications for limiting conditions for operations and safety limits

References: ITS 3.7.1 and its bases.

This is a new question.

Level(RO/SRO):	SRO	Tier:	1
Group:	2	Importance Rating:	2.5/3.7
Type(Bank/Mod/New):	NEW	Cog(Knowledge/Comp):	COMPREHENSIVE
Reference(Y/N):	Y	Last Exam(Y):	N

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System. is not available.

Five MSSVs are located on each main steam header. outside containment. upstream of the main steam isolation valves. as described in the UFSAR, Section 10.3.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the capacity of the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events. which are presented in the UFSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is typically the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip. assuming no credit for operation of the atmospheric or condenser steam dump valves. The UFSAR Section 15.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO. The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example. if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE

(continued)

APPLICABLE SAFETY ANALYSES (continued)

MSSVs. Thus, for multiple inopersble MSSVs on the same stem generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When Moderator Temperature Coefficient (MTC) is positive. the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus. for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 902% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure. and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

(continued)