Administrative Topics Outline
Task Summary

Facility:	Fort Calhoun Station	Date of Examination:	Dec / 2015
Examination Level	SRO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations (SA1)	N, R	2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc. (4.2). JPM: Perform an Alternate Decay Heat Removal Method Determination (N).
Conduct of Operations (SA2)	M, R	2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (4.3). JPM: Calculate an Estimated Critical Boron Concentration and Minimum and Maximum Critical Rod Position (AJPM-SRO-RE-1).
Equipment Control (SA3)	N, R	2.2.40 Ability to apply Technical Specifications for a system (4.7). JPM: Determine In-Core Instrumentation Operability (N).
Radiation Control (SA4)	N, R	2.3.7 Ability to approve release permits (3.8). JPM: Authorize a Liquid Waste Release (N).
Emergency Plan (SA5)	M, R	2.4.41 Knowledge of the emergency action level thresholds and classification (4.6). JPM: Classify an Emergency Plan Event (AJPM-SRO-EP-1).

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

*Type Codes & Criteria:
- (C)ontrol room, (S)imulator, or Class(R)oom
- (D)irect from bank (≤ 3 for ROs; ≤ for 4 for SROs & RO retakes)
- (N)ew or (M)odified from bank (≥ 1)
- (P)revious 2 exams (≤ 1; randomly selected)

Administrative Topics Outline
Task Summary

SA1 The applicant will determine an alternate decay heat removal method per AOP-19, Loss of Shutdown Cooling, Attachment D, Alternate Decay Heat Removal Method Determination. Critical steps include determining the decay heat removal method given time after shutdown, Reactor Coolant System pressure boundary condition, available cooling pumps, and flow paths. This is a new JPM.

SA2 The applicant will calculate an Estimated Critical Boron Concentration and Minimum and Maximum Critical Rod Position per TDB-V.1.B, Technical Data Book Estimated Critical Conditions Worksheet and TDB-II, Technical Data Book Reactivity Curves. The critical steps include calculating the Estimated Critical Boron Concentration and Minimum and Maximum Critical Rod Position. This is a modified bank JPM.

SA3 The applicant will determine In-Core Instrumentation operability per OI-NI-2, In-Core Instrumentation Operability Requirements. The critical steps include assessing the condition of the In-Core Instrumentation and applying Technical Specifications as required. This is a new JPM.

SA4 The applicant will review a liquid waste release per FC-211, Waste Liquid Tank Release Permit. Critical steps include determining if conditions necessary to start the release are met. These include maximum allowable flow rate, unloader flow rate, and dilution pump alignment. This is a new JPM.

SA5 The applicant will classify an Emergency Plan event per EPIP-OSC-1, Emergency Classification. Critical steps include determining the Event Category and Event Classification using the Hot and Cold Emergency Action Level Classification Charts and/or EPIP Bases document. This is a time critical, modified bank JPM.

Administrative Topics Outline
Task Summary

Facility:	Fort Calhoun Station	Date of Examination:	Dec / 2015
Examination Level	RO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations (RA1)	M, R	2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc. (3.9). JPM: Perform a Time to Boil Determination (AJPM-RO-SC-1).
Conduct of Operations (RA2)	M, R	2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (4.1). JPM: Calculate an Estimated Critical Boron Concentration (AJPM-RO-RE-1).
Equipment Control (RA3)	M, R	2.2.35 Ability to determine Technical Specification Mode of Operation (3.6). JPM: Determine Technical Specification MODE of Operation (AJPM-RO-TS-1).
Radiation Control (RA4)	N, R	2.3.13 Ability to control radiation releases (3.8). JPM: Respond to Voids in Reactor Coolant System (N).
Emergency Plan	—	—

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

*Type Codes & Criteria:
- (C)ontrol room, (S)imulator, or Class(R)oom
- (D)irect from bank (≤ 3 for ROs; ≤ for 4 for SROs & RO retakes)
- (N)ew or (M)odified from bank (≥ 1)
- (P)revious 2 exams (≤ 1; randomly selected)

Administrative Topics Outline
Task Summary

RA1 The applicant will perform a determination of the time to boil per AOP-19, Loss of Shutdown Cooling, Attachment B, Time to Boil Determination Worksheet, and Attachment C, Time to Boil Determination. Critical steps include determining the time to boil given time after shutdown, initial Reactor Coolant System (RCS) temperature, and RCS pressure boundary condition. This is a modified bank JPM.

RA2 The applicant will calculate an Estimated Critical Boron Concentration per TDB-V.1.B, Technical Data Book Estimated Critical Conditions Worksheet and TDB-II, Technical Data Book Reactivity Curves. The critical steps include calculating the Estimated Critical Boron Concentration based on CEA position, Inverse Boron Worth, Xenon Worth, and time after shutdown to determine the Critical Boron Concentration. This is a modified bank JPM.

RA3 The applicant will determine the Technical Specification MODE per Technical Specifications, Core Operating Limits Report (COLR), and Technical Data Book (TDB). The critical steps include identifying the MODE based on Reactor Coolant System (RCS) temperature and then adjusting MODE determination based on boron concentration, RCS pressure, plant operating conditions, and burnup using the COLR and TDB. This is a modified bank JPM.

RA4 The applicant will determine venting path for voids in the Reactor Coolant System (RCS) per OI-RC-12, Post Accident Venting of Noncondensable Gases from the Reactor Coolant System, following a Small Break Loss Of Coolant Accident. The critical steps include determining the RCS vent source and vent path to minimize effects of post-accident void elimination. This is a new JPM.

Facility: FCS JPM # NRC RA1 Task # 1361 K/A # 2.1.25 3.9 / 4.2

Title: Perform a Time to Boil Determination

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- The plant was shut down 5 days ago for a Reactor Coolant Pump seal repair.
- AOP-19, Loss of Shutdown Cooling, was just entered at 0800.
- Refer to the Attached ERFCS printout for page 195, Shutdown Status Board, for current plant conditions.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- PERFORM a Time to Boil Determination per AOP-19, Loss of Shutdown Cooling, Attachment B, Time to Boil Determination Worksheet.
- RECORD required information on Attachment B, Time to Boil Determination Worksheet wherever "➔" appears.

Task Standard: Utilizing AOP-19, located RCS at Mid Loop graph, recorded appropriate Time to Boil data, and determined Time to Boil at 18 ± 1 minutes.

Required Materials: AOP-19, Loss of Shutdown Cooling, Rev. 18.

Validation Time: 7 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **AOP-19, Loss of Shutdown Cooling.**
  - **Attachment B, Time to Boil Determination Worksheet.**
  - **Attachment C, Time to Boil Determination.**

√ - **Check Mark Denotes Critical Step**

START TIME:	

Examiner Note:	**The following steps are from AOP-19, Attachment B.**
Perform Step: 1 1	Time Shutdown Cooling was lost: _____
Standard:	RECORDED time Shutdown Cooling was lost as **0800** on Attachment B.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√ 2	Last known RCS/SDCS temperature: _____ °F from instrument number: _____
Standard:	DETERMINED representative RCS temperature should be recorded for Core Exit Thermocouples, and RECORDED last known and HIGHEST RCS/SDCS temperature of **110°F** from **CETs** on Attachment B.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3√ 3	Record the following information and inform the Shift Manager on 10 minute intervals.
Standard:	DETERMINED from ERFCS printout that the RCS is at MID LOOP, and referred to Mid Loop graph from Attachment B, and RECORDED **18 ± 1 minutes** on Attachment B.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

INITIAL CONDITIONS: Given the following conditions:

- The plant was shut down 5 days ago for a Reactor Coolant Pump seal repair.
- AOP-19, Loss of Shutdown Cooling, was just entered at 0800.
- Refer to the Attached ERFCS printout for page 195, Shutdown Status Board, for current plant conditions.

INITIATING CUE: The Control Room Supervisor directs you to PERFORM the following:

- PERFORM a Time to Boil Determination per AOP-19, Loss of Shutdown Cooling, Attachment B, Time to Boil Determination Worksheet.
- RECORD required information on Attachment B, Time to Boil Determination Worksheet wherever "➔" appears.

Attachment A – ERFCS Page 195, Shutdown Status Board

1. CET TEMPERATURES

HIGHEST		REPRESENTATIVE A/B	
110.	DEG F	110.	DEG F
		110.	DEG F

2. HOT LEG TEMP.

LOOP1	100.	DEG F	100.	DEG F
LOOP2	100.	DEG F	100.	DEG F

3. COLD LEG TEMP.

LOOP 1A	100.	DEG F	LOOP 2A	100.	DEG F
LOOP 1B	100.	DEG F	LOOP 2B	100.	DEG F

4. RCS LEVEL

RVLMS A	29.0	%	L119	1006.68	FT
RVLMS B	29.0	%	L197	1006.80	FT
L0101X	48.00	%			
L0101Y	0.00	%			

5. SDC HEAT EXCHANGER OUTLET TEMP.

A	91.64	DEG F
B	91.54	DEG F

6. CCW PARAMETERS

PRESSURE	103.33	PSIG
TEMPERATURE	69.31	DEG F
SURGE TANK LEVEL	41.48	IN

7. CVCS

FLOW		GPM
CH-11A LEVEL	88.14	%
CH-11B LEVEL	87.54	%
SIRWT	191.78	IN

8. LPSI FLOW TOTAL 0.0 GPM

LOOP 1A	0.0	GPM
LOOP 1B	0.0	GPM
LOOP 2A	0.0	GPM
LOOP 2B	0.0	GPM

9. HPSI FLOWTOTAL 0.0 GPM

LOOP 1A	0.0	GPM
LOOP 1B	0.0	GPM
LOOP 2A	0.0	GPM
LOOP 2B	0.0	GPM

10. SI PUMPS RUNNING

SI-1A	OFF	SI-2A	OFF	SI-3A	OFF
SI-1B	OFF	SI-2B	OFF	SI-3B	OFF
		SI-2C	OFF	SI-3C	OFF

11. CONTAINMENT FLOOR LEVEL 0.00
0.00

12. CONTAINMENT TEMP. 94.15

13. 4 KV BUSES

1A1	4144.8	VOLTS
1A2	4133.5	VOLTS
1A3	4229.4	VOLTS
1A4	4211.9	VOLTS

14. DG-1 0.000
DG-2 0.000

15. DC BUSES #1 130.0
#2 130.0

16. INSTRUMENT AIR 106.16

PAGE 195 - SHUTDOWN STATUS BOARD

Attachment B

Time to Boil Determination Worksheet

➔1. Time Shutdown Cooling was lost: **0800**

➔2. Last known RCS/SDCS temperature: **110°F**
from instrument number: **CETs**

➔3. Record the following information and inform the Shift Manager on 10 minute intervals.

TIME	CET/HJTC °F	HEATUP RATE	TIME TO BOIL
N/A	N/A	N/A	➔ **18 ± 1 minutes**

Alternate Method: $T_b = T_a + T_0 - T_c$

Where:
T_b is the remaining time to boil
T_a is the approximate time to boil from the appropriate curve
T_o is the time SDC was lost
T_c is the current time

Attachment B

Time to Boil Determination Worksheet

➔1. Time Shutdown Cooling was lost: __________

➔2. Last known RCS/SDCS temperature: __________°F
from instrument number:______

➔3. Record the following information and inform the Shift Manager on 10 minute intervals.

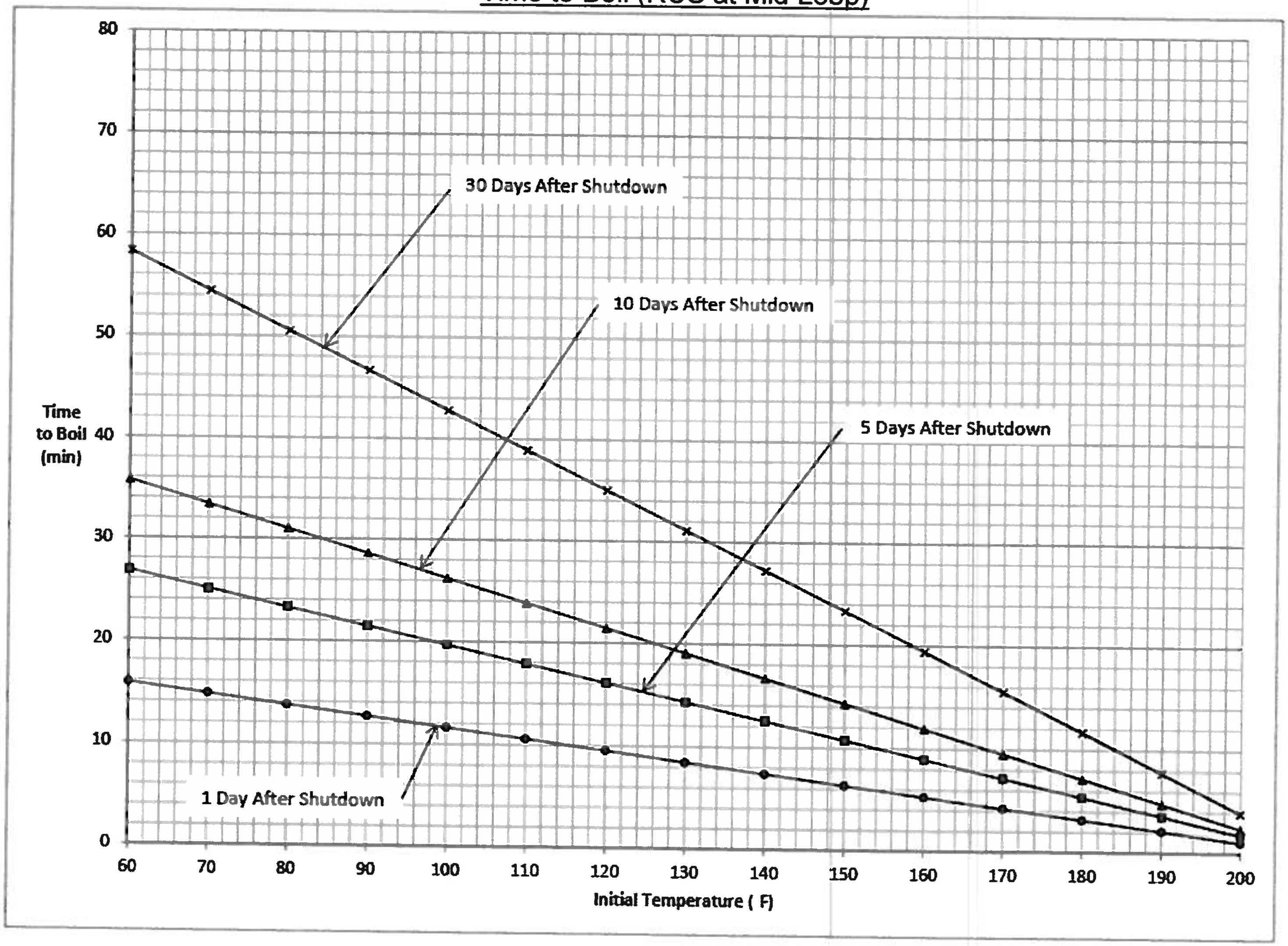
TIME	CET/HJTC °F	HEATUP RATE	TIME TO BOIL
N/A	N/A	N/A	➔

Alternate Method: $T_b = T_a + T_0 - T_c$

Where: T_b is the remaining time to boil
T_a is the approximate time to boil from the appropriate curve
T_o is the time SDC was lost
T_c is the current time

Attachment B
Time to Boil Determination Worksheet

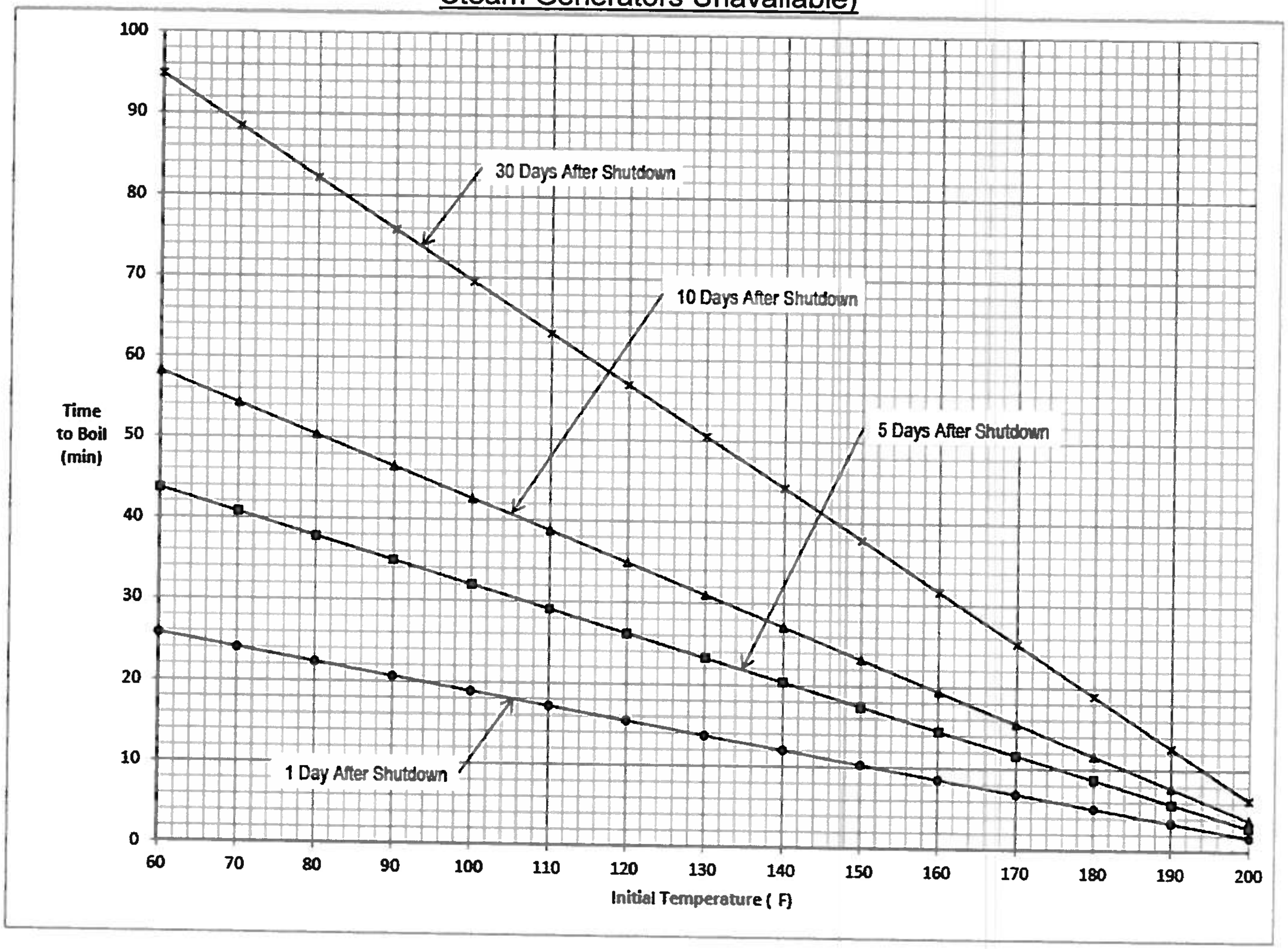
Time to Boil (RCS at Mid Loop)



Attachment B
Time to Boil Determination Worksheet

Time to Boil (RCS at Reactor Vessel Flange)

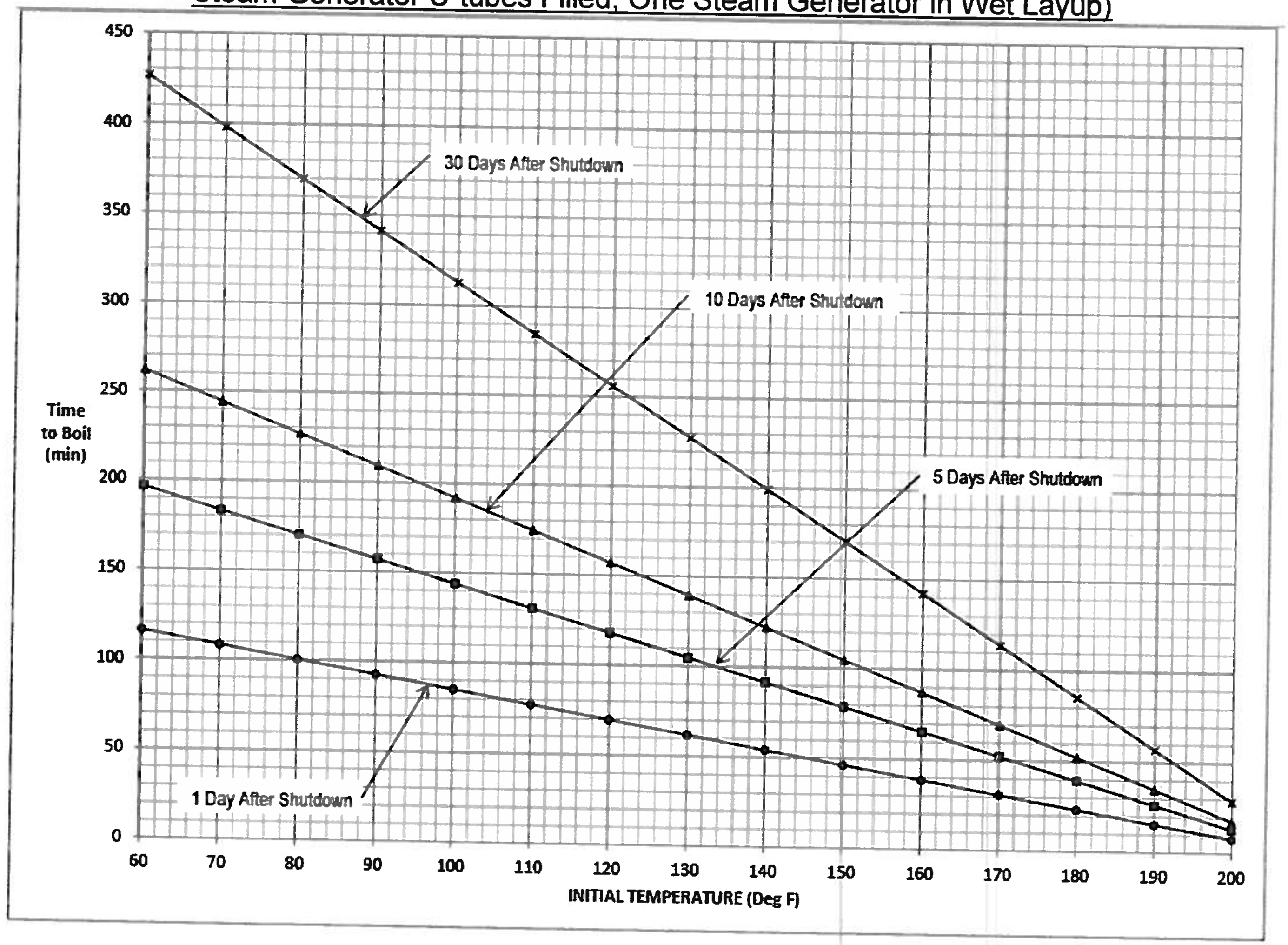
Steam Generators Unavailable)



Attachment B
Time to Boil Determination Worksheet

Time to Boil (RCS Open, RV Head on, RCS Level > 1013',

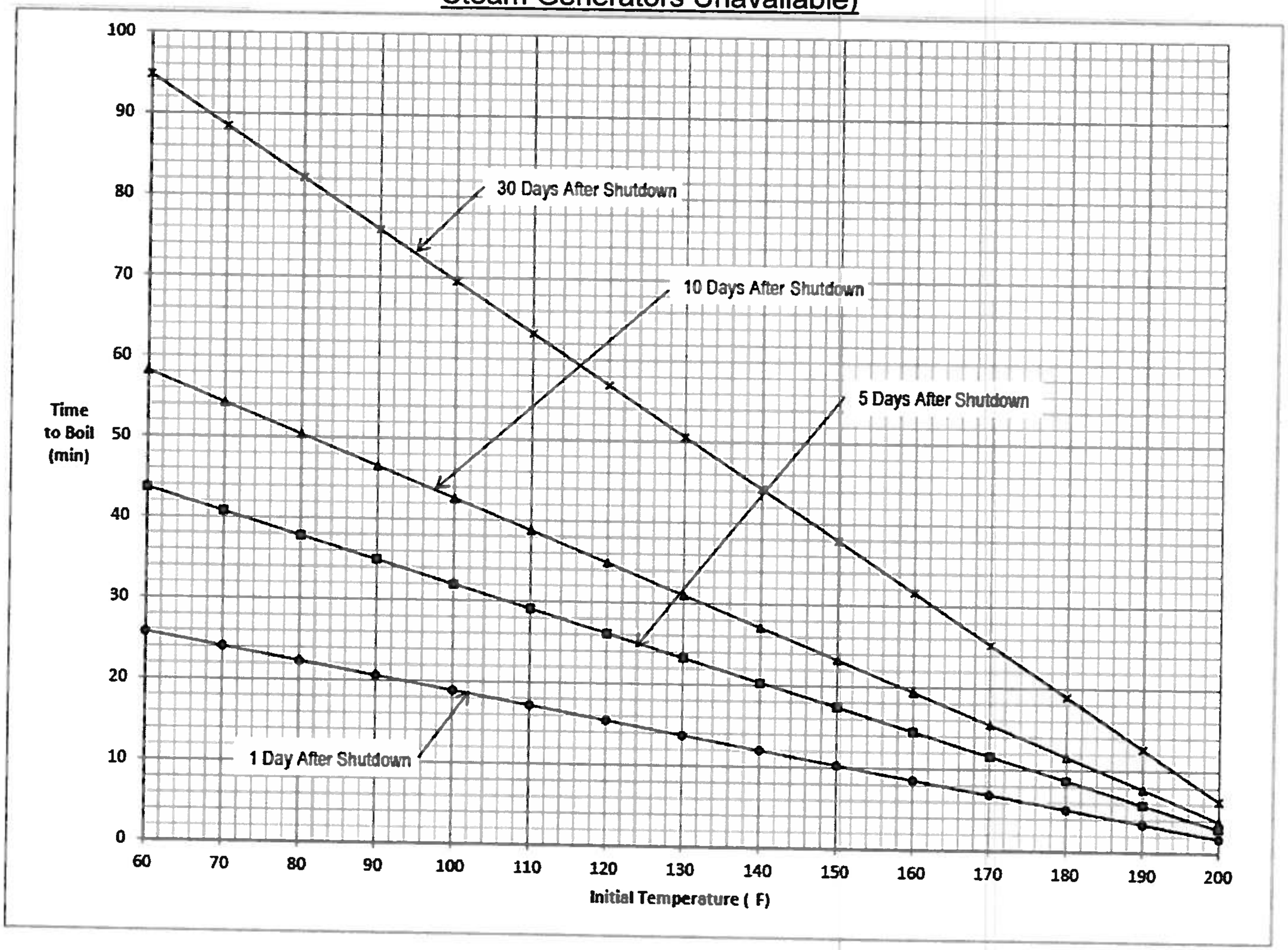
Steam Generator U-tubes Filled, One Steam Generator in Wet Layup)



Attachment B
Time to Boil Determination Worksheet

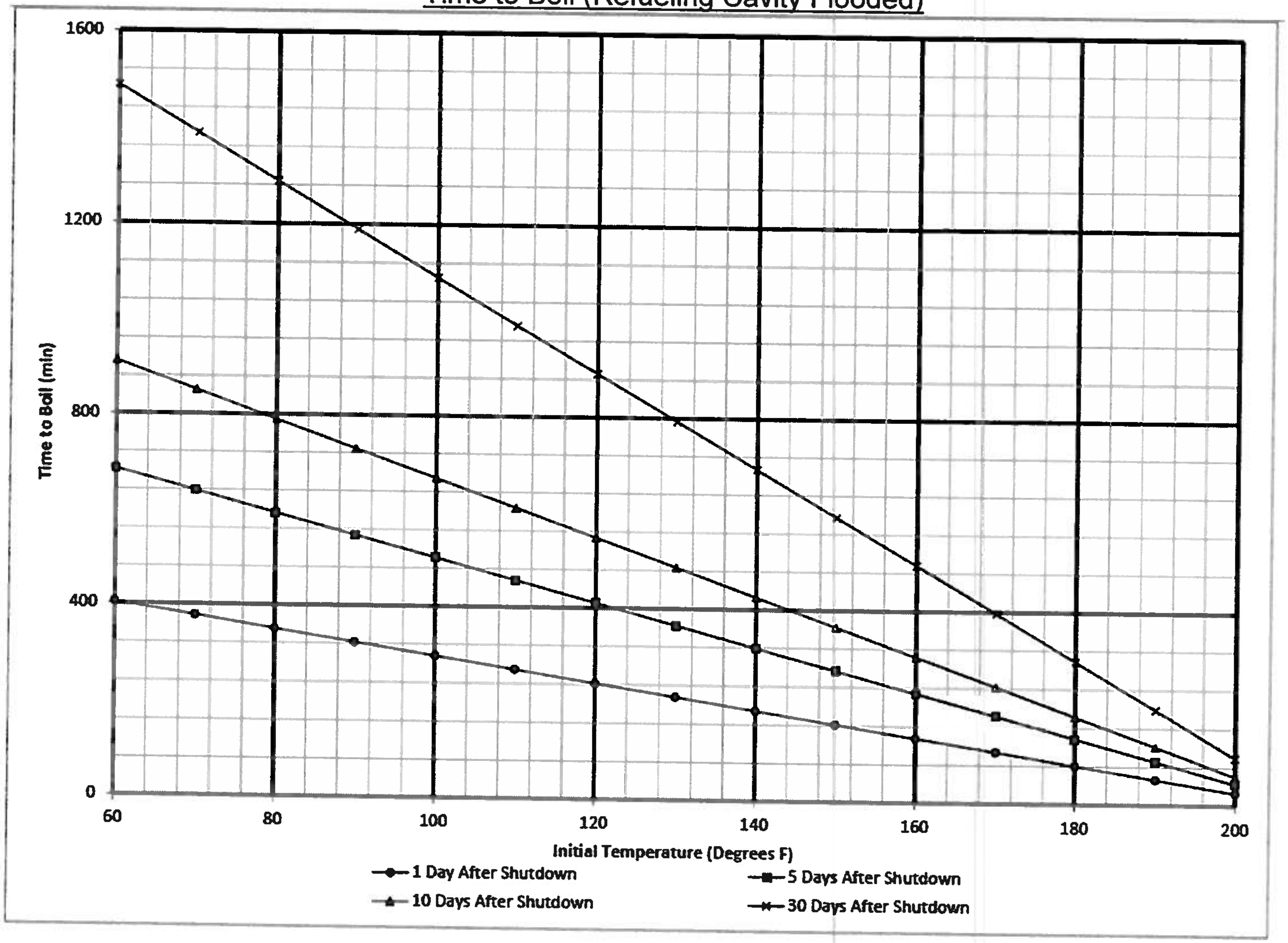
Time to Boil (RCS at Normal Operating Volume

Steam Generators Unavailable)



Attachment B
Time to Boil Determination Worksheet

Time to Boil (Refueling Cavity Flooded)



End of Attachment B

Attachment C

Time to Boil Determination

INSTRUCTIONS	CONTINGENCY ACTIONS
1. **(STA)** Record items 1 and 2 on Attachment B, Time to Boil Determination Worksheet.	
2. **(STA) IF** initial RCS temperature is greater than 210°F, **THEN** monitor RCS conditions by performing the following:	
a. Monitor RCS pressure and core temperature.	
b. Verify Subcooled Margin is greater than 20°F.	b.1 Notify the Shift Manager that saturation conditions exist in the RCS.

Attachment C

Time to Boil Determination

INSTRUCTIONS

3. **(STA)** Determine the remaining time to boil using the current CET or HJTC temperatures and **ONE** of the following figures from Attachment B, Time to Boil Determination Worksheet:

   - Time to Boil (RCS at Mid Loop)
   - Time to Boil (RCS at Reactor Vessel Flange Steam Generators Unavailable)
   - Time to Boil (RCS open, RV Head on RCS level greater than 1013 ft, Steam Generator U-tubes filled, one Steam Generator in wet layup)
   - Time to Boil (RCS at Normal Operating Volume Steam Generators Unavailable)

4. **(STA)** Determine the remaining time to boil using the last valid TR 346, "SHUTDOWN COOLING INLET/OUTLET TEMPERATURE" inlet temperature and the Time to Boil (Refueling Cavity Flooded) figure from Attachment B, Time to Boil Determination Worksheet.

CONTINGENCY ACTIONS

3.1 **IF** CET or HJTC temperatures are **NOT** available,
THEN perform the Alternate Method on Attachment B, Time to Boil Determination Worksheet.

Attachment C

Time to Boil Determination

INSTRUCTIONS | CONTINGENCY ACTIONS

5. **IF** time to boil is less than 30 minutes, **THEN** establish Containment Closure by performing the following:

   a. **(SFM)** Close the Equipment Hatch.

   b. Close FH-11, "FUEL TRANSFER TUBE ISOLATION VALVE" (Spent Fuel Pool).

   c. Close any other breach of Containment.

Attachment C

Time to Boil Determination

INSTRUCTIONS	CONTINGENCY ACTIONS
6. Evacuate Containment by performing the following:	
a. Sound the Emergency Alarm for 10-15 seconds.	
b. Announce and repeat the following using the Plant communications system: "Attention all personnel. Attention all personnel. All personnel evacuate Containment".	
7. **WHEN** SDC is returned to service, **THEN** GO TO Section 5.0, Exit Conditions.	

End of Attachment C

Facility: FCS JPM # NRC RA2 Task # 1528 K/A # 2.1.43 4.1 / 4.3

Title: Calculate an Estimated Critical Boron Concentration

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A Reactor Trip from 100% power occurred on 12/03/15 at 0400 following 6 months of full power operation.
- All Control Rods were fully withdrawn at the time of the trip.
- Boron concentration prior to the trip was 400 ppm.
- Average Core Burnup is 6 GWD/MTU.
- Current boron concentration is 500 ppm.
- Criticality is scheduled to occur with Regulating Group 4 at 78 inches.
- Reactor Engineering reports no correction is needed for boron depletion.
- Reactor Startup is scheduled for 12/10/15 at 0600.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- CALCULATE estimated critical boron concentration per TDB-V-1-B, Estimated Critical Conditions Worksheet.
- COMPLETE data entry through TDB-V.1.B, Step E.3.d, Estimated Critical Boron Concentration.

Task Standard: Utilizing TDB-V.1.B and TDB-II, calculated Estimated Critical Boron Concentration.

Required Materials: TDB-V.1.B, Estimated Critical Conditions Worksheet, Rev. 26.
TDB-II, Technical Data Book Reactivity Curves, Rev. 35.

Validation Time: 40 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **TDB-V.1.B, Estimated Critical Conditions Worksheet.**
- **TDB-II, Technical Data Book Reactivity Curves.**
- **Calculator**
- **Straight Edge**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from TDB-V.1.B**
Perform Step: 1 A	Conditions at Time of Shutdown.
Standard:	ENTERED Conditions at Time of Shutdown in TDB-V-1-B Steps A.1 to A.5.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 B	Conditions at Time of Startup.
Standard:	ENTERED Conditions at Time of Startup in TDB-V.1.B Steps B.1 to B.4.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 C	ECC Applicability.
Standard:	DETERMINED early and late date/time limits for ECC Applicability and entered data in TDB-V.1.B Steps C.1 to C.4.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ D	Reactivity Changes Due To Shutdown.
Standard:	CALCULATED and ENTERED Reactivity Changes Due To Shutdown in TDB-V.1.B Steps D.1 to D.5.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ E	Estimated Critical Boron Concentration.
Standard:	CALCULATED and ENTERED Estimated Critical Boron Concentration in TDB-V.1.B Steps E.1 to E.3. (+/- 25 ppm from key)
Examiner Note:	**Information found on Answer Key.**
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

<u>INITIAL</u> <u>CONDITIONS</u>: **Given the following conditions:**

- **A Reactor Trip from 100% power occurred on 12/03/15 at 0400 following 6 months of full power operation.**
- **All Control Rods were fully withdrawn at the time of the trip.**
- **Boron concentration prior to the trip was 400 ppm.**
- **Average Core Burnup is 6 GMWD/MTU.**
- **Current boron concentration is 500 ppm.**
- **Criticality is scheduled to occur with Regulating Group 4 at 78 inches.**
- **Reactor Engineering reports no correction is needed for boron depletion.**
- **Reactor Startup is scheduled for 12/10/15 at 0600.**

<u>INITIATING</u> <u>CUE</u>: **The Control Room Supervisor directs you to PERFORM the following:**

- **CALCULATE estimated critical boron concentration per TDB-V.1.B, Estimated Critical Conditions Worksheet.**
- **COMPLETE data entry through TDB-V.1.B, Step E.3.d, Estimated Critical Boron Concentration.**

ESTIMATED CRITICAL CONDITION WORKSHEET

Part I - Performed after a mid-cycle shutdown where Xenon may be present.

____
Criticality No.

TODAY
Date

A. Conditions at Time of Shutdown

A.1. Shutdown date and time. 12/03/15 (DATE) 0400 (TIME)

A.2. Reactor power before shutdown. 100 %

A.3. CEA Positions before time of shutdown.

Group 1 126 inches

Group 2 126 inches

Group 3 126 inches

Group 4 126 inches

Group N 126 inches

A.4. Reactor Coolant System boron concentration before shutdown. 400 PPM

A.5. Core average burnup. 6000 MWD/MTU

B. Conditions at Time of Startup

B.1. Startup date and time. 12/10/15 (DATE) 0600 (TIME)

B.2. Time interval between shutdown and startup. 170 hours

B.3. Desired CEA positions at time of startup. (Manual Sequential Mode)

Group 1 126 inches

Group 2 126 inches

Group 3 126 inches

(1) Group 4 78 inches

Group N 126 inches

(1) Normally 85 inches or as specified by the Reactor Engineer.

B.4. Present Reactor Coolant System boron concentration. 500 PPM

C. ECC Applicability

C.1. Using table 1, determine early and late date/time limits for ECC applicability.

Table 1

Time since shutdown (hours)	0-48	48-84	84+
ECC Applicability (hours)	+/-1	+/-2	No limit

C.2. ECC Part I is applicable for plus/minus No Limit hours of predicted startup date and time recorded in B.1.

C.3. Early date and time. N/A (DATE) N/A (TIME)

C.4. Late date and time. N/A (DATE) N/A (TIME)

D. Reactivity Changes Due to Shutdown

NOTE: Due to transient conditions which may not have been assumed in the generation of TDB figures, the Reactor Engineer may request DEN to provide present condition data. This data will be generated with approved computer codes and may be entered in place of TDB figure data. Such entries shall be annotated.

NOTE: When using TDB figures to obtain reactivity values, enter absolute values (no signs). When performing calculations with reactivity values the answer will be in the proper sign. Record calculated value with sign.

D.1. Reactivity change due to change in power:

Using TDB Figure II.C.2.a or II.C.2.b, the power level from A.2, and the Average Core Burnup from A.5, record the power defect added due to the shutdown.

+ 1.778 or + 1.74 %Δρ
D.1

D.2. Reactivity change due to control rod position change:

NOTE: The applicable TDB Figures II.B.2 or II.B.3 can be used, but the same figure must be used in both Steps D.2.a. and D.2.b.

D.2.a. Reactivity change due to rod insertion:

Using the applicable TDB Figure, the CEA positions from A.3, and the Average Core Burnup from A.5, record the reactivity contribution due to inserting all regulating rods.

3.29 or 0.337 %Δρ
D.2.a

D.2.b. Reactivity change due to rod withdrawal:

Using the applicable TDB Figure, the CEA positions from B.3, and the Average Core Burnup from A.5, record the reactivity contribution due to withdrawal of CEA's to startup positions.

3.10 or 0.15 %Δρ
D.2.b

D.2.c. Calculate the net reactivity change due to net control rod position change. Subtract D.2.a from D.2.b. Be sure to enter the sign of the difference.

(3.10 or 0.15) - (3.29 or 0.337) = - 0.19 (-0.187) %Δρ
D.2.b D.2.a D.2.c

D.3. Reactivity change due to changes in Xenon Concentrations.

NOTE: If equilibrium conditions existed before plant shutdown, TDB Figures II.D.1.a, II.D.1.b or II.D.2 may be used to determine the shutdown Xenon worth.

NOTE: If equilibrium conditions existed before shutdown, and the plant tripped, TDB Figure II.D.2 may be used to determine both shutdown and startup Xenon worth.

NOTE: If the above conditions do not apply, use the conditions of Parts A and B above to determine the reactivity change due to Xenon.

D.3.a. Shutdown Xenon worth $\underset{\text{D.3.a}}{\underline{\boxed{2.70}}}$ %Δρ

D.3.b. Startup Xenon worth $\underset{\text{D.3.b}}{\underline{\boxed{0.0}}}$ %Δρ

D.3.c. Calculate the net reactivity change due to Xenon. Subtract D.3.b from D.3.a. Be sure to enter the algebraic sign of the difference.

$$(\underset{\text{D.3.a}}{\underline{\boxed{2.70}}}) - (\underset{\text{D.3.b}}{\underline{\boxed{0.0}}}) = \underset{\text{D.3.c}}{\underline{\boxed{+2.70}}}\ \%\Delta\rho$$

D.4. Reactivity change due to change in boron concentration.

D.4.a. If there is no change in boron concentration enter 0 on line D.4.e below and proceed to Step D.5.

D.4.b. Enter TDB Figure II.A.4 using the burnup of A.5 above. Record the HZP inverse soluble boron worth.

$$\underset{\text{D.4.b}}{\underline{\boxed{113}}}\ \text{ppm}/\%\Delta\rho$$

D.4.c. Reactivity due to soluble boron concentration at time of shutdown. Divide boron concentration A.4 by inverse boron worth D.4.b.

$$(\underset{\text{A.4}}{\underline{\boxed{400}}}) / (\underset{\text{D.4.b}}{\underline{\boxed{113}}}) = \underset{\text{D.4.c}}{\underline{\boxed{3.54}}}\ \%\Delta\rho$$

D.4.d. Reactivity due to soluble boron concentration at time of startup. Divide present boron concentration B.4 by inverse boron worth D.4.b.

$$(\underset{\text{B.4}}{\underline{\boxed{500}}}) / (\underset{\text{D.4.b}}{\underline{\boxed{113}}}) = \underset{\text{D.4.d}}{\underline{\boxed{4.43}}}\ \%\Delta\rho$$

D.4.e. Calculate the net reactivity change due to boron concentration change. Subtract D.4.d from D.4.c above. Enter the algebraic sign of the difference.

$$(\underset{\text{D.4.c}}{\underline{\boxed{3.54}}}) - (\underset{\text{D.4.d}}{\underline{\boxed{4.43}}}) = \underset{\text{D.4.e}}{\underline{\boxed{-0.89}}}\ \%\Delta\rho$$

D.5. Sum of reactivity changes:

(Find the algebraic sum of items D.1, D.2.c, D.3.c and D.4.e. Be sure to include the algebraic sign.)

D.5.a. Change in reactivity due to change in power. $\boxed{+ 1.778 \text{ (or } 1.74)}$ %Δρ (D.1)

D.5.b. Change in reactivity due to control rod position change. $\boxed{- 0.19}$ %Δρ (D.2.c)

D.5.c. Change in reactivity due to Xenon Transient. $\boxed{+2.70}$ %Δρ (D.3.c)

D.5.d. Change in reactivity due to change in boron concentration. $\boxed{- 0.89}$ %Δρ (D.4.e)

D.5.e. Total $\boxed{+ 3.398 \text{ (or } 3.36)}$ %Δρ (D.5.e)

E. <u>Estimated Critical Boron Concentration</u>

E.1. Find and record inverse boron worth using TDB Figure II.A.4 at HZP and the burnup A.5 above.

Inverse boron worth = $\boxed{113}$ ppm/% Δρ (E.1)

E.2. Find the change in boron concentration by multiplying D.5.e by E.1. (Be sure to transcribe the algebraic sign of D.5.e.)

$\boxed{+ 3.398 \text{ (or } 3.36)}$ (D.5.e) % Δρ X $\boxed{113}$ (E.1) ppm/% Δρ = $\boxed{384 \text{ (or } 380)}$ (E.2) ppm

E.3. Find the estimated boron concentration.

E.3.a. Present boron concentration $\boxed{500}$ ppm (B.4)

E.3.b. Change in boron concentration $\boxed{384 \text{ (or } 380)}$ ppm (E.2)

E.3.c. If large RCS Volume changes (eg., draining to mid-loop) are made, an adjustment may be needed to compensate for changes in B-10 concentration. Adjustments may be required, depending on various factors such as time in life or the length of the operating cycle between shutdowns. Contact Reactor Performance Analysis for assistance in determining this value.

[0] ppm
E.3.c

E.3.d. Estimated critical boron concentration (ECB)

= ([500] E.3.a) + ([384 (or 380)] E.3.b) + ([0] E.3.c) = [884 (±25)] 0 ppm

NRC RO APPLICANTS STOP HERE

F. Minimum and Maximum Critical Rod Position

NOTE: Mark Steps F and G N/A if diluting to critical.

F.1. Determination of critical position CEA worth.

F.1.a. Use data from B.3, D.2.b for desired critical position.

Group ____________ at __________ inches F.1.a

F.1.b. Using desired position from F.1.a, applicable TDB Figure II.B.2 and core average burnup A.5, determine critical position CEA worth.

CEA worth at desired critical position: ____________ %Δρ
F.1.b

F.2. Find the maximum critical CEA position.

F.2.a. Calculate critical CEA worth +0.5% Δρ by adding 0.5% Δρ to the critical CEA worth F.1.b.

(__________ F.1.b) + 0.5% Δρ = _________ F.2.a % Δρ

F.2.b. Using F.2.a, the applicable TDB Figure II.B.2 the Core Average Burnup A.5, determine the maximum critical CEA position.

Group ____________ at __________ inches F.2.b

Fort Calhoun Station
Unit 1

TDB-V.1.B

TECHNICAL DATA BOOK

ESTIMATED CRITICAL CONDITIONS WORKSHEET

Change No.	EC 65304
Reason for Change	Change procedure so there is only one start up procedure.
Requestor	T. Korenak
Preparer	K. Bessey
Issue Date	05-15-15 3:00 pm

ESTIMATED CRITICAL CONDITION WORKSHEET

Part I - Performed after a mid-cycle shutdown where Xenon may be present.

____
Criticality No.

____
Date

A. Conditions at Time of Shutdown

A.1. Shutdown date and time. ____________ ____________
DATE TIME

A.2. Reactor power before shutdown. __________%

A.3. CEA Positions before time of shutdown.

Group 1 ___________ inches

Group 2 ___________ inches

Group 3 ___________ inches

Group 4 ___________ inches

Group N ___________ inches

A.4. Reactor Coolant System boron concentration before shutdown. _________ PPM

A.5. Core average burnup. _________ MWD/MTU

B. Conditions at Time of Startup

B.1. Startup date and time. ____________ ____________
DATE TIME

B.2. Time interval between shutdown and startup. __________ hours

B.3. Desired CEA positions at time of startup. (Manual Sequential Mode)

Group 1 ____________ inches

Group 2 ____________ inches

Group 3 ____________ inches

(1) Group 4 ____________ inches

Group N ____________ inches

(1) Normally 85 inches or as specified by the Reactor Engineer.

B.4. Present Reactor Coolant System boron concentration. __________ PPM

C. ECC Applicability

C.1. Using table 1, determine early and late date/time limits for ECC applicability.

Table 1

Time since shutdown (hours)	0-48	48-84	84+
ECC Applicability (hours)	+/-1	+/-2	No limit

C.2. ECC Part I is applicable for plus/minus _______ hours of predicted startup date and time recorded in B.1.

C.3. Early date and time. __________ __________
DATE TIME

C.4. Late date and time. __________ __________
DATE TIME

D. Reactivity Changes Due to Shutdown

NOTE: Due to transient conditions which may not have been assumed in the generation of TDB figures, the Reactor Engineer may request DEN to provide present condition data. This data will be generated with approved computer codes and may be entered in place of TDB figure data. Such entries shall be annotated.

NOTE: When using TDB figures to obtain reactivity values, enter absolute values (no signs). When performing calculations with reactivity values the answer will be in the proper sign. Record calculated value with sign.

D.1. Reactivity change due to change in power:

Using TDB Figure II.C.2.a or II.C.2.b, the power level from A.2, and the Average Core Burnup from A.5, record the power defect added due to the shutdown.

______________ %Δρ
D.1

D.2. Reactivity change due to control rod position change:

NOTE: The applicable TDB Figures II.B.2 or II.B.3 can be used, but the same figure must be used in both Steps D.2.a. and D.2.b.

D.2.a. Reactivity change due to rod insertion:

Using the applicable TDB Figure, the CEA positions from A.3, and the Average Core Burnup from A.5, record the reactivity contribution due to inserting all regulating rods.

______________ %Δρ
D.2.a

D.2.b. Reactivity change due to rod withdrawal:

Using the applicable TDB Figure, the CEA positions from B.3, and the Average Core Burnup from A.5, record the reactivity contribution due to withdrawal of CEA's to startup positions.

______________ %Δρ
D.2.b

D.2.c. Calculate the net reactivity change due to net control rod position change. Subtract D.2.a from D.2.b. Be sure to enter the sign of the difference.

(______________) - (______________) = ____________ %Δρ
D.2.b D.2.a D.2.c

D.3. Reactivity change due to changes in Xenon Concentrations.

NOTE: If equilibrium conditions existed before plant shutdown, TDB Figures II.D.1.a, II.D.1.b or II.D.2 may be used to determine the shutdown Xenon worth.

NOTE: If equilibrium conditions existed before shutdown, and the plant tripped, TDB Figure II.D.2 may be used to determine both shutdown and startup Xenon worth.

NOTE: If the above conditions do not apply, use the conditions of Parts A and B above to determine the reactivity change due to Xenon.

D.3.a. Shutdown Xenon worth ____________ %Δρ
D.3.a

D.3.b. Startup Xenon worth ____________ %Δρ
D.3.b

D.3.c. Calculate the net reactivity change due to Xenon. Subtract D.3.b from D.3.a. Be sure to enter the algebraic sign of the difference.

(____________) - (____________) = ____________ %Δρ
D.3.a D.3.b D.3.c

D.4. Reactivity change due to change in boron concentration.

D.4.a. If there is no change in boron concentration enter 0 on line D.4.e below and proceed to Step D.5.

D.4.b. Enter TDB Figure II.A.4 using the burnup of A.5 above. Record the HZP inverse soluble boron worth.

____________ ppm/%Δρ
D.4.b

D.4.c. Reactivity due to soluble boron concentration at time of shutdown. Divide boron concentration A.4 by inverse boron worth D.4.b.

(____________) / (____________) = ____________ %Δρ
A.4 D.4.b D.4.c

D.4.d. Reactivity due to soluble boron concentration at time of startup. Divide present boron concentration B.4 by inverse boron worth D.4.b.

(____________) / (____________) = ____________ %Δρ
B.4 D.4.b D.4.d

D.4.e. Calculate the net reactivity change due to boron concentration change. Subtract D.4.d from D.4.c above. Enter the algebraic sign of the difference.

(____________) - (____________) = ____________ %Δρ
D.4.c D.4.d D.4.e

D.5. Sum of reactivity changes:

(Find the algebraic sum of items D.1, D.2.c, D.3.c and D.4.e. Be sure to include the algebraic sign.)

D.5.a. Change in reactivity due to change in power. ______________ %Δρ
D.1

D.5.b. Change in reactivity due to control rod position change.

______________ %Δρ
D.2.c

D.5.c. Change in reactivity due to Xenon Transient. ______________ %Δρ
D.3.c

D.5.d. Change in reactivity due to change in boron concentration.

______________ %Δρ
D.4.e

D.5.e. Total

______________ %Δρ
D.5.e

E. Estimated Critical Boron Concentration

E.1. Find and record inverse boron worth using TDB Figure II.A.4 at HZP and the burnup A.5 above.

Inverse boron worth = ______________ ppm/% Δρ
E.1

E.2. Find the change in boron concentration by multiplying D.5.e by E.1. (Be sure to transcribe the algebraic sign of D.5.e.)

____________ % Δρ X ______________ ppm/% Δρ = ______________ ppm
D.5.e E.1 E.2

E.3. Find the estimated boron concentration.

E.3.a. Present boron concentration ______________ ppm
B.4

E.3.b. Change in boron concentration ___________ ppm
E.2

E.3.c. If large RCS Volume changes (eg., draining to mid-loop) are made, an adjustment may be needed to compensate for changes in B-10 concentration. Adjustments may be required, depending on various factors such as time in life or the length of the operating cycle between shutdowns. Contact Reactor Performance Analysis for assistance in determining this value.

______________ ppm
E.3.c

E.3.d. Estimated critical boron concentration (ECB)

= (__________) + (__________) + (__________) = __________ ppm
E.3.a E.3.b E.3.c 0

NRC RO APPLICANTS STOP HERE

F. Minimum and Maximum Critical Rod Position

NOTE: Mark Steps F and G N/A if diluting to critical.

F.1. Determination of critical position CEA worth.

F.1.a. Use data from B.3, D.2.b for desired critical position.

Group ______________ at ___________ inches F.1.a

F.1.b. Using desired position from F.1.a, applicable TDB Figure II.B.2 and core average burnup A.5, determine critical position CEA worth.

CEA worth at desired critical position: ______________ %$\Delta\rho$
F.1.b

F.2. Find the maximum critical CEA position.

F.2.a. Calculate critical CEA worth +0.5% $\Delta\rho$ by adding 0.5% $\Delta\rho$ to the critical CEA worth F.1.b.

(____________) + 0.5% $\Delta\rho$ = __________ % $\Delta\rho$
F.1.b F.2.a

F.2.b. Using F.2.a, the applicable TDB Figure II.B.2 the Core Average Burnup A.5, determine the maximum critical CEA position.

Group ______________at ____________ inches F.2.b

F.2.c. Use the lower of CEA position of F.2.b and ARO as the maximum critical CEA position.

Group ______________ at ______________ inches F.2.c

F.3. Find the minimum critical CEA position.

F.3.a. Calculate the critical CEA worth – 0.5% $\Delta\rho$ by subtracting 0.5% $\Delta\rho$ from the critical CEA worth F.1.b.

__________ - 0.5% $\Delta\rho$ = _______ % $\Delta\rho$
F.1.b F.3.a

F.3.b. Using F.3.a, the applicable TDB Figure II.B.2 and the Core Average Burnup A.5, find the CEA position corresponding to estimated critical rod position – 0.5%$\Delta\rho$ and enter here.

Group ____________at __________ inches F.3.b

F.3.c. Using TDB Section VI Figure 2 record the zero power dependent insertion limit by obtaining PDIL or transient insertion limit value for 0% power.

Group ____________at __________ inches F.3.c

F.3.d. Use the higher of CEA positions from F.3.b and F.3.c as the minimum critical CEA position.

Group ____________at ___________ inches F.3.d

F.4. Estimated Critical Condition Summary

Present Boron Concentration______________ppm

Estimated Critical Boron Concentration_________________ppm

Minimum Critical Position	Group___________	at_________	inches (F.3.d)
Estimated Critical Position	Group___________	at_________	inches (F.1.a)
Maximum Critical Position	Group___________	at_________	inches (F.2.c)

Reactor Engineer (QNE)______________________ Date/Time _______/______

G. <u>Perform the Following If the Reactor Is Not Critical with Group 4 at 115", Otherwise this Step Is N/A</u>

G.1. Using TDB Figure II.B.3, determine the amount of reactivity needed to offset inserting Group 4 to 85"

__________ - __________ = __________ % Δρ
CEA worth @ 115"   CEA worth @ 85"   G.1

G.2. Calculate twice the CEA worth difference:

__________ X 2 = __________ % Δρ
G.1   G.2

G.3. Determine rod worth at twice the reactivity difference:

__________ - __________ = __________ % Δρ
CEA worth @ 115"   G.2   G.3

G.4. Using TDB Figure II.B.3 and the rod worth calculated in Step G.3, determine the minimum position of Group 4.

__________ inches withdrawn
Group 4

G.5. Calculate FCS boron dilution change using inverse boron worth (Step E.1) and the reactivity difference calculated in Step G.1.

__________ ppm/%Δρ X __________ % Δρ = __________ ppm
E.1   G.1   G.5

G.6. Calculate boron concentration change,

__________ ppm - __________ ppm = __________ ppm
0   G.5   G.6

G.7. Performed by: ____________________ Date/Time ______/______

G.8. Verify Step G calculations reviewed prior to Reactor Criticality.

Reactor Engineer: ____________________ Date/Time ______/______

H. Minimum and Maximum Critical Boron Concentration for Reactor Startup by Dilution

NOTE: Mark Step H N/A if using rod position for criticality.

H.1. Calculation of maximum critical boron concentration, ECB + 0.5%Δρ.

__________ ppm + (__________ ppm/%Δρ X 0.5%Δρ) = __________ ppm
0 D.4.b Max Boron Conc

H.2. Calculate minimum critical boron concentration. Limit + ECB - 50 ppm

__________ ppm - 50 ppm = __________ ppm
0 Min Boron Conc

H.3. Results of this calculation have been independently reviewed prior to Reactor Criticality.

Performed by: ____________________ Date/Time ______ / ______

Reactor Engineer: ____________________ Date/Time ______ / ______

I. Actual Critical Data

I.1. After achieving criticality, complete the following record:

I.1.a. CEA Positions

Group 1 ____________ inches

Group 2 ____________ inches

Group 3 ____________ inches

Group 4 ____________ inches

Group N ____________ inches

I.1.b. Reactor Coolant System boron concentration: ______________ ppm

I.1.c. Time at which criticality was attained: _____________

I.1.d. Reactor Coolant System average temperature (T_{AVG}) _____ °F

I.2. Ensure actual critical data recorded in Step I.1 is logged in the Control Room Log.

Completed by: ____________________________________ Date/Time _______/______

J. Review

J.1. Forward to Reactor Engineer for review.

NOTE: If the reactivity difference between actual and estimated critical conditions is greater than 0.5%Δρ, then the Reactor Engineer will determine if the appropriate curves should be upgraded.

Reactor Engineer ________________________________ _________/______

Reactor Engineer   Date/Time

Part II - Performed after a refueling outage.

____
Criticality No.

____
Date

A. <u>Conditions at Time of Startup</u>

A.1. Startup date and time. ______________ ____________
DATE TIME

A.2. Desired CEA Group 4 position at time of startup. (Manual Sequential Mode)

Group 4 ______________ inches

A.3. Sequential Rod Worth at ARO (TDB Figure II.B.2.a) ______________ $\%\Delta\rho$

A.4. Sequential Rod Worth, Group 4 at position from A.2 (TDB Figure II.B.2.a) ______________ $\%\Delta\rho$

A.5. Critical Boron Concentration for BOC, ARO, HZP No Xenon (TDB Figure II.A.1.a.1) _____________ ppm

A.6. Reactor Coolant System Inverse Boron Worth (TDB Figure II.A.4) ______________ ppm / $\%\Delta\rho$

A.7. Current RCS Boron Concentration ___________ ppm Sample Date/Time __________

B. <u>Estimated Critical Boron Concentration</u>

B.1. Change in reactivity due to CEA Group 4

______________ $\%\Delta\rho$ – ______________ $\%\Delta\rho$ = ______________ $\%\Delta\rho$
A.3 A.4 B.1

B.2. Change in boron due to CEA Group 4

______________ $\%\Delta\rho$ X ______________ ppm/$\%\Delta\rho$ = ______________ ppm
B.1 A.6 B.2

B.3. Estimated Critical Boron Concentration (ECB)

______________ ppm – ______________ ppm = ______________ ppm
A.5 B.2 B.3

C. Minimum and Maximum Critical Rod Position

C.1. Find the minimum critical CEA position.

C.1.a. Calculate the critical CEA worth – 0.5%Δρ by subtracting 0.5%Δρ from the critical CEA worth A.4.

______________ (A.4) – 0.5%Δρ = ______________ (C.1.a) %Δρ

C.1.b. Using TDB Figure II.B.2.a, find the CEA position corresponding to estimated critical rod position – 0.5%Δρ and enter here.

Group ______________ at ______________ inches

C.1.c. Using TDB Section VI COLR Figure 2, record the zero power dependent insertion limit by obtaining PDIL or transient insertion limit value for 0% power.

Group ______________ at ______________ inches

C.1.d. Use the higher of CEA positions from C.1.b and C.1.c as the minimum critical CEA position.

Group ______________ at ______________ inches

C.2. Find the maximum critical CEA position.

C.2.a. Calculate critical CEA worth +0.5% Δρ by adding 0.5% Δρ to the critical CEA worth A.4.

______________ (A.4) + 0.5%Δρ = ______________ (C.1.a.1) %Δρ

C.2.b. Using TDB Figure II.B.2.a, find the CEA position corresponding to estimated critical rod position + 0.5%Δρ (C.1.a.1) and enter here.

Group ______________ at ______________ inches

C.2.c. Use the lower of CEA position of C.2.b and ARO as the maximum critical CEA position.

Group ______________ at ______________ inches

Present Boron Concentration______________ppm

Estimated Critical Boron Concentration_________________ppm

Minimum Critical Position	Group____________	at_________	inches (C.1.d)
Estimated Critical Position	Group____________	at_________	inches (A.2)
Maximum Critical Position	Group____________	at_________	inches (C.2.c)

C.3. Completed by ______________________________ Date/Time ________/________

C.4. Results of this calculation have been independently reviewed prior to Reactor Criticality.

Reactor Engineer ______________________________ Date/Time ________/________

D. Perform the Following if the Reactor Is Not Critical with Group 4 at 115" (Otherwise this is N/A)

D.1. Using Figure TDB II.B.3.a, determine the amount of reactivity needed to offset inserting Group 4 to 85".

______________ – ______________ = ______________ %Δρ
CEA worth @ 115" CEA worth @ 85" D.1

D.2. Calculate RCS boron dilution change using Inverse Boron Worth (Step A.6) and the reactivity difference calculated in Step D.1.

______________ ppm/%Δρ X ______________ %Δρ = ______________ ppm
A.6 D.1 D.2

D.3. Calculate the New desired critical boron concentration.

______________ ppm – ______________ ppm = ______________ ppm
B.3 D.2 D.3

D.4. Performed by ______________________________ Date/Time ________/________

D.5. Verify Step D calculations reviewed prior to Reactor Criticality.

Reactor Engineer ______________________________ Date/Time ________/________

E. Actual Critical Data

E.1. After achieving criticality, complete the following record:

E.1.a. CEA Positions

Group 1 ______________ inches

Group 2 ______________ inches

Group 3 ______________ inches

Group 4 ______________ inches

Group N ______________ inches

E.1.b. Reactor Coolant System boron concentration: ______________ ppm

E.1.c. Time at which criticality was attained: ______________

E.1.d. Reactor Coolant System average temperature (T_{AVG}): ____________ °F

E.2. Completed by ______________________________ Date/Time _______/_______

F. Review

F.1. Forward to Reactor Engineer for review.

NOTE: If the reactivity difference between actual and estimated critical conditions is greater than 0.5%$\Delta\rho$, then the Reactor Engineer will review the results with the Supervisor-Reactor Physics and determine the appropriate actions, if necessary.

F.2. Reviewed by ______________________________ Date/Time _______/_______
Supervisor-Reactor Performance Analysis

F.3. Reviewed by ______________________________ Date/Time _______/_______
Reactor Engineer

Facility: FCS JPM # NRC RA3 Task # 0066 K/A # 2.2.35 3.6 / 4.5

Title: Determine Technical Specification MODE of Operation

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Reactor Coolant System temperature is 200°F.
- Reactor Coolant System pressure is 150 psia.
- Reactor Coolant System boron concentration is 2180 ppm.
- Core Burnup is 1500 MWD/MTU.
- A Shutdown Cooling Pump is in operation.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- DETERMINE Fort Calhoun Station Technical Specification Reactor Operating Condition.
  - Refueling Boron Concentration _____ ppm.
  - Operating Mode _____.

Task Standard: Utilizing Technical Specifications and Core Operating Limits Report, determined Fort Calhoun Station is in Operating Mode 5, Refueling Shutdown Condition.

Required Materials: Fort Calhoun Station Technical Specifications, Amendment #283.
TDB-VI, Core Operating Limits Report, Rev. 42.

Validation Time: 5 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **Fort Calhoun Station Technical Specifications.**
- **TDB-VI, Core Operating Limits Report.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from Technical Specifications and the Core Operating Limits Report.**
Perform Step: 1	Refer to Technical Specifications for MODE definition.
Standard:	REFERRED to Technical Specification Definitions, Page 2 and DETERMINED that plant is either in MODE 4 or 5 depending on boron concentration.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√	Refer to Core Operating Limits Report to determine REFUELING BORON CONCENTRATION.
Standard:	REFERRED to Core Operating Limits Report and DETERMINED REFUELING BORON CONCENTRATION at 1500 MWD/MTU is **2160 ppm**.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√	Determine Plant Operational Mode based on Reactor Coolant System Boron Concentration.
Standard:	REFERRED to Technical Specification Definitions, Page 2 and DETERMINED that Plant is in **Operating Mode 5, Refueling Shutdown Condition** based on Reactor Coolant System Boron Concentration greater than REFUELING BORON CONCENTRATION.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **Reactor Coolant System temperature is 200°F.**
- **Reactor Coolant System pressure is 150 psia.**
- **Reactor Coolant System boron concentration is 2180 ppm.**
- **Core Burnup is 1500 MWD/MTU.**
- **A Shutdown Cooling Pump is in operation.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **DETERMINE Fort Calhoun Station Technical Specification Reactor Operating Condition.**
  - **Refueling Boron Concentration _____ ppm.**
  - **Operating Mode _____.**

TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of 1500 MWt. |

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ SHUTDOWN BORON CONCENTRATION but < REFUELING BORON CONCENTRATION.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ REFUELING BORON CONCENTRATION.

Refueling Operation

Any operation involving the shuffling, removal, or replacement of irradiated fuel outside of the reactor pressure vessel. The suspension of any REFUELING OPERATION shall not preclude completion of movement of a component to a safe, conservative position.

The Refueling Boron Concentration

A reactor coolant boron concentration of at least that specified in the CORE OPERATING LIMITS REPORT which corresponds to a shutdown margin of not less than 5% with all CEA's withdrawn.

Shutdown Boron Concentration

The boron concentration required to make the reactor subcritical by the amount defined in Section 2.10.

Refueling Outage or Refueling Shutdown

A plant outage or shutdown to perform refueling operations upon reaching the planned fuel depletion for a specific core.

Plant Operating Cycle

The time period from a REFUELING SHUTDOWN to the next REFUELING SHUTDOWN.

6. LINEAR HEAT RATE

The allowable peak linear heat rate is shown in Figure 3.

7. EXCORE MONITORING OF LHR

The allowable operation for power versus axial shape index for monitoring of LHR with excore detectors is shown in Figure 4.

8. PEAKING FACTOR LIMITS

The maximum full power value for the maximum radial peaking factor (F_R^T) is 1.732.

9. DNB MONITORING

The core operating limits for monitoring of DNB are provided in Figure 5. This figure provides the allowable power versus axial shape index for the cycle.

10. F_R^T AND CORE POWER LIMITATIONS

Core power limitations versus F_R^T are shown in Figure 6.

11. REFUELING BORON CONCENTRATION

The refueling boron concentration is required to ensure a shutdown margin of not less than 5% with all CEAs withdrawn. The refueling boron concentration must be at least **1,900 ppm** through the end of Cycle 27 operation and is valid until the beginning of core reload for Cycle 28.

Listed below in Table 2 are the refueling boron concentration values for Cycle 28 operations:

Table 2 - Refueling Boron Concentrations

Cycle Average Burnup (MWD/MTU)	Refueling Boron Concentration (ppm)
BOC	2,160
≥ 2,000	2,016
≥ 4,000	1,900

Facility: FCS JPM # NRC RA4 Task # 1269 K/A # 2.3.11 3.8 / 4.3

Title: Respond to Voids in the Reactor Coolant System

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A Small Break Loss of Coolant Accident (LOCA) has occurred.
- EOP-03, Loss of Coolant Accident, has been implemented.
- Reactor Coolant System (RCS) conditions:
  - RCS Pressure = 450 psia.
  - RCS T_{COLD} = 402°F.
- Pressurizer (PZR) conditions:
  - PZR Level [actual] = 60% and stable.
  - PZR Temperature = 456°F and stable.
- Reactor Vessel Level Monitoring System (RVLMS) is 83% and stable.
- Containment conditions:
  - Containment Safety Function is satisfied.
  - All Containment Ventilation Fans are operating.
  - Containment Pressure = 1.2 psig.
  - Containment Temperature = 118°F.
  - Containment Hydrogen concentration = 1.2%.
- RC-5, Pressurizer Quench Tank (PZR QT), conditions:
  - PZR QT Level = 70%.
  - PZR QT Pressure = 5 psig.
- HPSI Stop and Throttle has been performed for a LOCA.
- RCS and Pressurizer sample results are normal.
- The Technical Support Center (TSC) is activated.
- Use of EOP/AOP Attachment IC-14, RCS Void Elimination, has been unsuccessful in eliminating the RCS voids.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- IMPLEMENT OI-RC-12, Post Accident Venting of Noncondensable Gases from the Reactor Coolant System.
  - VENT from (CIRCLE one): Reactor Vessel Head PZR
  - VENT to (CIRCLE one): Containment PZR Quench Tank

Task Standard: Utilizing OI-RC-12, determined vent path from the Reactor Vessel Head to the Pressurizer Quench Tank is required.

Required Materials: OI-RC-12, Post Accident Venting of Noncondensable Gases from the Reactor Coolant System, Rev. 11.

Validation Time: 12 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **OI-RC-12, Post Accident Venting of Noncondensable Gases from the Reactor Coolant System.**
- **Steam Tables**

√ - **Check Mark Denotes Critical Step**

START TIME:	

Examiner Note:	**The following step is from OI-RC-12, Attachment 1, Prerequisites.**

Perform Step: 1
1, 2, & 3
	<u>PREREQUISITES</u>:
	• Procedure Revision Verification Revision No. _____ Date: _____
	• The Reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8).
	• The RCS is being maintained in a stable condition with the following:
	  ◦ Pressurizer (PZR) Level is between 49% and 93%
	  ◦ Charging flow is in operation
	  ◦ RCS subcooling is between 20°F and 200°F

Standard:	DETERMINED the following per the Initial Conditions: • Procedure Revision is as provided. • Reactor Coolant System T_{AVE} is ~402°F. • Pressurizer Level is 60%. • Charging flow is in operation (based on HPSI Stop and Throttle has been performed). • RCS subcooling is ~54°F.
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**The following steps are from OI-RC-12, Attachment 1, Procedure.**

NOTES

1. Determination of non-condensable gas in the Pressurizer can be indirectly determined by departure from saturation conditions. At any given PZR temperature, PZR pressure will be greater than saturation by an amount dependent upon the steam bubble volume and the amount of gas in the PZR steam space. The use of Figure 1 and 2 can assist with this determination.
2. If non-condensable gases are discovered in the RV Head, it is assumed that the Pressurizer has these same gases even if the gas volume can not be measured in the Pressurizer.
3. The objective of venting the Pressurizer is not to remove all gas, but to remove enough gas to maintain efficient pressure control.

CAUTION

Figure 1, Pressurizer Pressure with Non-Condensable Gas, can be used to determine an approximate volume of non-condensable gases in the Pressurizer. It should not be used as an absolute determination since this method is for determining large gas volumes and may not accurately reflect the true amount of gas.

Perform Step: 2 1 & all bullets	IF one or more of the following conditions are present in the PZR, THEN determine the venting path per Attachment 2: • Figure 1 indicates the presence of non-condensable gases • Departure from saturation • Sluggish pressure control • Sampling results indicate non-condensable gases
Standard:	IDENTIFIED that the Pressurizer does not indicate a departure from saturated conditions and no other conditions warrant Pressurizer venting.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√ 2	Determine if bubble exists in the RV Head by monitoring RV level less than 100% via the Reactor Vessel Level Monitoring System (RVLMS), THEN determine the venting path per Attachment 2.
Standard:	IDENTIFIED that a bubble exists in the Reactor Vessel Head and CIRCLED Vent <u>from</u>: **Reactor Vessel Head**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following step is from OI-RC-12, Attachment 2, Prerequisites.**

Perform Step: 4
1, 2, 3, 4, 5, & 6
	<u>PREREQUISITES</u>:
	• Procedure Revision Verification Revision No. _____ Date: _____
	• The reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8).
	• Containment Isolation has been verified per EOP Safety Function Status Check.
	• All available Containment Ventilation Units are in operation:
	  • VA-3A, Cntmt Vent Fan
	  • VA-3B, Cntmt Vent Fan
	  • VA-7C, Cntmt Vent Fan
	  • VA-7D, Cntmt Vent Fan
	• The RCS is being maintained in a stable condition with the following:
	  • Pressurizer (PZR) Level is between 49% and 93%
	  • Charging flow is in operation
	  • RCS subcooling is between 20°F and 200°F
	• TSC has been activated.

Standard:	DETERMINED the following per the Initial Conditions: • Procedure Revision is as provided. • Reactor Coolant System T_{AVE} is ~402°F. • Containment Isolation has been verified. • All Containment Ventilation Units are in operation. • Pressurizer Level is 60%. • Charging flow is in operation (based on HPSI Stop and Throttle has been performed). • RCS subcooling is ~54°F. • Technical Support Center has been activated.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps are from OI-RC-12, Attachment 2, Procedure.**

NOTES

1. If Containment H_2 is approaching or at combustible levels, with PRC approval, the H_2 levels may be decreased using the Hydrogen Purge System.
2. Since venting to an intact Pressurizer Quench Tank (PQT) does not cause a release to Containment, this flow path is preferred when available.

CAUTIONS

1. When venting H_2 to Containment directly or through the PQT Rupture Disk, the Containment H_2 concentration will increase.
2. If large quantities of H_2 must be vented, H_2 may approach combustible levels.
3. Venting the RV Head should have priority over Containment H_2 limits only if the potential for loss of Core Cooling exists.
4. Venting the PZR should have priority over Containment H_2 limits only if the bubble is seriously interfering with the ability to maintain pressure control.

Perform Step: 5
1, 1.a, & all bullets
	IF one or more of the following conditions exist, THEN the Containment vent path should be used per the following:
	• Verify:
	  ◦ There is no water in the RC-5, PQT AND DC Bus 1 electrical power source is available to RCGVS Valves
	  ◦ Large quantities of gas need to be vented
	  ◦ Rapid venting is required
	  ◦ The potential for loss of core cooling exists
	  ◦ There is serious interference with the ability to maintain pressure control

Standard:	IDENTIFIED that the Containment vent path is NOT preferred.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√
2, 2.a, & all bullets
	IF the following conditions exist, THEN RC-5, Pressurizer Quench Tank (PQT) vent path should be used per the following:
	• Verify:
	  • There is water in the PQT AND DC Bus 2 electrical power source is available to RCGVS Valves
	  • Small quantities of gas need to be vented
	  • Rapid venting is not required

Standard:	IDENTIFIED the following: • A bubble exists in the Reactor Vessel Head. (**NOT critical**) • Control power exists to the valves. (**NOT critical**) • Quench Tank is the preferred venting path. (**NOT critical**) • Attachment 4, Venting RV Head to the Pressurizer Quench Tank. (**NOT critical**) CIRCLED VENT to: **PZR Quench Tank (critical)**
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: Given the following conditions:

- A Small Break Loss of Coolant Accident (LOCA) has occurred.
- EOP-03, Loss of Coolant Accident, has been implemented.
- Reactor Coolant System (RCS) conditions:
  - RCS Pressure = 450 psia.
  - RCS T_{COLD} = 402°F.
- Pressurizer (PZR) conditions:
  - PZR Level [actual] = 60% and stable.
  - PZR Temperature = 456°F and stable.
- Reactor Vessel Level Monitoring System (RVLMS) is 83% and stable.
- Containment conditions:
  - Containment Safety Function is satisfied.
  - All Containment Ventilation Fans are operating.
  - Containment Pressure = 1.2 psig.
  - Containment Temperature = 118°F.
  - Containment Hydrogen concentration = 1.2%.
- RC-5, Pressurizer Quench Tank (PZR QT), conditions:
  - PZR QT Level = 70%.
  - PZR QT Pressure = 5 psig.
- HPSI Stop and Throttle has been performed for a LOCA.
- RCS and Pressurizer sample results are normal.
- The Technical Support Center (TSC) is activated.
- Use of EOP/AOP Attachment IC-14, RCS Void Elimination, has been unsuccessful in eliminating the RCS voids.

INITIATING CUE: The Control Room Supervisor directs you to PERFORM the following:

- IMPLEMENT OI-RC-12, Post Accident Venting of Noncondensable Gases from the Reactor Coolant System.
  - VENT from (CIRCLE one):
    Reactor Vessel Head PZR
  - VENT to (CIRCLE one):
    Containment PZR Quench Tank

Fort Calhoun Station
Unit No. 1

OI-RC-12

OPERATING INSTRUCTION

POST ACCIDENT VENTING OF NONCONDENSABLE GASES FROM THE REACTOR COOLANT SYSTEM

Change No.	EC 65359
Reason for Change	Editorial correction for procedure reference.
Requestor	Ch Hayes
Preparer	K. Bessey
Issue Date	02-05-15 3:00pm

POST ACCIDENT VENTING OF NONCONDENSABLE GASES FROM THE REACTOR COOLANT SYSTEM

SAFETY RELATED

ATT PURPOSE PAGE

PRECAUTIONS

1. Actions to eliminate non-condensable gases may cause significant changes in RCS pressure. RCS pressure and temperature limits shall be maintained per Technical Specifications 2.1.2, TDB-III.7.a and EOP Attachment PC-12.

2. Non-condensable gases mentioned in this procedure have a high probability of containing various amounts of Hydrogen (H_2) gas and should be carefully considered while performing each step.

3. When venting the RCS, a pressure drop should be expected. The severity of the pressure drop is determined by vent location, Charging Pump availability AND initial RCS pressure and temperature. Venting may need to be temporarily terminated to restore RCS temperature and pressure conditions.

PRECAUTIONS (continued)

4. Venting H_2 to Containment directly OR through the Pressurizer Quench Tank Rupture Disk will cause an increase in the Containment H_2 concentration. If large quantities of H_2 must be vented to Containment, venting operations may need to be terminated early or continued based upon the following criteria:

   - Venting the RV has priority over Containment H_2 limits if there is a potential for an interruption in core cooling
   - Venting the Pressurizer has priority over Containment H_2 limits if pressure control is seriously affected by the accumulation of non-condensable gases
   - If containment H_2 concentration reaches 4%, venting should be terminated. With PRC approval the Hydrogen Purge System may be used to decrease the H_2 concentration to less than 4%

5. If the following conditions are present and possible void formation is indicated, EOP/AOP Attachment IC-14, RCS Void Elimination, should be reviewed:

   - Letdown flow is greater than charging flow
   - Pressurizer level increases significantly more than expected while pressurizer sprays are operating
   - RV Level Monitoring System(RVLMS) indicates RV voiding
   - Erratic Steam Generator (S/G) differential pressure
   - Reactor Coolant Pump (RCP) vibration or erratic motor current
   - RCS subcooling less than 20°F.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specifications:

   - 1.1.2, Reactor Coolant System Pressure SL
   - 2.1.2, Reactor Coolant System, Heatup and Cooldown
   - 2.1.8, Reactor Coolant System Vents

2. USAR:

   - 4.0, Reactor Coolant System
   - 14.15, Loss of Coolant Accident
   - 14.17, Generation of Hydrogen in Containment
   - 7.4, Regulating Systems
   - 7.5, Instrumentation Systems
   - 9.2, Chemical and Volume Control System

REFERENCES/COMMITMENT DOCUMENTS (continued)

3. Technical Data Book (TDB):

- TDB-III.1.a, Temperature Correction for Pressurizer Level Indicators LI-101X/Y
- TDB-III.1.b - e, Pressurizer Level Correction (Density Effects Correction), L-101 (120°F, 200°F, 250°F, 300°F)
- TDB-III.2, Actual Pressurizer Level vs. Indicated Level in the Pressurizer, LI-106
- TDB-III.5, Pressurizer Level Program
- TDB-III.6, Pressurizer Pressure Control Program
- TDB-III.7.a, RCS Pressure and Temperature Limits
- TDB-III.27, Decay Heat Vs. Time After Shutdown and Time to Boil Curves and Tables
- TDB-III.28, Reactor Vessel Level Monitoring System
- TDB-IX, Figure 5-1, Fort Calhoun Station Unit 1 Composite P/T Limits

4. NUREG-0737:

- Item II.B.1
- Item II.K.2.17

5. Combustion Engineering Report CEN-199

6. OPPD letter to the NRC (LIC-82-221) dated 6-1-82

7. Drawings

Drawings	File	Description
D-4078	20663	Reactor Coolant Gas Vent System
E-23866-210-110, Sh 1	10475	Reactor Coolant System Flow Diagram
E-23866-210-110, Sh 1A	42107	Reactor Coolant System Flow Diagram

APPENDICES

Attachment 1 - Determination of Need to Vent

PREREQUISITES

(√) INITIALS

1. Procedure Revision Verification

   Revision No. 11 Date: today ____ γ

2. The Reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). ____

3. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) Level is between 49% and 93% ____
   - Charging flow is in operation ____
   - RCS Subcooling is between 20°F and 200°F ____ ____

Attachment 1 - Determination of Need to Vent

PROCEDURE

(√) INITIALS

NOTES

1. Determination of non-condensable gas in the Pressurizer can be indirectly determined by departure from saturation conditions. At any given PZR temperature, PZR pressure will be greater than saturation by an amount dependent upon the steam bubble volume and the amount of gas in the PZR steam space. The use of Figure 1 and 2 can assist with this determination.

2. If non-condensable gases are discovered in the RV Head, it is assumed that the Pressurizer has these same gases even if the gas volume can not be measured in the Pressurizer.

3. The objective of venting the Pressurizer is not to remove all gas, but to remove enough gas to maintain efficient pressure control.

CAUTION

Figure 1, Pressurizer Pressure with Non-Condensable Gas, can be used to determine an approximate volume of non-condensable gases in the Pressurizer. It should not be used as an absolute determination since this method is for determining large gas volumes and may not accurately reflect the true amount of gas.

1. IF one or more of the following conditions are present in the PZR, THEN determine the venting path per Attachment 2:

   - Figure 1 indicates the presence of non-condensable gases ___
   - Departure from saturation ___
   - Sluggish pressure control ___
   - Sampling results indicate non-condensable gases ___ _______

2. Determine if bubble exists in the RV Head by monitoring RV level less than 100% via the Reactor Vessel Level Monitoring System (RVLMS), THEN determine the venting path per Attachment 2. _______

Completed by_______________________________________Date/Time______/_____

Attachment 2 - Determination of a Vent Path

PREREQUISITES

(√) INITIALS

1. Procedure Revision Verification

   Revision No. 11 Date: today X

2. The reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). ______

3. Containment Isolation has been verified per EOP Safety Function Status Check. ______

4. All available Containment Ventilation Units are in operation:

   - VA-3A, Cntmt Vent Fan ___
   - VA-3B, Cntmt Vent Fan ___
   - VA-7C, Cntmt Vent Fan ___
   - VA-7D, Cntmt Vent Fan ___ ______

5. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) level is between 49% and 93% ___
   - Charging flow is in operation ___
   - RCS Subcooling is between 20°F and 200°F ___ ______

6. TSC has been activated.

Attachment 2 - Determination of a Vent Path

PROCEDURE (√) INITIALS

NOTES

1. If Containment H_2 is approaching or at combustible levels, with PRC approval, the H_2 levels may be decreased using the Hydrogen Purge System.

2. Since venting to an intact Pressurizer Quench Tank (PQT) does not cause a release to Containment, this flow path is preferred when available.

CAUTIONS

1. When venting H_2 to Containment directly or through the PQT Rupture Disk, the Containment H_2 concentration will increase.

2. If large quantities of H_2 must be vented, H_2 may approach combustible levels.

3. Venting the RV Head should have priority over Containment H_2 limits only if the potential for loss of Core Cooling exists.

4. Venting the PZR should have priority over Containment H_2 limits only if the bubble is seriously interfering with the ability to maintain pressure control.

1. IF one or more of the following conditions exist,
THEN the Containment vent path should be used per the following:

   a. Verify:

      - There is no water in RC-5, PQT AND DC Bus 1 electrical power source is available to RCGVS valves ___
      - Large quantities of gas need to be vented ___
      - Rapid venting is required ___
      - The potential for loss of core cooling exists ___
      - There is serious interference with the ability to maintain pressure control ___ ______

Attachment 2 – Determination of a Vent Path

PROCEDURE (continued) (√) INITIALS

1.
 b. Contact TSC for assistance in vent path alignment. ______

 c. GO TO Step 3. ______

2. IF the following conditions exist,
THEN RC-5, Pressurizer Quench Tank (PQT) vent path should be used per the following:

 a. Verify:

- There is water in the PQT AND DC Bus 2 electrical power source is available to RCGVS Valves ___
- Small quantities of gas need to be vented ___
- Rapid venting is not required ___ ______

 b. Contact TSC for assistance in vent path alignment. ______

3. Perform the applicable Attachment as follows:

- Attachment 3 Venting RV Head to Containment ___
- Attachment 4 Venting RV Head to the Pressurizer Quench Tank ___
- Attachment 5 Venting PZR to Containment ___
- Attachment 6 Venting PZR to the Pressurizer Quench Tank ___ ______

Completed by____Date/Time______/____

Attachment 3 - Venting RV Head to Containment

PREREQUISITES (√) INITIALS

1. Procedure Revision Verification

   Revision No.______ Date:__________ ________

2. The reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). ________

3. Containment Isolation has been verified using the EOP/AOP Safety Function Status Checks. ________

4. All available Containment Ventilation Units are in operation:

   - VA-3A, Cntmt Vent Fan ___
   - VA-3B, Cntmt Vent Fan ___
   - VA-7C, Cntmt Vent Fan ___
   - VA-7D, Cntmt Vent Fan ___ ________

5. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) level is between 49% and 93% ___
   - Charging flow is in operation ___
   - RCS Subcooling is between 20°F and 200°F ___ ________

6. Containment Hydrogen (H_2) concentration is being monitored and is less than 3%. ________

7. Power is available to the RCGVS. ________

PROCEDURE

i	NOTE Refer to Figure 4, Reactor Coolant Gas Vent System.	i

1. Permission to vent the RV Head via the RCGVS has been granted.

   ____
   Mgr FCS

2. Energize PZR Heaters as necessary to minimize the RCS pressure drop. ________

3. Open HCV-181, RCG Vent Header Release Valve (AI-65A). ________

Attachment 3 – Venting RV Head to Containment

PROCEDURE (continued) (√) INITIALS

NOTE

Venting duration for the RV Head should be long enough to remove the entire gas bubble from the RV Head. Figure 3 provides guidance for determination.

CAUTIONS

1. A pressure drop should be expected. The severity of the pressure drop is determined by vent location, Charging Pump availability and initial RCS pressure and temperature.
2. Venting may need to be temporarily terminated to restore the required RCS temperature and pressure conditions.
3. If large quantities of H_2 must be vented to Containment, venting operations should be terminated or continued based upon the following criteria.
   - Venting the RV Head takes priority over the 3% Containment H_2 limits only if there is a potential for an interruption in Core Cooling
   - If Containment H_2 concentration reaches 4%, venting should be terminated.
   - With PRC approval the Hydrogen Purge System may be used to decrease H_2 concentration to less than 4%

4. Open one of the following valves:
   - HCV-176, Reactor Head RCG Vent Valve (AI-65A) ___
   - HCV-177, Reactor Head RCG Vent Valve (AI-65B) ___ ______
5. Record Start Time ______. ______
6. WHEN one of the following conditions have been met, THEN venting should be terminated:
   - RCS Subcooling is 20°F ___
   - PZR Level drops to 49% Actual ___
   - PZR Level increases to 93% Actual ___
   - Containment H_2 concentration reaches 3% ___
   - Ten minutes has elapsed ___ ______

Attachment 3 – Venting RV Head to Containment

PROCEDURE (continued) (√) INITIALS

7. Close the valve opened in Step 4:

- HCV-176 (AI-65A) ___
- HCV-177 (AI-65B) ___ ________

____
Ind Verif

8. Close HCV-181 (AI-65A). ________

____
Ind Verif

9. Record the following Data:

- RCS Pressure ________PSIA
- T_c ________°F
- Vent Duration ________min
- Containment H_2 Conc. ________% ________

10. Restore RCS Pressure AND PZR Level to desired values. ________

11. Evaluate the effectiveness of venting. (Refer to OI-RC-12, Attachment 1). ________

12. IF necessary,
THEN repeat venting by reperforming Steps 2 through 11. ________

Completed by____Date/Time______/____

Attachment 4 - Venting RV Head to the Pressurizer Quench Tank

PREREQUISITES (√) INITIALS

1. Procedure Revision Verification

   Revision No.______ Date:_________ _______

2. The Reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). _______

3. Containment Isolation has been verified. _______

4. All available Containment Ventilation Units are in operation:

   - VA-3A, Cntmt Vent Fan ___
   - VA-3B, Cntmt Vent Fan ___
   - VA-7C, Cntmt Vent Fan ___
   - VA-7D, Cntmt Vent Fan ___ _______

5. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) level is between 49% and 93% ___
   - Charging flow is in operation ___
   - RCS Subcooling is between 20°F and 200°F ___ _______

6. Containment Hydrogen (H_2) concentration is being monitored by VA-81A/B Hydrogen Analyzer AND is less than 3%. _______

7. Power is available to the RCGVS per OI-RC-10, Reactor Coolant Gas Vent System. _______

PROCEDURE

i	NOTE Refer to Figure 4, Reactor Coolant Gas Vent System.	i

1. Permission to vent the RV Head via the RCGVS has been granted. _______ Mgr FCS

2. Energize the PZR Heaters as necessary to minimize the RCS pressure drop. _______

3. Open HCV-180, RCG Vent Header Release Valve (AI-65B). _______

Attachment 4 - Venting RV Head to the Pressurizer Quench Tank

PROCEDURE (continued) (√) INITIALS

NOTES

1. Pressurizer Quench Tank (PQT) pressure should be monitored during venting to prevent lifting the Relief Valve at 70 psig OR rupturing the Rupture Disk at 75 psig.
2. Venting duration for the RV Head should be long enough to remove the entire gas bubble from the RV Head.

CAUTIONS

1. A pressure drop should be expected. The severity of the pressure drop is determined by vent location, Charging Pump availability and initial RCS pressure and temperature.
2. Venting may need to be temporarily terminated to restore the required RCS temperature and pressure conditions.

4. Open one of the following valves:
   - HCV-176, Reactor Head RCG Vent Valve (AI-65A) ___
   - HCV-177, Reactor Head RCG Vent Valve (AI-65B) ___ ______
5. Record Start Time______. ______
6. WHEN one of the following conditions have been met, THEN venting should be terminated:
   - RCS Subcooling is 20°F
   - PZR level drops to 49% Actual ___
   - PZR level increases to 93% Actual ___
   - Ten minutes has elapsed ___
   - Insufficient Quench Tank level (< 67%) ___
   - Containment H_2 reaches 3% ___ ______
7. Close the following valves opened in Step 4:
   - HCV-176 (AI-65A) ___
   - HCV-177 (AI-65B) ___ ______

______
Ind Verif

Attachment 4 - Venting RV Head to the Pressurizer Quench Tank

PROCEDURE (continued) (√) INITIALS

8. Close HCV-180 (AI-65B). ________

____
Ind Verif

9. Record the following Data:

- RCS Pressure ______PSIA
- T_c ______°F
- Vent Duration ______min
- Containment H_2 Conc. ______% ________

10. Restore RCS pressure AND PZR level to required values per applicable EOP. ________

11. Evaluate the effectiveness of venting. (Refer to OI-RC-12, Attachment 1.) ________

12. IF necessary,
THEN repeat venting by reperforming Steps 2-11. ________

Completed by________________________________Date/Time______/_____

Attachment 5 - Venting PZR to Containment

PREREQUISITES (√) INITIALS

1. Procedure Revision Verification

   Revision No.______ Date:______ _______

2. The Reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). _______

3. Containment Isolation has been verified. _______

4. All available Containment Ventilation Units are in operation:

   - VA-3A, Cntmt Vent Fan ___
   - VA-3B, Cntmt Vent Fan ___
   - VA-7C, Cntmt Vent Fan ___
   - VA-7D, Cntmt Vent Fan ___ _______

5. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) Level is between 49% and 93% ___
   - Charging flow is in operation ___
   - RCS Subcooling is between 20°F and 200°F ___ _______

6. Containment Hydrogen (H_2) concentration is being monitored and is less than 3%. _______

7. Power is available to the RCGVS. _______

PROCEDURE

i	**NOTE** Refer to Figure 4, Reactor Coolant Gas Vent System.	i

1. Permission to vent the Pressurizer via the RCGVS has been granted.

   ____
   Mgr FCS

Attachment 5 - Venting PZR to Containment

PROCEDURE (continued) (√) INITIALS

CAUTIONS

1. A pressure drop should be expected. The severity of the pressure drop is determined by vent location, Charging Pump availability and RCS initial pressure and temperature.

2. Venting may need to be temporarily terminated to restore the required RCS temperature and pressure conditions.

3. If large quantities of H_2 must be vented to Containment, venting operations should be terminated OR continued based upon the following criteria:

    - Venting the PZR takes priority over the 3% Containment hydrogen limit only if Pressure Control is seriously affected by the accumulation of non-condensable gases.
    - If Containment H_2 concentration reaches 4%, venting should be terminated. With PRC approval the Hydrogen Purge System may be used to decrease H_2 concentration to less than 4%.

2. Energize the PZR Heaters as necessary to minimize the RCS pressure drop. ______

3. Open HCV-181, RCG Vent Header Release Valve (AI-65A). ______

4. Open one of the following valves:

    - HCV-178, Pressurizer RCG Vent Valve (AI-65A) __
    - HCV-179, Pressurizer RCG Vent Valve (AI-65B) __ ______

5. Record Start Time ______. ______

Attachment 5 - Venting PZR to Containment

PROCEDURE (continued) (√) INITIALS

NOTE

Refer to Figure 2, Pressurizer Bubble Vs. Pressurizer Level, and Figure 3, Venting Duration from PZR of $T=T_{sat}$.

6. WHEN one of the following conditions have been met, THEN venting should be terminated:

   - RCS Subcooling is 20°F
   - PZR level drops to 49% Actual ___
   - PZR level increases to 93% Actual ___
   - Containment H_2 concentration reaches 3% ___
   ___ ______

7. Close the valve opened in Step 4:

   - HCV-178 (AI-65A) ___
   - HCV-179 (AI-65B) ___ ______

   ______
   Ind Verif

8. Close HCV-181 (AI-65A). ______

   ______
   Ind Verif

9. Record the following data:

   - RCS Pressure ______PSIA
   - T_c ______°F
   - Vent Duration ______min
   - Containment H_2 Conc. ______% ______

10. Restore RCS pressure AND PZR level to desired values. ______

Attachment 5 - Venting PZR to Containment

PROCEDURE (continued) (√) INITIALS

11. Evaluate the effectiveness of venting. (Refer to OI-RC-12, Attachment 1.) ______

12. IF necessary,
THEN repeat venting by reperforming Steps 2-11. ______

Completed by____Date/Time______/____

Attachment 6 - Venting the PZR to Pressurizer Quench Tank

PREREQUISITES (√) INITIALS

1. Procedure Revision Verification

   Revision No.______ Date:_______ _______

2. The Reactor is subcritical with a Tave less than 515°F (Ref. Technical Specification 2.1.8). _______

3. Containment Isolation has been verified. _______

4. All available Containment Ventilation Units are in operation:

   - VA-3A, Cntmt Vent Fan ___
   - VA-3B, Cntmt Vent Fan ___
   - VA-7C, Cntmt Vent Fan ___
   - VA-7D, Cntmt Vent Fan ___ _______

5. The RCS is being maintained in a stable condition with the following:

   - Pressurizer (PZR) Level is between 49% and 93% ___
   - Charging flow is in operation ___
   - RCS Subcooling is between 20°F and 200°F ___ _______

6. Containment Hydrogen (H_2) concentration is being monitored and is less than 3%. _______

7. Power is available or restored to the RCGVS. _______

PROCEDURE

i	**NOTE** Refer to Figure 4, Reactor Coolant Gas Vent System.	i

1. Permission to vent the Pressurizer via the RCGVS has been granted.

   ____
   Mgr FCS

2. Energize PZR Heaters as necessary to minimize the RCS pressure drop. _______

Attachment 6 - Venting the PZR to Pressurizer Quench Tank

PROCEDURE (continued) (√) INITIALS

3. Open HCV-180, RCG Vent Header Release Valve (AI-65B). ______

NOTES

Pressurizer Quench Tank (PQT) pressure should be monitored during venting to prevent lifting the Relief Valve at 70 psig OR rupturing the Rupture Disk at 75 psig.

CAUTIONS

1. A pressure drop should be expected. The severity of the pressure drop is determined by vent location, Charging Pump availability and initial RCS pressure and temperature.
2. Venting may need to be temporarily terminated to restore the required RCS temperature and pressure conditions.

4. Open one of the following valves:
   - HCV-178, Pressurizer RCG Vent Valve (AI-65A) ___
   - HCV-179, Pressurizer RCG Vent Valve (AI-65B) ___ ______

5. Record Start Time ______. ______

NOTE

Refer to Figure 2, Pressurizer Bubble Vs. Pressurizer Level, and Figure 3, Venting Duration from PZR at $T=T_{sat}$.

6. WHEN one of the following conditions have been met, THEN venting should be terminated:
   - RCS Subcooling is 20°F ___
   - PZR level drops to 49% Actual ___
   - PZR level increases to 93% Actual ___
   - Ten minutes has elapsed ___
   - Insufficient Quench Tank level (< 67%) ___
   - Containment H_2 concentration reaches 3% ___ ______

Attachment 6 - Venting the PZR to Pressurizer Quench Tank

PROCEDURE (continued) (√) INITIALS

7. Close the following valve opened in Step 4:

   - HCV-178 (AI-65A) ___
   - HCV-179 (AI-65B) ___ ________

   ____
   Ind Verif

8. Close HCV-180 (AI-65B). ________

   ____
   Ind Verif

9. Record the following data:

   - RCS Pressure ________PSIA
   - T_c ________°F
   - Vent Duration ________min
   - Containment H_2 Conc. ________% ________

10. Restore RCS Pressure AND PZR Level to desired values. ________

11. Evaluate the effectiveness of venting. (Refer to OI-RC-12, Attachment 1.) ________

12. If necessary, repeat venting by reperforming Steps 2-11. ________

Completed by__________________________________Date/Time______/_____

Figure 1 - Pressurizer Pressure with Non-Condensable Gas

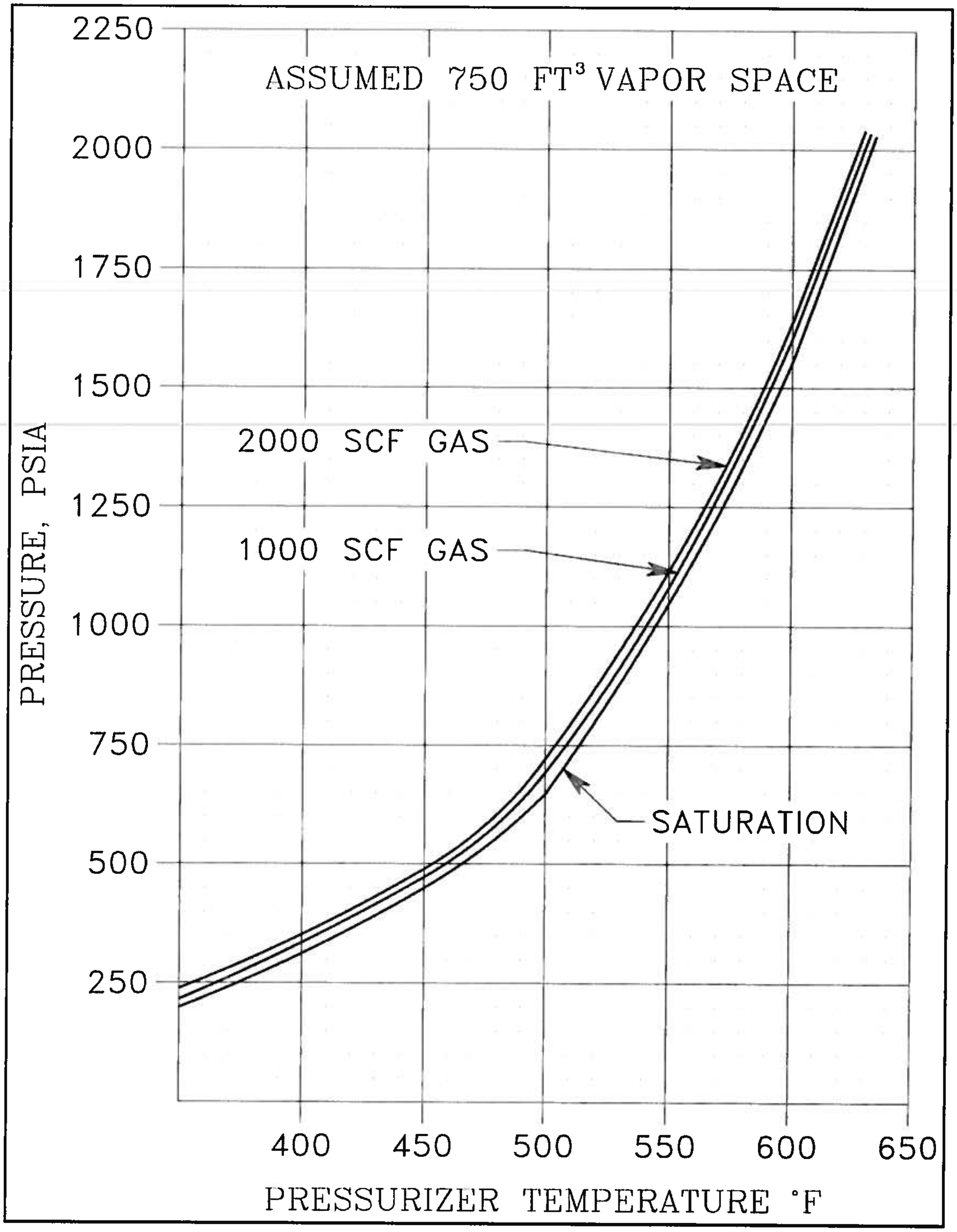
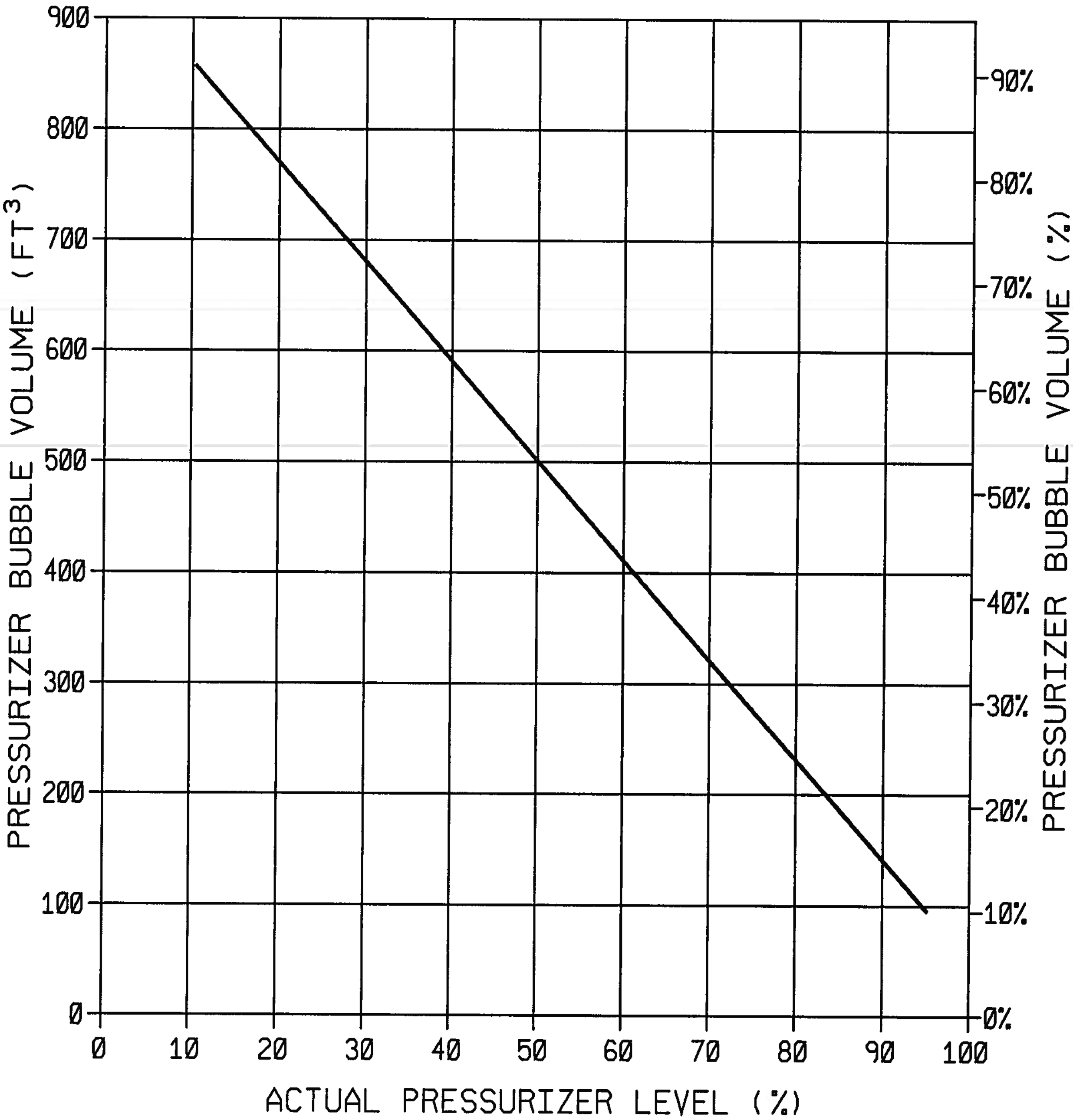


Figure 2 - Pressurizer Bubble Vs Pressurizer Level



NOTE: THIS FIGURE IS VALID ONLY OVER THE RANGE OF 10.4 % ≤ PRESSURIZER LEVEL (%) ≤ 95.4 %

Figure 3 - Venting Duration from PZR at $T=T_{SAT}$

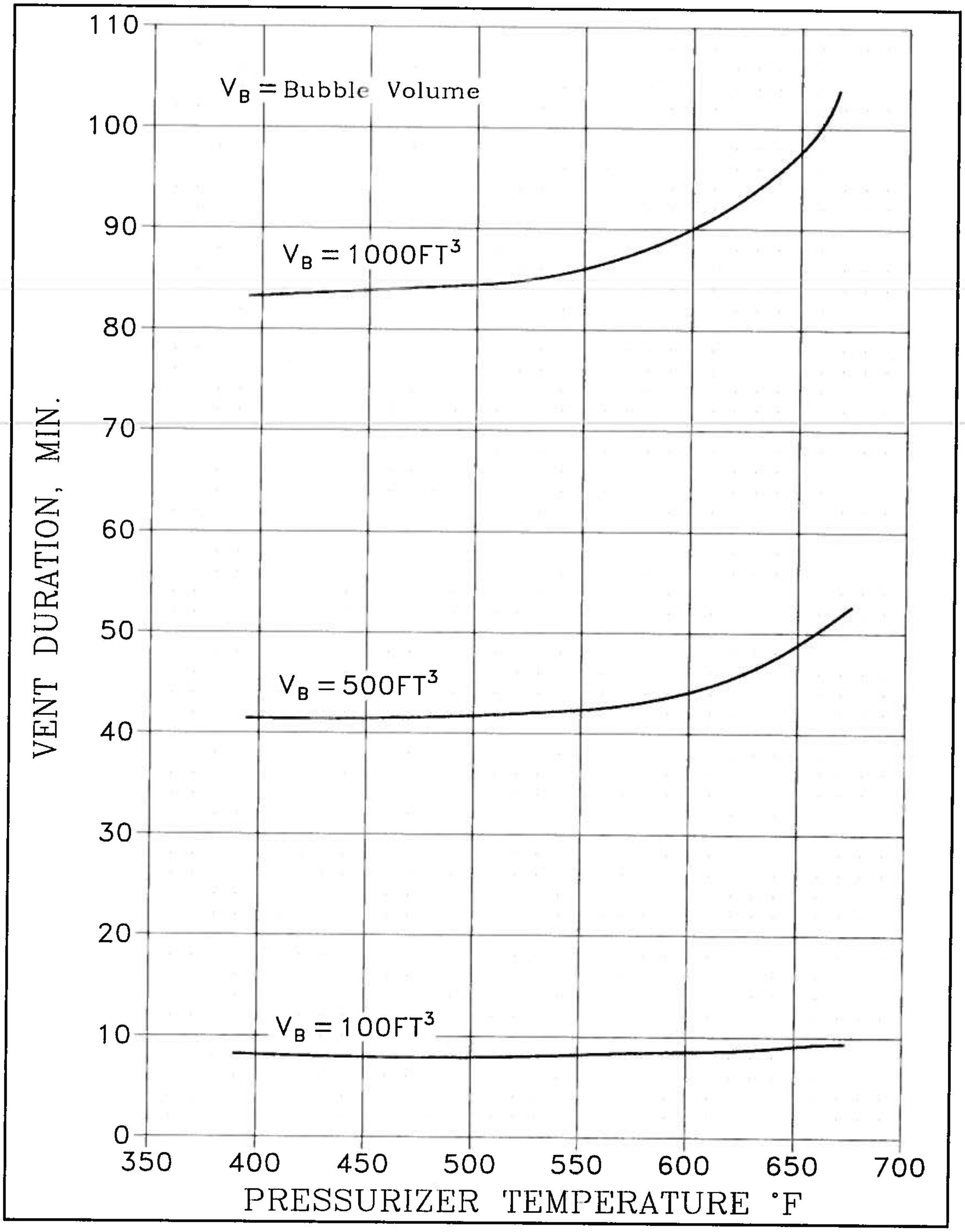
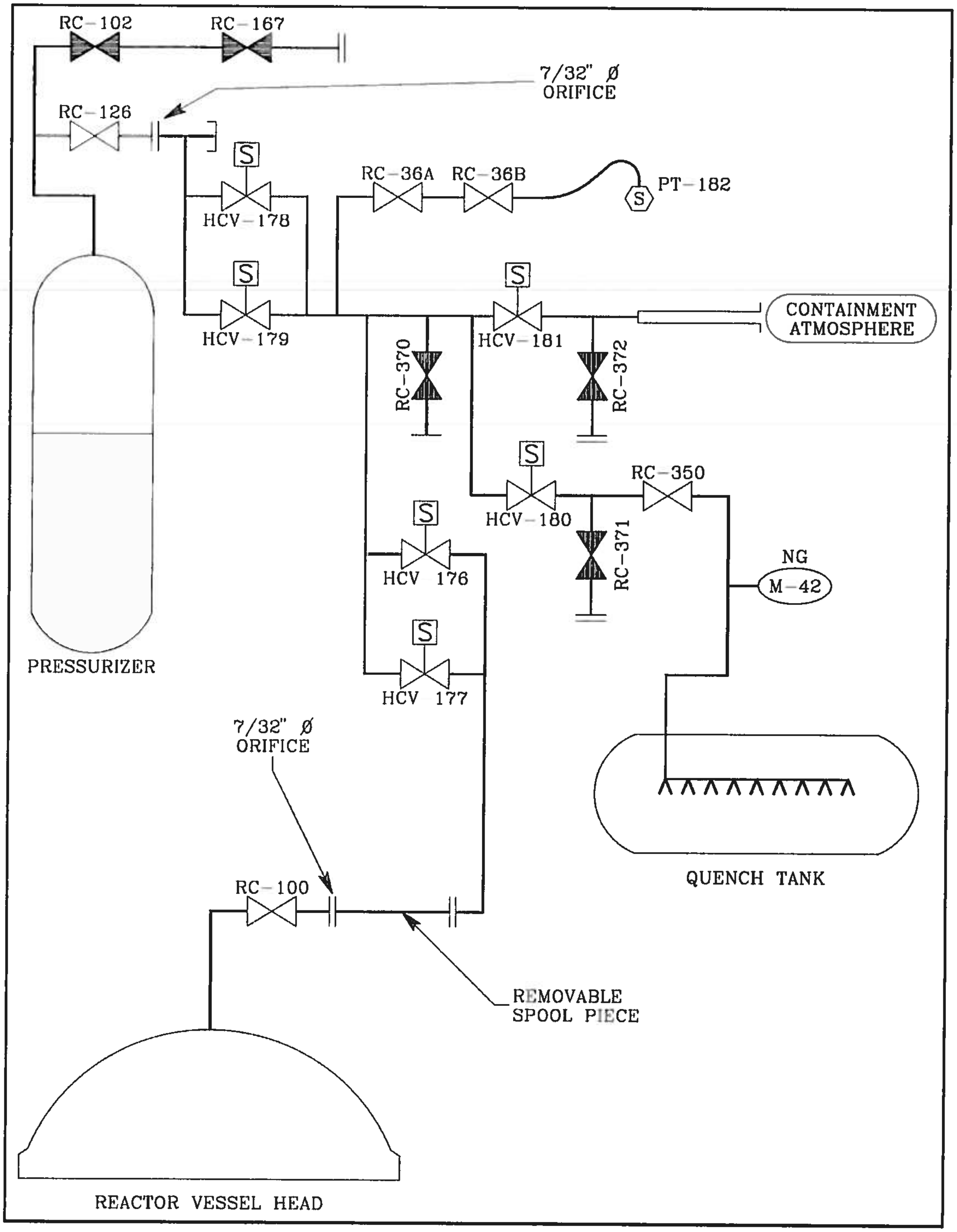


Figure 4 - Reactor Coolant Gas Vent System



Facility: FCS JPM # NRC SA1 Task # 1363 K/A # 2.1.25 3.9 / 4.2

Title: Perform an Alternate Decay Heat Removal Method Determination

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Plant was operating for 6 weeks at 100% power when a Reactor Coolant Pump seal failed.
- AOP-19, Loss of Shutdown Cooling, has been entered 7 days after shutdown.
- The Pressurizer manway has been removed.
- HCV-347, Shutdown Cooling Loop 2 Isolation Valve, is closed and cannot be reopened.
- High Pressure Safety Injection (HPSI) Pump SI-2A is available with flow of 300 gpm at 250 psia discharge pressure.

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- PERFORM AOP-19, Loss of Shutdown Cooling, Attachment D, Alternate Decay Heat Removal Method Determination.
- CIRCLE the appropriate Alternate Decay Heat Removal Attachment on Attachment D (indicate your decision path on Attachment D).

Task Standard: Utilizing AOP-19, determined Reactor Coolant System pressure boundary was not intact, Reactor Vessel Head was installed, no Shutdown Cooling flow was available, HPSI flow was available but insufficient, and identified Attachment E as the Alternate Decay Heat Removal Method.

Required Materials: AOP-19, Loss of Shutdown Cooling, Rev. 18.

Validation Time: 11 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **AOP-19, Loss of Shutdown Cooling.**
    - **Attachment D, Alternate Decay Heat Removal Method Determination.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from AOP-19, Attachment D.**
Examiner Note:	**REFER to Answer Key to follow Attachment D flowpath.**
Perform Step: 1√ 1	IS RCS Pressure Boundary Intact?
Standard:	DETERMINED answer was **NO** on Attachment D and PROCEEDED to next box.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√ 2	IS Reactor Vessel Head on?
Standard:	DETERMINED answer was **YES** on Attachment D and PROCEEDED to next box.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3√ 3	Is SDC Discharge Available?
Standard:	DETERMINED answer was **NO** on Attachment D and PROCEEDED to next box.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4√ 4	Is HPSI Discharge Available?
Standard:	DETERMINED answer was **YES** on Attachment D and PROCEEDED to next box.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5√ 5	Is Sufficient Injection Available?
Standard:	PERFORMED the following: • REFERRED to note (*) and DETERMINED plant operation at 100% power for greater than 30 days. • DETERMINED Time after Shutdown was 7 days ago. • DETERMINED Required (gpm) is >310 gpm but < 385 gpm. • DETERMINED Sufficient Injection Flow NOT available and PROCEEDED to the next box.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 6	<u>GO TO</u> Attachment E Alternate Decay Heat Removal by Boiling.
Standard:	DETERMINED Attachment E, Alternate Decay Heat Removal by Boiling is the appropriate Attachment.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: Given the following conditions:

- The plant was operating for 6 weeks at 100% power when a Reactor Coolant Pump seal failed.
- AOP-19, Loss of Shutdown Cooling, has been entered 7 days after shutdown.
- The Pressurizer manway has been removed.
- HCV-347, Shutdown Cooling Loop 2 Isolation Valve, is closed and cannot be reopened.
- High Pressure Safety Injection (HPSI) Pump SI-2A is available with flow of 300 gpm at 250 psia discharge pressure.

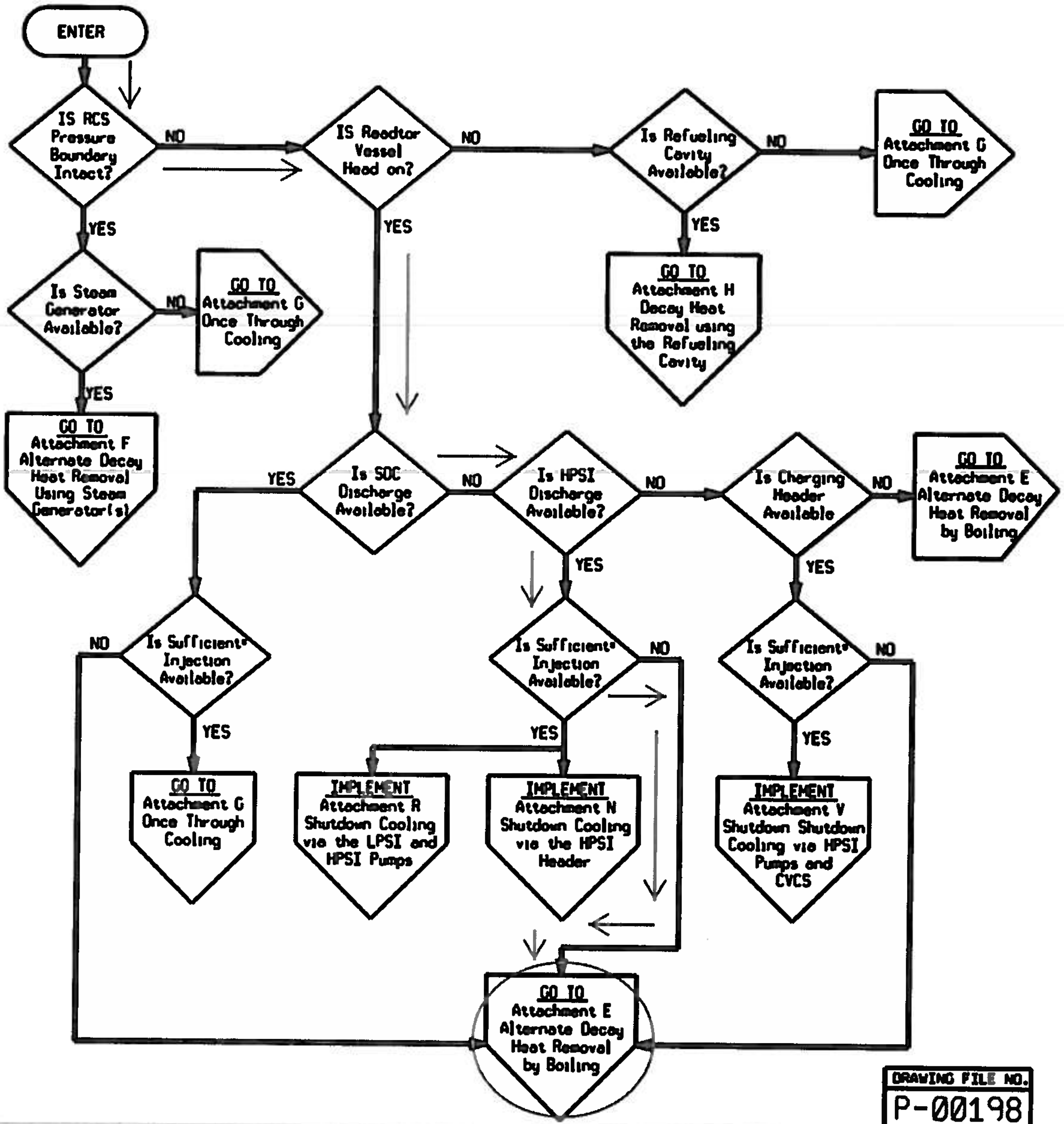
INITIATING CUE: The Shift Manager directs you to PERFORM the following:

- PERFORM AOP-19, Loss of Shutdown Cooling, Attachment D, Alternate Decay Heat Removal Method Determination.
- CIRCLE the appropriate Alternate Decay Heat Removal Attachment on Attachment D (indicate your decision path on Attachment D).

Attachment D

Alternate Decay Heat Removal Method Determination

Determine the method of Alternate Decay Heat Removal from the Flow Chart.



* Appropriate flow required to remove heat by injection, **based on** 100% full power operation for greater than 30 days.

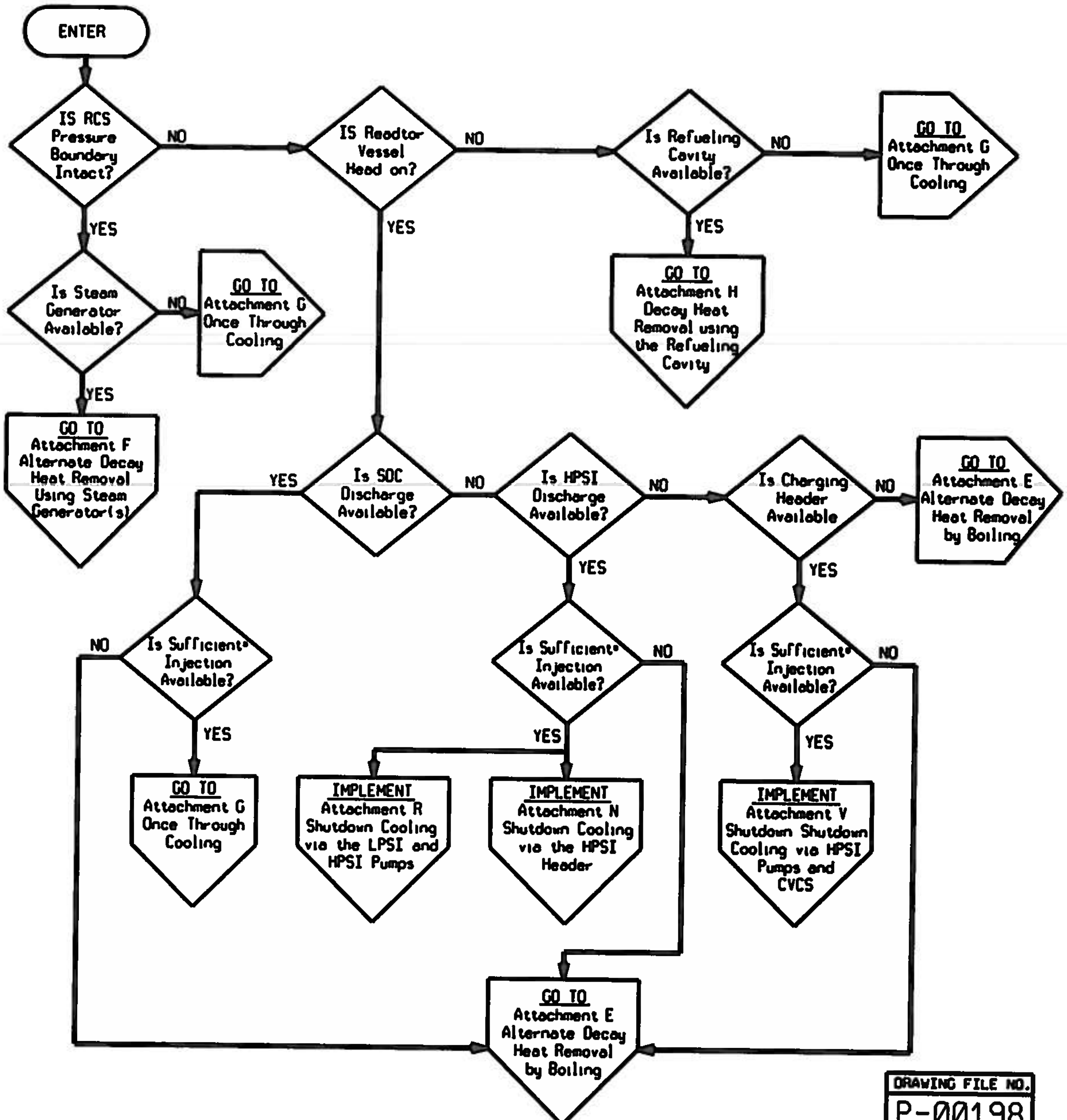
Time after shutdown (days)	Required (gpm)
1	575
5	385
10	310
30	230

End of Attachment D

Attachment D

Alternate Decay Heat Removal Method Determination

Determine the method of Alternate Decay Heat Removal from the Flow Chart.



* Appropriate flow required to remove heat by injection, **based on** 100% full power operation for greater than 30 days.

Time after shutdown (days)	Required (gpm)
1	575
5	385
10	310
30	230

End of Attachment D

Facility: FCS JPM # NRC SA2 Task # 1528 K/A # 2.1.43 4.1 / 4.3

Title: Calculate an Estimated Critical Boron Concentration

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A Reactor Trip from 100% power occurred on 12/03/15 at 0400 following 6 months of full power operation.
- All Control Rods were fully withdrawn at the time of the trip.
- Boron concentration prior to the trip was 400 ppm.
- Average Core Burnup is 6 GWD/MTU.
- Current boron concentration is 500 ppm.
- Criticality is scheduled to occur with Regulating Group 4 at 78 inches.
- Reactor Engineering reports no correction is needed for boron depletion.
- Reactor Startup is scheduled for 12/10/15 at 0600.

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- CALCULATE estimated critical boron concentration per TDB-V.1.B, Estimated Critical Conditions Worksheet.
- COMPLETE data entry through TDB-V.1.B, Step F.4, Estimated Critical Condition Summary.

Task Standard: Utilizing TDB-V.1.B and TDB-II, calculated Estimated Critical Boron Concentration and Minimum and Maximum Critical Rod Position, completed the Estimated Critical Condition Summary.

Required Materials: TDB-V.1.B, Estimated Critical Conditions Worksheet, Rev. 26.
TDB-II, Technical Data Book Reactivity Curves, Rev. 35.
TDB-VI, Core Operating Limits Report, Rev. 42

Validation Time: 37 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

CLASSROOM SETUP

EXAMINER:
PROVIDE the examinee with a copy of:

- **TDB-V.1.B, Estimated Critical Conditions Worksheet**
- **TDB-II, Technical Data Book Reactivity Curves**
- **TDB-VI, Core Operating Limits Report**
- **Calculator**
- **Straight Edge**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from TDB-V.1.B.**
Perform Step: 1 A	Conditions at Time of Shutdown.
Standard:	ENTERED Conditions at Time of Shutdown in TDB-V.1.B Steps A.1 to A.5.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 B	Conditions at Time of Startup.
Standard:	ENTERED Conditions at Time of Startup in TDB-V.1.B Steps B.1 to B.4.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 C	ECC Applicability.
Standard:	DETERMINED early and late date/time limits for ECC Applicability and entered data in TDB-V.1.B Steps C.1 to C.4.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ D	Reactivity Changes Due To Shutdown.
Standard:	CALCULATED and ENTERED Reactivity Changes Due To Shutdown in TDB-V.1.B Steps D.1 to D.5.
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ E	Estimated Critical Boron Concentration.
Standard:	CALCULATED and ENTERED Estimated Critical Boron Concentration in TDB-V.1.B Steps E.1 to E.3. (+/- 25 ppm from key)
Examiner Note:	**Information found on Answer Key.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ F	Minimum and Maximum Critical Rod Position.
Standard:	CALCULATED and ENTERED Minimum and Maximum Critical Rod Position in TDB-V.1.B Steps F.1 to F.4. (+/- 5 inches from key)
Examiner Note:	**Information found on Answer Key.**
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **A Reactor Trip from 100% power occurred on 12/03/15 at 0400 following 6 months of full power operation.**
- **All Control Rods were fully withdrawn at the time of the trip.**
- **Boron concentration prior to the trip was 400 ppm.**
- **Average Core Burnup is 6 GWD/MTU.**
- **Current boron concentration is 500 ppm.**
- **Criticality is scheduled to occur with Regulating Group 4 at 78 inches.**
- **Reactor Engineering reports no correction is needed for boron depletion.**
- **Reactor Startup is scheduled for 12/10/15 at 0600.**

INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **CALCULATE estimated critical boron concentration per TDB-V.1.B, Estimated Critical Conditions Worksheet.**
- **COMPLETE data entry through TDB-V.1.B, Step F.4, Estimated Critical Condition Summary.**

Fort Calhoun Station
Unit 1

TDB-V.1.B

TECHNICAL DATA BOOK

ESTIMATED CRITICAL CONDITIONS WORKSHEET

Change No.	EC 65304
Reason for Change	Change procedure so there is only one start up procedure.
Requestor	T. Korenak
Preparer	K. Bessey
Issue Date	05-15-15 3:00 pm

ESTIMATED CRITICAL CONDITION WORKSHEET

Part I - Performed after a mid-cycle shutdown where Xenon may be present.

Criticality No. ________

TODAY
Date

A. Conditions at Time of Shutdown

A.1. Shutdown date and time. 12/03/15 (DATE) 0400 (TIME)

A.2. Reactor power before shutdown. 100 %

A.3. CEA Positions before time of shutdown.

Group 1 126 inches

Group 2 126 inches

Group 3 126 inches

Group 4 126 inches

Group N 126 inches

A.4. Reactor Coolant System boron concentration before shutdown. 400 PPM

A.5. Core average burnup. 6000 MWD/MTU

B. Conditions at Time of Startup

B.1. Startup date and time. 12/10/15 (DATE) 0600 (TIME)

B.2. Time interval between shutdown and startup. 170 hours

B.3. Desired CEA positions at time of startup. (Manual Sequential Mode)

Group 1 126 inches

Group 2 126 inches

Group 3 126 inches

(1) Group 4 78 inches

Group N 126 inches

(1) Normally 85 inches or as specified by the Reactor Engineer.

B.4. Present Reactor Coolant System boron concentration. 500 PPM

C. ECC Applicability

C.1. Using table 1, determine early and late date/time limits for ECC applicability.

Table 1

Time since shutdown (hours)	0-48	48-84	84+
ECC Applicability (hours)	+/-1	+/-2	No limit

C.2. ECC Part I is applicable for plus/minus No Limit hours of predicted startup date and time recorded in B.1.

C.3. Early date and time. N/A (DATE) N/A (TIME)

C.4. Late date and time. N/A (DATE) N/A (TIME)

D. Reactivity Changes Due to Shutdown

NOTE: Due to transient conditions which may not have been assumed in the generation of TDB figures, the Reactor Engineer may request DEN to provide present condition data. This data will be generated with approved computer codes and may be entered in place of TDB figure data. Such entries shall be annotated.

NOTE: When using TDB figures to obtain reactivity values, enter absolute values (no signs). When performing calculations with reactivity values the answer will be in the proper sign. Record calculated value with sign.

D.1. Reactivity change due to change in power:

Using TDB Figure II.C.2.a or II.C.2.b, the power level from A.2, and the Average Core Burnup from A.5, record the power defect added due to the shutdown.

+ 1.778 or + 1.74 %Δρ
D.1

D.2. Reactivity change due to control rod position change:

NOTE: The applicable TDB Figures II.B.2 or II.B.3 can be used, but the same figure must be used in both Steps D.2.a. and D.2.b.

D.2.a. Reactivity change due to rod insertion:

Using the applicable TDB Figure, the CEA positions from A.3, and the Average Core Burnup from A.5, record the reactivity contribution due to inserting all regulating rods.

3.29 or 0.337 %Δρ
D.2.a

D.2.b. Reactivity change due to rod withdrawal:

Using the applicable TDB Figure, the CEA positions from B.3, and the Average Core Burnup from A.5, record the reactivity contribution due to withdrawal of CEA's to startup positions.

3.10 or 0.15 %Δρ
D.2.b

D.2.c. Calculate the net reactivity change due to net control rod position change. Subtract D.2.a from D.2.b. Be sure to enter the sign of the difference.

(3.10 or 0.15) - (3.29 or 0.337) = - 0.19 (-0.187) %Δρ
D.2.b D.2.a D.2.c

D.3. Reactivity change due to changes in Xenon Concentrations.

NOTE: If equilibrium conditions existed before plant shutdown, TDB Figures II.D.1.a, II.D.1.b or II.D.2 may be used to determine the shutdown Xenon worth.

NOTE: If equilibrium conditions existed before shutdown, and the plant tripped, TDB Figure II.D.2 may be used to determine both shutdown and startup Xenon worth.

NOTE: If the above conditions do not apply, use the conditions of Parts A and B above to determine the reactivity change due to Xenon.

D.3.a. Shutdown Xenon worth $\boxed{2.70}$ %Δρ (D.3.a)

D.3.b. Startup Xenon worth $\boxed{0.0}$ %Δρ (D.3.b)

D.3.c. Calculate the net reactivity change due to Xenon. Subtract D.3.b from D.3.a. Be sure to enter the algebraic sign of the difference.

($\boxed{2.70}$ D.3.a) - ($\boxed{0.0}$ D.3.b) = $\boxed{+2.70}$ D.3.c %Δρ

D.4. Reactivity change due to change in boron concentration.

D.4.a. If there is no change in boron concentration enter 0 on line D.4.e below and proceed to Step D.5.

D.4.b. Enter TDB Figure II.A.4 using the burnup of A.5 above. Record the HZP inverse soluble boron worth.

$\boxed{113}$ ppm/%Δρ (D.4.b)

D.4.c. Reactivity due to soluble boron concentration at time of shutdown. Divide boron concentration A.4 by inverse boron worth D.4.b.

($\boxed{400}$ A.4) / ($\boxed{113}$ D.4.b) = $\boxed{3.54}$ D.4.c %Δρ

D.4.d. Reactivity due to soluble boron concentration at time of startup. Divide present boron concentration B.4 by inverse boron worth D.4.b.

($\boxed{500}$ B.4) / ($\boxed{113}$ D.4.b) = $\boxed{4.43}$ D.4.d %Δρ

D.4.e. Calculate the net reactivity change due to boron concentration change. Subtract D.4.d from D.4.c above. Enter the algebraic sign of the difference.

($\boxed{3.54}$ D.4.c) - ($\boxed{4.43}$ D.4.d) = $\boxed{-0.89}$ D.4.e %Δρ

D.5. Sum of reactivity changes:

(Find the algebraic sum of items D.1, D.2.c, D.3.c and D.4.e. Be sure to include the algebraic sign.)

D.5.a. Change in reactivity due to change in power. $\boxed{+ 1.778 \text{ (or } 1.74)}$ %Δρ
D.1

D.5.b. Change in reactivity due to control rod position change.

$\boxed{- 0.19}$ %Δρ
D.2.c

D.5.c. Change in reactivity due to Xenon Transient. $\boxed{+ 2.70}$ %Δρ
D.3.c

D.5.d. Change in reactivity due to change in boron concentration.

$\boxed{- 0.89}$ %Δρ
D.4.e

D.5.e. Total $\boxed{+ 3.398 \text{ (or } 3.36)}$ %Δρ
D.5.e

E. Estimated Critical Boron Concentration

E.1. Find and record inverse boron worth using TDB Figure II.A.4 at HZP and the burnup A.5 above.

Inverse boron worth = $\boxed{113}$ ppm/% Δρ
E.1

E.2. Find the change in boron concentration by multiplying D.5.e by E.1. (Be sure to transcribe the algebraic sign of D.5.e.)

$\boxed{+ 3.398 \text{ (or } 3.36)}$ % Δρ X $\boxed{113}$ ppm/% Δρ = $\boxed{384 \text{ (or } 380)}$ ppm
D.5.e E.1 E.2

E.3. Find the estimated boron concentration.

E.3.a. Present boron concentration $\boxed{500}$ ppm
B.4

E.3.b. Change in boron concentration $\boxed{384 \text{ (or } 380)}$ ppm
E.2

E.3.c. If large RCS Volume changes (eg., draining to mid-loop) are made, an adjustment may be needed to compensate for changes in B-10 concentration. Adjustments may be required, depending on various factors such as time in life or the length of the operating cycle between shutdowns. Contact Reactor Performance Analysis for assistance in determining this value.

0 ppm
E.3.c

E.3.d. Estimated critical boron concentration (ECB)

= (500) + (384 (or 380)) + (0) = 884 (+25) ppm
E.3.a E.3.b E.3.c 0

F. Minimum and Maximum Critical Rod Position

NOTE: Mark Steps F and 0 N/A if diluting to critical.

F.1. Determination of critical position CEA worth.

F.1.a. Use data from B.3, D.2.b for desired critical position.

Group 4 at 78 inches F.1.a

F.1.b. Using desired position from F.1.a, applicable TDB Figure II.B.2 and core average burnup A.5, determine critical position CEA worth.

CEA worth at desired critical position: 3.10 %Δρ
F.1.b

F.2. Find the maximum critical CEA position.

F.2.a. Calculate critical CEA worth +0.5% Δρ by adding 0.5% Δρ to the critical CEA worth F.1.b.

(3.10) + 0.5% Δρ = 3.60 % Δρ
F.1.b F.2.a

F.2.b. Using F.2.a, the applicable TDB Figure II.B.2 the Core Average Burnup A.5, determine the maximum critical CEA position.

Group 4 at >126 (ARO) inches F.2.b

F.2.c. Use the lower of CEA position of F.2.b and ARO as the maximum critical CEA position.

Group [4] at [>126 (ARO)] inches F.2.c

F.3. Find the minimum critical CEA position.

F.3.a. Calculate the critical CEA worth – 0.5% Δρ by subtracting 0.5% Δρ from the critical CEA worth F.1.b.

[3.10] (F.1.b) - 0.5% Δρ = [2.60] (F.3.a) % Δρ

F.3.b. Using F.3.a, the applicable TDB Figure II.B.2 and the Core Average Burnup A.5, find the CEA position corresponding to estimated critical rod position – 0.5%Δρ and enter here.

Group [3] at [75-76] inches F.3.b

F.3.c. Using TDB Section VI Figure 2 record the zero power dependent insertion limit by obtaining PDIL or transient insertion limit value for 0% power.

Group [Group 2 at 126 or Group 3 at 25.2] inches F.3.c

F.3.d. Use the higher of CEA positions from F.3.b and F.3.c as the minimum critical CEA position.

Group [3] at [75-76] inches F.3.d

F.4. Estimated Critical Condition Summary

Present Boron Concentration [500] ppm

Estimated Critical Boron Concentration [884] ppm

Minimum Critical Position	Group	3	at	75-76	inches (F.3.d)
Estimated Critical Position	Group	4	at	78	inches (F.1.a)
Maximum Critical Position	Group	4	at	126	inches (F.2.c)

Reactor Engineer (QNE)________________ Date/Time ______/______

NRC SRO APPLICANTS STOP HERE

Fort Calhoun Station
Unit 1

TDB-V.1.B

TECHNICAL DATA BOOK

ESTIMATED CRITICAL CONDITIONS WORKSHEET

Change No.	EC 65304
Reason for Change	Change procedure so there is only one start up procedure.
Requestor	T. Korenak
Preparer	K. Bessey
Issue Date	05-15-15 3:00 pm

ESTIMATED CRITICAL CONDITION WORKSHEET

Part I - Performed after a mid-cycle shutdown where Xenon may be present.

____
Criticality No.

____
Date

A. Conditions at Time of Shutdown

A.1. Shutdown date and time. ____________ ____________
DATE TIME

A.2. Reactor power before shutdown. __________%

A.3. CEA Positions before time of shutdown.

Group 1 ___________ inches

Group 2 ___________ inches

Group 3 ___________ inches

Group 4 ___________ inches

Group N ___________ inches

A.4. Reactor Coolant System boron concentration before shutdown. _________ PPM

A.5. Core average burnup. _________ MWD/MTU

B. Conditions at Time of Startup

B.1. Startup date and time. ______________ ____________
DATE TIME

B.2. Time interval between shutdown and startup. __________ hours

B.3. Desired CEA positions at time of startup. (Manual Sequential Mode)

Group 1 ____________ inches

Group 2 ____________ inches

Group 3 ____________ inches

(1) Group 4 ____________ inches

Group N ____________ inches

(1) Normally 85 inches or as specified by the Reactor Engineer.

B.4. Present Reactor Coolant System boron concentration. __________ PPM

C. <u>ECC Applicability</u>

C.1. Using table 1, determine early and late date/time limits for ECC applicability.

Table 1

Time since shutdown (hours)	0-48	48-84	84+
ECC Applicability (hours)	+/-1	+/-2	No limit

C.2. ECC Part I is applicable for plus/minus _______ hours of predicted startup date and time recorded in B.1.

C.3. Early date and time. ___________ ___________
DATE TIME

C.4. Late date and time. ___________ ___________
DATE TIME

D. <u>Reactivity Changes Due to Shutdown</u>

NOTE: Due to transient conditions which may not have been assumed in the generation of TDB figures, the Reactor Engineer may request DEN to provide present condition data. This data will be generated with approved computer codes and may be entered in place of TDB figure data. Such entries shall be annotated.

NOTE: When using TDB figures to obtain reactivity values, enter absolute values (no signs). When performing calculations with reactivity values the answer will be in the proper sign. Record calculated value with sign.

D.1. Reactivity change due to change in power:

Using TDB Figure II.C.2.a or II.C.2.b, the power level from A.2, and the Average Core Burnup from A.5, record the power defect added due to the shutdown.

______________ %Δρ
D.1

D.2. Reactivity change due to control rod position change:

NOTE: The applicable TDB Figures II.B.2 or II.B.3 can be used, but the same figure must be used in both Steps D.2.a. and D.2.b.

D.2.a. Reactivity change due to rod insertion:

Using the applicable TDB Figure, the CEA positions from A.3, and the Average Core Burnup from A.5, record the reactivity contribution due to inserting all regulating rods.

______________ %Δρ
D.2.a

D.2.b. Reactivity change due to rod withdrawal:

Using the applicable TDB Figure, the CEA positions from B.3, and the Average Core Burnup from A.5, record the reactivity contribution due to withdrawal of CEA's to startup positions.

______________ %Δρ
D.2.b

D.2.c. Calculate the net reactivity change due to net control rod position change. Subtract D.2.a from D.2.b. Be sure to enter the sign of the difference.

(______________) - (______________) = ______________ %Δρ
D.2.b D.2.a D.2.c

D.3. Reactivity change due to changes in Xenon Concentrations.

NOTE: If equilibrium conditions existed before plant shutdown, TDB Figures II.D.1.a, II.D.1.b or II.D.2 may be used to determine the shutdown Xenon worth.

NOTE: If equilibrium conditions existed before shutdown, and the plant tripped, TDB Figure II.D.2 may be used to determine both shutdown and startup Xenon worth.

NOTE: If the above conditions do not apply, use the conditions of Parts A and B above to determine the reactivity change due to Xenon.

D.3.a. Shutdown Xenon worth _____________ %Δρ
D.3.a

D.3.b. Startup Xenon worth _______________ %Δρ
D.3.b

D.3.c. Calculate the net reactivity change due to Xenon. Subtract D.3.b from D.3.a. Be sure to enter the algebraic sign of the difference.

(_______________) - (_______________) = _____________ %Δρ
D.3.a D.3.b D.3.c

D.4. Reactivity change due to change in boron concentration.

D.4.a. If there is no change in boron concentration enter 0 on line D.4.e below and proceed to Step D.5.

D.4.b. Enter TDB Figure II.A.4 using the burnup of A.5 above. Record the HZP inverse soluble boron worth.

_______________ ppm/%Δρ
D.4.b

D.4.c. Reactivity due to soluble boron concentration at time of shutdown. Divide boron concentration A.4 by inverse boron worth D.4.b.

(_______________) / (_______________) = _____________ %Δρ
A.4 D.4.b D.4.c

D.4.d. Reactivity due to soluble boron concentration at time of startup. Divide present boron concentration B.4 by inverse boron worth D.4.b.

(_______________) / (_______________) = _____________ %Δρ
B.4 D.4.b D.4.d

D.4.e. Calculate the net reactivity change due to boron concentration change. Subtract D.4.d from D.4.c above. Enter the algebraic sign of the difference.

(_______________) - (_______________) = _____________ %Δρ
D.4.c D.4.d D.4.e

D.5. Sum of reactivity changes:

(Find the algebraic sum of items D.1, D.2.c, D.3.c and D.4.e. Be sure to include the algebraic sign.)

D.5.a. Change in reactivity due to change in power. ______________ %Δρ
D.1

D.5.b. Change in reactivity due to control rod position change.

______________ %Δρ
D.2.c

D.5.c. Change in reactivity due to Xenon Transient. ______________ %Δρ
D.3.c

D.5.d. Change in reactivity due to change in boron concentration.

______________ %Δρ
D.4.e

D.5.e. Total

______________ %Δρ
D.5.e

E. <u>Estimated Critical Boron Concentration</u>

E.1. Find and record inverse boron worth using TDB Figure II.A.4 at HZP and the burnup A.5 above.

Inverse boron worth = ______________ ppm/% Δρ
E.1

E.2. Find the change in boron concentration by multiplying D.5.e by E.1. (Be sure to transcribe the algebraic sign of D.5.e.)

____________ % Δρ X ______________ ppm/% Δρ = ______________ ppm
D.5.e E.1 E.2

E.3. Find the estimated boron concentration.

E.3.a. Present boron concentration ____________ ppm
B.4

E.3.b. Change in boron concentration __________ ppm
E.2

E.3.c. If large RCS Volume changes (eg., draining to mid-loop) are made, an adjustment may be needed to compensate for changes in B-10 concentration. Adjustments may be required, depending on various factors such as time in life or the length of the operating cycle between shutdowns. Contact Reactor Performance Analysis for assistance in determining this value.

______________ ppm
E.3.c

E.3.d. Estimated critical boron concentration (ECB)

= (__________) + (__________) + (__________) = __________ ppm
E.3.a E.3.b E.3.c 0

F. Minimum and Maximum Critical Rod Position

NOTE: Mark Steps F and 0 N/A if diluting to critical.

F.1. Determination of critical position CEA worth.

F.1.a. Use data from B.3, D.2.b for desired critical position.

Group ______________ at ___________ inches F.1.a

F.1.b. Using desired position from F.1.a, applicable TDB Figure II.B.2 and core average burnup A.5, determine critical position CEA worth.

CEA worth at desired critical position: ______________ %$\Delta\rho$
F.1.b

F.2. Find the maximum critical CEA position.

F.2.a. Calculate critical CEA worth +0.5% $\Delta\rho$ by adding 0.5% $\Delta\rho$ to the critical CEA worth F.1.b.

(____________) + 0.5% $\Delta\rho$ = __________ % $\Delta\rho$
F.1.b F.2.a

F.2.b. Using F.2.a, the applicable TDB Figure II.B.2 the Core Average Burnup A.5, determine the maximum critical CEA position.

Group ______________at ____________ inches F.2.b

F.2.c. Use the lower of CEA position of F.2.b and ARO as the maximum critical CEA position.

Group ______________ at ______________ inches F.2.c

F.3. Find the minimum critical CEA position.

F.3.a. Calculate the critical CEA worth – 0.5% Δρ by subtracting 0.5% Δρ from the critical CEA worth F.1.b.

__________ - 0.5% Δρ = _______ % Δρ
F.1.b F.3.a

F.3.b. Using F.3.a, the applicable TDB Figure II.B.2 and the Core Average Burnup A.5, find the CEA position corresponding to estimated critical rod position – 0.5%Δρ and enter here.

Group ____________at __________ inches F.3.b

F.3.c. Using TDB Section VI Figure 2 record the zero power dependent insertion limit by obtaining PDIL or transient insertion limit value for 0% power.

Group ____________at __________ inches F.3.c

F.3.d. Use the higher of CEA positions from F.3.b and F.3.c as the minimum critical CEA position.

Group ____________at __________ inches F.3.d

F.4. Estimated Critical Condition Summary

Present Boron Concentration______________ppm			
Estimated Critical Boron Concentration________________ppm			
Minimum Critical Position	Group__________	at_________	inches (F.3.d)
Estimated Critical Position	Group__________	at_________	inches (F.1.a)
Maximum Critical Position	Group__________	at_________	inches (F.2.c)

Reactor Engineer (QNE)______________________ Date/Time _______/_____

NRC SRO APPLICANTS STOP HERE

Perform the Following If the Reactor Is Not Critical with Group 4 at 115", Otherwise this Step Is N/A

F.5. Using TDB Figure II.B.3, determine the amount of reactivity needed to offset inserting Group 4 to 85"

_______________ - _______________ = _______________ % Δρ
CEA worth @ 115"   CEA worth @ 85"   F.5

F.6. Calculate twice the CEA worth difference:

_______________ X 2 = _______________ % Δρ
F.5   F.6

F.7. Determine rod worth at twice the reactivity difference:

_______________ - _______________ = _______________ % Δρ
CEA worth @ 115"   F.6   F.7

F.8. Using TDB Figure II.B.3 and the rod worth calculated in Step F.7, determine the minimum position of Group 4.

_______________ inches withdrawn
Group 4

F.9. Calculate FCS boron dilution change using inverse boron worth (Step E.1) and the reactivity difference calculated in Step F.5.

_______________ ppm/%Δρ X _______________ % Δρ = _______________ ppm
E.1   F.5   F.9

F.10. Calculate boron concentration change,

_______________ ppm - _______________ ppm = _______________ ppm
0   F.9   F.10

F.11. Performed by: ______________________________ Date/Time _________/______

F.12. Verify Step 0 calculations reviewed prior to Reactor Criticality.

Reactor Engineer: ______________________________ Date/Time _________/______

G. Minimum and Maximum Critical Boron Concentration for Reactor Startup by Dilution

NOTE: Mark Step G N/A if using rod position for criticality.

G.1. Calculation of maximum critical boron concentration, ECB + 0.5%Δρ.

__________ ppm + (__________ ppm/%Δρ X 0.5%Δρ) = __________ ppm
0 D.4.b Max Boron Conc

G.2. Calculate minimum critical boron concentration. Limit + ECB - 50 ppm

__________ ppm - 50 ppm = __________ ppm
0 Min Boron Conc

G.3. Results of this calculation have been independently reviewed prior to Reactor Criticality.

Performed by: ____________________ Date/Time _____/_____

Reactor Engineer: ____________________ Date/Time _____/_____

H. Actual Critical Data

H.1. After achieving criticality, complete the following record:

H.1.a. CEA Positions

Group 1 ____________ inches

Group 2 ____________ inches

Group 3 ____________ inches

Group 4 ____________ inches

Group N ____________ inches

H.1.b. Reactor Coolant System boron concentration: ______________ ppm

H.1.c. Time at which criticality was attained: _____________

H.1.d. Reactor Coolant System average temperature (T_{AVG}) _____ °F

H.2. Ensure actual critical data recorded in Step H.1 is logged in the Control Room Log.

Completed by: ______________________________________ Date/Time _______/_____

I. Review

I.1. Forward to Reactor Engineer for review.

NOTE: If the reactivity difference between actual and estimated critical conditions is greater than 0.5%Δρ, then the Reactor Engineer will determine if the appropriate curves should be upgraded.

Reactor Engineer _________________________________ ________/______
Reactor Engineer Date/Time

Part II - Performed after a refueling outage.

____
Criticality No.

____
Date

A. Conditions at Time of Startup

A.1. Startup date and time. ____________ (DATE) ____________ (TIME)

A.2. Desired CEA Group 4 position at time of startup. (Manual Sequential Mode)

Group 4 ____________ inches

A.3. Sequential Rod Worth at ARO (TDB Figure II.B.2.a) ____________ %Δρ

A.4. Sequential Rod Worth, Group 4 at position from A.2 (TDB Figure II.B.2.a) ____________ %Δρ

A.5. Critical Boron Concentration for BOC, ARO, HZP No Xenon (TDB Figure II.A.1.a.1) ____________ ppm

A.6. Reactor Coolant System Inverse Boron Worth (TDB Figure II.A.4) ____________ ppm / %Δρ

A.7. Current RCS Boron Concentration ____________ ppm Sample Date/Time ____________

B. Estimated Critical Boron Concentration

B.1. Change in reactivity due to CEA Group 4

____________ (A.3) %Δρ – ____________ (A.4) %Δρ = ____________ (B.1) %Δρ

B.2. Change in boron due to CEA Group 4

____________ (B.1) %Δρ X ____________ (A.6) ppm/%Δρ = ____________ (B.2) ppm

B.3. Estimated Critical Boron Concentration (ECB)

____________ (A.5) ppm – ____________ (B.2) ppm = ____________ (B.3) ppm

C. Minimum and Maximum Critical Rod Position

C.1. Find the minimum critical CEA position.

C.1.a. Calculate the critical CEA worth – 0.5%Δρ by subtracting 0.5%Δρ from the critical CEA worth A.4.

______________ – 0.5%Δρ = ______________ %Δρ
A.4 C.1.a

C.1.b. Using TDB Figure II.B.2.a, find the CEA position corresponding to estimated critical rod position – 0.5%Δρ and enter here.

Group ______________ at ______________ inches

C.1.c. Using TDB Section VI COLR Figure 2, record the zero power dependent insertion limit by obtaining PDIL or transient insertion limit value for 0% power.

Group ______________ at ______________ inches

C.1.d. Use the higher of CEA positions from C.1.b and C.1.c as the minimum critical CEA position.

Group ______________ at ______________ inches

C.2. Find the maximum critical CEA position.

C.2.a. Calculate critical CEA worth +0.5% Δρ by adding 0.5% Δρ to the critical CEA worth A.4.

______________ + 0.5%Δρ = ______________ %Δρ
A.4 C.1.a.1

C.2.b. Using TDB Figure II.B.2.a, find the CEA position corresponding to estimated critical rod position + 0.5%Δρ (C.1.a.1) and enter here.

Group ______________ at ______________ inches

C.2.c. Use the lower of CEA position of C.2.b and ARO as the maximum critical CEA position.

Group ______________ at ______________ inches

Present Boron Concentration______________ppm

Estimated Critical Boron Concentration_________________ppm

Minimum Critical Position	Group____________	at_________	inches (C.1.d)
Estimated Critical Position	Group____________	at_________	inches (A.2)
Maximum Critical Position	Group____________	at_________	inches (C.2.c)

C.3. Completed by _________________________ Date/Time _______/_______

C.4. Results of this calculation have been independently reviewed prior to Reactor Criticality.

Reactor Engineer _________________________ Date/Time _______/_______

D. <u>Perform the Following if the Reactor Is Not Critical with Group 4 at 115"</u> (Otherwise this is N/A)

D.1. Using Figure TDB II.B.3.a, determine the amount of reactivity needed to offset inserting Group 4 to 85".

______________ – ______________ = ______________ %Δρ
CEA worth @ 115" CEA worth @ 85" D.1

D.2. Calculate RCS boron dilution change using Inverse Boron Worth (Step A.6) and the reactivity difference calculated in Step D.1.

______________ ppm/%Δρ X ______________ %Δρ = ______________ ppm
A.6 D.1 D.2

D.3. Calculate the New desired critical boron concentration.

______________ ppm – ______________ ppm = ______________ ppm
B.3 D.2 D.3

D.4. Performed by _________________________ Date/Time _______/_______

D.5. Verify Step D calculations reviewed prior to Reactor Criticality.

Reactor Engineer _________________________ Date/Time _______/_______

E. Actual Critical Data

E.1. After achieving criticality, complete the following record:

E.1.a. CEA Positions

Group 1 ______________ inches

Group 2 ______________ inches

Group 3 ______________ inches

Group 4 ______________ inches

Group N ______________ inches

E.1.b. Reactor Coolant System boron concentration: ______________ ppm

E.1.c. Time at which criticality was attained: ______________

E.1.d. Reactor Coolant System average temperature (T_{AVG}): ____________ °F

E.2. Completed by ______________________________ Date/Time ______/______

F. Review

F.1. Forward to Reactor Engineer for review.

NOTE: If the reactivity difference between actual and estimated critical conditions is greater than 0.5%$\Delta\rho$, then the Reactor Engineer will review the results with the Supervisor-Reactor Physics and determine the appropriate actions, if necessary.

F.2. Reviewed by ______________________________ Date/Time ______/______
Supervisor-Reactor Performance Analysis

F.3. Reviewed by ______________________________ Date/Time ______/______
Reactor Engineer

Facility: FCS JPM # NRC SA3 Task # 1260 K/A # 2.2.40 3.4 / 4.7

Title: Determine In-Core Instrumentation Operability

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Plant is at 100% power.
- An In-Core Detector Status Map was just completed for Cycle 28.

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- EVALUATE the In-Core detector system indications and detector status per OI-NI-2, In-Core Instrumentation Operability Requirements.
  - In-Core Instrumentation System OPERABILITY per OI-NI-2 (CIRCLE): YES / NO
  - IDENTIFY required actions, if any:
    - ______________________
    - ______________________

Task Standard: Utilizing OI-NI-2, evaluated In-Core Instrumentation Map and determined Technical Specification LCO 2.10.4(1)(a)(i) & (ii) applies.

Required Materials: OI-NI-2, In-Core Instrumentation Operability Requirements, Rev. 9.
TDB-I.A.7.C, Core Exit Thermocouple Status, Rev. 89.
Fort Calhoun Station Technical Specifications, Amendment #283.

Validation Time: 22 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: __________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **OI-NI-2, In-Core Instrumentation Operability Requirements.**
- **TDB-I.A.7.C, Core Exit Thermocouple Status.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OI-NI-2, Attachment 1.**
Perform Step: 1√ 1, 1.a, 1.b, & 1.b.1)	WHEN either of the following conditions are met, THEN the In-core Detector System is considered operable: • At least 75% of all In-core Detector Strings are operable and at least two In-core Detector Strings are operable per full Axial Quadrant. • Between 28% and 75% of all In-core Detector Strings are operable and: ◦ At least two In-core Detector Strings are operable per full Axial Quadrant
Standard:	REVIEWED TDB-I.A.&.C, Core Exit Thermocouple Status map and PERFORMED the following: • DETERMINED 8 of 28 In-Core Detector Strings are inoperable (71.4%), which is between 28% and 75% • DETERMINED at least two In-core Detector Strings are OPERABLE per full Axial Quadrant. • CIRCLED **YES**.
Comment:	**SAT ☐ UNSAT ☐**

<table>
<tr><td>Perform Step: 2√
1, 1.b, 1.b.2), & 1.b.3)</td><td>WHEN either of the following conditions are met, THEN the In-core Detector System is considered operable:
Between 28% and 75% of all In-core Detector Strings are operable and:
An increase of 1% to the total uncertainties shall be applied to the maximum radial peaking factor (F_R^T) and the total peaking factor (F_q^T) and
The frequency of performing RE-ST-RX-0001 is changed to a minimum of once every 15 days.</td></tr>
<tr><td>Standard:</td><td>IDENTIFIED Required Actions and RECORDED the following:
Apply an increase of 1% to the total uncertainties for maximum radial peaking factor (FRT) and the total peaking factor (FQT), and
Increase the frequency of performing RE-ST-RX-0001 to a minimum of once every 15 days.</td></tr>
<tr><td>Terminating Cue:</td><td>This JPM is complete.</td></tr>
<tr><td colspan="2">Comment: SAT ☐ UNSAT ☐</td></tr>
</table>

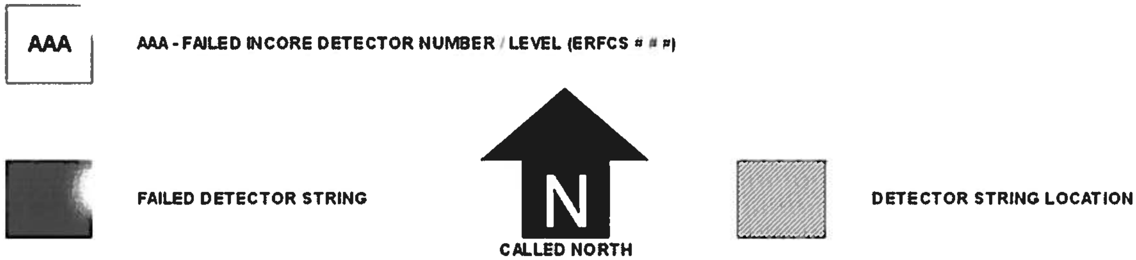
STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

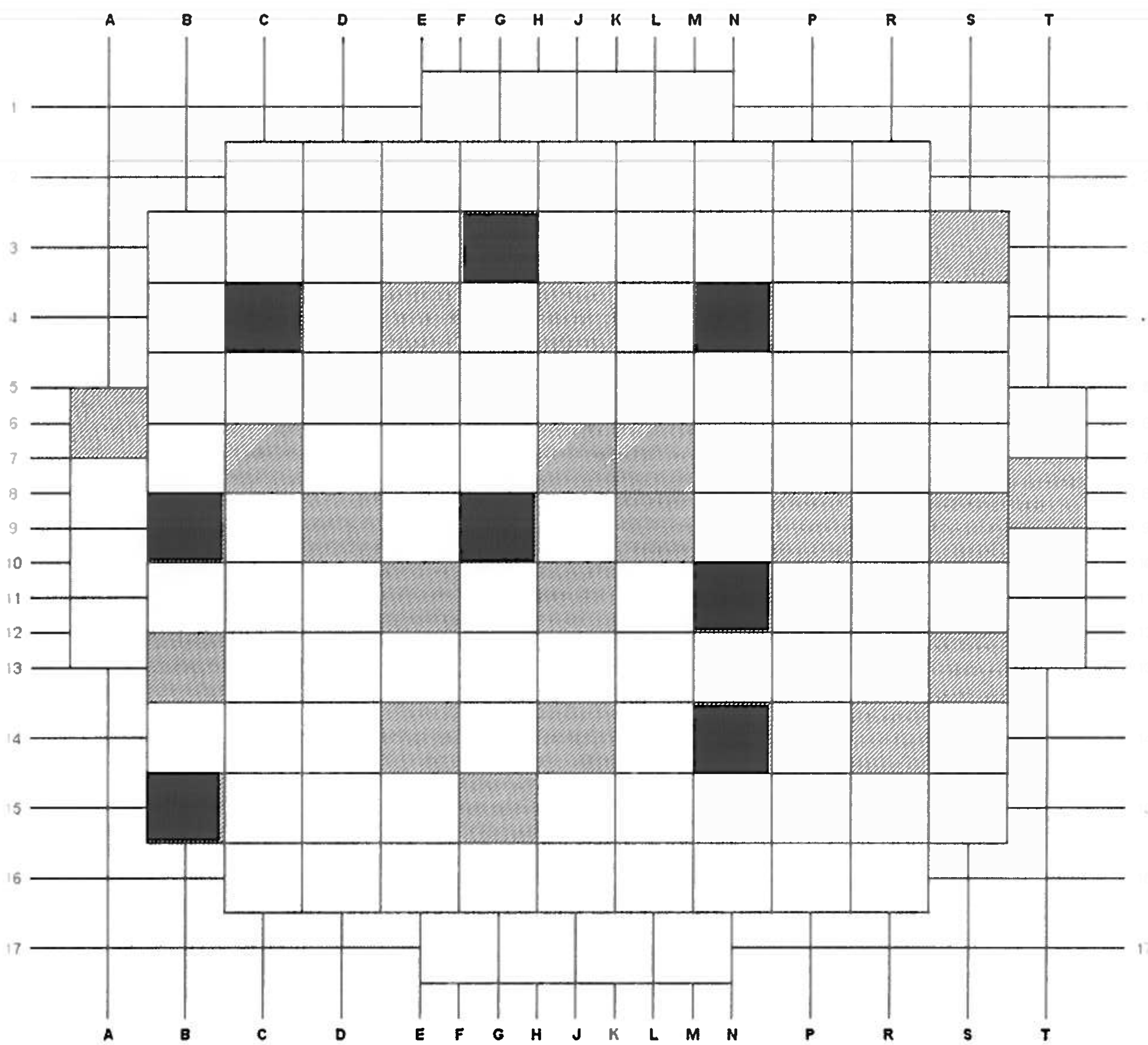
- **Plant is at 100% power.**
- **An In-Core Detector Status Map, was just completed for Cycle 28.**

INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **EVALUATE the In-Core detector system indications and detector status per OI-NI-2, In-Core Instrumentation Operability Requirements.**
  - **In-Core Instrumentation System OPERABILITY per OI-NI-2 (CIRCLE): YES / NO**
  - **IDENTIFY required actions, if any:**
    - ______________________________
    - ______________________________



FAILED INCORE DETECTOR STATUS
CYCLE 28



REFER TO OI-NI-2

Fort Calhoun Station
Unit No. 1

OI-NI-2

OPERATING INSTRUCTION

IN-CORE INSTRUMENTATION OPERABILITY REQUIREMENTS

Change No.	EC 59910, 58911
Reason for Change	Delete reference to TDB-I.A.7B. Changes reflect implementation fo License Amendment 11-03, "Incorporate New Radial Peaking Factor Definition and Clarify Limiting conditions for Operation (LCO) 2.10.2(6)".
Requestor	S. Lindquist, S. Baughn
Preparer	K. Bessey, S. Baughn
Issue Date	04-30-13 3:00 pm

IN-CORE INSTRUMENTATION OPERABILITY REQUIREMENTS

SAFETY RELATED

ATT PURPOSE PAGE

PRECAUTIONS

1. Definition:
   - Core Quadrant - An area containing six or more In-core Detector Strings. Core Quadrants are not strictly defined.
   - In-core Detector String - four Rhodium Detectors and one Core Exit Thermocouple (CET)
   - Operable In-core Detector String - three or more operable Rhodium Detectors
   - Quadrant Symmetric In-core Detector String Location - consist of a location with a symmetric counterpart in any other quadrant
   - Tilt Groups - Sets of four approximately symmetric incore detectors used in the calculation of tilts

2. Loss of the ERF renders the In-core Detector System inoperable AND Technical Specification 2.10.4(1)(b) applies.

3. The minimum number of detectors and proper distribution must be met to ensure operation within the Limits used as Initial Conditions for the Safety Analysis are met:

   a. Maximum Radial Peaking Factor, F_R^T, is less than the limits of Technical Specifications 2.10.4(2) as provided in the COLR.

   b. Specified Kw/ft Limits are less than the Peak Linear Heat Rate vs. Burnup figure in the COLR AND ensured by actuating alarms set on each individual instrument.

   c. To determine the Axial Shape Index for the periodic calibration verification of the Ex-core Detector System.

   d. To determine azimuthal power tilt.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specification:

   - 2.10.4: Power Distribution Limits

2. Technical Data Book:

   - TDB-I.A.6 IN-CORE INSTRUMENTATION LOCATION
   - TDB-I.A.7.a IN-CORE INSTRUMENTATION MIMIC NUMBERS AND SERIAL NUMBERS
   - TDB-I.A.7.c FAILED INCORE DETECTOR STATUS
   - TDB-VI CORE OPERATING LIMITS REPORT

3. USAR Section 7.5.4

APPENDICES

None

Attachment 1 - Operability Requirements

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

Revision Number ________ Date ________ ________

PROCEDURE

1. WHEN either of the following conditions are met,
THEN the In-core Detector System is considered operable:

a. At least 75% of all In-core Detector Strings are operable and at least two In-core Detector Strings are operable per full Axial Quadrant. ________

b. Between 28% and 75% of all In-core Detector Strings are operable and:

1) At least two In-core Detector Strings are operable per full Axial Quadrant, ___

2) An increase of 1% to the total uncertainties shall be applied to the maximum radial peaking factor (F_R^T) and the total peaking factor (F_q^T) and ___

3) The frequency of performing RE-ST-RX-0001 is changed to a minimum of once every 15 days. ___ ________

2. IF the In-core Detector System is inoperable,
THEN do not use the system to monitor F_R^T, Radial Power Distribution, and Peak Linear Heat Rate and Azimuthal Tilt. ________

Attachment 1 - Operability Requirements

PROCEDURE (continued) (✓) INITIALS

CAUTION

Reactor Power shall be restricted to less than 75% of Peak Linear Heat Rate when initial measurements cannot be made.

3. The initial measurements of $F_R{}^T$, Linear Heat Rate, and Azimuthal Power Tilt after each fuel loading shall be made with the following:

   a. An operable In-core Detector System with the following:

      1) At least 75% of all In-core Detector Strings operable. ___

      2) At least two Quadrant Symmetric In-core Detector String Locations per Core Quadrant. ___

   b. Power Level greater than 40% for the first Full Core Power Distribution Calculation based on In-core Detector Signals. ___ ________

4. IF calibrating the Ex-core Detectors,
THEN a minimum of four In-core Locations at each In-core Detector Level (16 detectors total) with at least one location in the center seven rows AND one location outside the center seven rows of fuel assemblies shall be operable. ________

Completed by ______________________ Date/Time ________/______

Facility: FCS JPM # NRC SA4 Task # 0741 K/A # 2.3.7 3.5 / 3.8

Title: Authorize a Liquid Waste Release

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	____	Classroom:	X
Actual Performance:	X	Simulator:	____
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Raw Water Pumps AC-10A, AC-10B, and AC-10C are operating.
- Monitor Tank "A" is being released and was placed on recirculation four hours ago.
- The permit has just been received in the Control Room to release Monitor Tank "A."

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- REVIEW the Liquid Release Permit and Plant Conditions and CIRCLE the results:
  - Correct Tank is being DISCHARGED? YES / NO
  - Maximum Allowable Flow DETERMINED? YES / NO
  - Unloader Flow Rate SATISFACTORY? YES / NO
  - Dilution Pump alignment SATISFACTORY? YES / NO

Task Standard: Utilizing FC-211, determined Monitor Tank A is being discharged, Maximum Allowable Flow Rate identified as 140 gpm, Unloader Flow Rate is unsatisfactory and improper Pump alignment is in service.

Required Materials: FC-211, Waste Liquid Tank Release Permit, Rev. 25.

Validation Time: 13 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **FC-211, Waste Liquid Tank Release Permit.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from FC-211.**
Perform Step: 1√	Is the correct Tank being discharged?
Standard:	DETERMINED Monitor Tank A is being released and CIRCLED **YES**.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**FC-211, Step IV, Maximum Release Rate Calculations - Set Unloader to a flow rate of 130 gpm which is less than or equal to 90% of the maximum release rate listed in Part IV.**
Perform Step: 2√	Is the maximum allowable flow rate determined?
Standard:	DETERMINED Maximum Allowable Flow Rate set at 140 gpm and CIRCLED **YES**.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**FC-211, Step VII, Special Instructions, Item B. - Set Unloader to a flow rate of 130 gpm which is less than or equal to 90% of the maximum release rate listed in Part IV.**
Perform Step: 3√	Is the Unloader Flow Rate satisfactory?
Standard:	DETERMINED Unloader Flow Rate is NOT satisfactory and CIRCLED **NO**. (Unloader Flow Rate should be 126 gpm (90% of 140 gpm).)
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**FC-211, Step VII, Special Instructions, Item C. - Maintain 2 Circulating Water Pumps in operation.**
Perform Step: 4√	Is the Pump alignment satisfactory?
Standard:	DETERMINED Pump alignment is NOT satisfactory and CIRCLED **NO**. (3 Raw Water Pumps are operating and Step VII, Item C calls for 2 Circulating Water Pumps.)
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **Raw Water Pumps AC-10A, AC-10B, and AC-10C are operating.**
- **Monitor Tank "A" is being released and was placed on recirculation four hours ago.**
- **The permit has just been received in the Control Room to release Monitor Tank "A."**

INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **REVIEW the Liquid Release Permit and Plant Conditions and CIRCLE the results:**
  - **Correct Tank is being DISCHARGED? YES / NO**
  - **Maximum Allowable Flow DETERMINED? YES / NO**
  - **Unloader Flow Rate SATISFACTORY? YES / NO**
  - **Dilution Pump alignment SATISFACTORY? YES / NO**

WASTE LIQUID TANK RELEASE PERMIT

RELEASE NUMBER: 2015210

"A" MONITOR TANK

I. Permit Information:

Issue Date: 05-DEC-2015 | Issue Time: 08:14

Sample Date: 03-DEC-2015 | Sample Time: 06:37

Preparer: AJR | Initial Level: 96 inches

II. Chemistry Analysis:

Oil	None Visible
Visual Turbidity	Clear
pH	6.8
pH Adjusted	6.8

III. Gamma Analysis (uCi/ml):

CO-58	1.55E-06
CO-60	2.25E-07
CS-137	2.98E-07
H-3	6.68E-02
SB-124	1.87E-07

WASTE LIQUID TANK RELEASE PERMIT

RELEASE NUMBER: 2015210

"A" MONITOR TANK

IV. Maximum Release Rate Calculations:

Based on 1/2 WEC Summation:	1.7E+03 gpm
Based on RM-055 Setpoints:	1.4E+02 gpm
Maximum Allowable Flowrate:	1.4E+02 gpm

V. Projected Release Information at 140 gpm:

Nuclide	Tank Conc. (uCi/ml)	Site Discharge Conc. (uCi/ml)	WEC Limit (uCi/ml)	Site Discharge Fraction	Activity (uCi)
CO-58	1.75E-06	1.75E-10	1.75E-06	1.75E-06	1.75E+01
CO-60	2.55E-07	2.55E-10	2.55E-07	2.55E-07	2.55E+00
CS-137	3.41E-07	3.41E-10	3.41E-07	3.41E-07	3.41E+00
H-3	7.09E-02	7.09E-05	7.09E-02	7.09E-02	2.32E+06
SB-124	2.11E-07	2.11E-11	2.11E-07	2.11E-07	2.11E+00
TOTALS	7.09E-02	7.09E-05		7.07E-02	2.32E+06
Totals:					

WASTE LIQUID TANK RELEASE PERMIT

RELEASE NUMBER: 2015210

"A" MONITOR TANK

VI. Compositing Requirements:

Monthly and Quarterly Composite samples collected per CH-AD-0010.

Qualified Chemistry Technician: C. James

VII. Special Instructions:

A. Release "A" Monitor Tank in accordance with OI-WDL-3.
B Set Unloader to a flow rate of 130 gpm which is less than or equal to 90% of the maximum release rate listed in Part IV.
C, Maintain 2 Circulating Water Pumps in operation.
D. Ensure Warm Water Recirc remains secured.
E.Steam Generator releases may be performed.
F. Establish and maintain the release flow rate such that the ALERT setpoint on RM-055 is not continuously exceeded. If this condition cannot be attained, secure the release and notify Chemistry.
G. Verify RR-049A recorder readings are tracking RM-055 readings.

Remarks: ______

MONITOR TANK RELEASE PERMIT 2015210

VIII. Projected Cumulative Dose Information

A. Liquid Effluents Dose

	Current Release	Year to Date	Annual Obj	Percent of Annual Obj
Total Body(mRem):	1.72E-04	1.52E-02	3.00E+00	0.38%
Critical Organ(mRem):	2.23E-04	1.68E-02	1.05E+01	0.15%

IX. Approvals:

Form Revision Number Agrees with Master Form Revision Number: C. James
Qualified Chem Tech

Permit Reviewed by: Alan Beebe Date: today
Shift Chemist

Release Approved: L. Shubert Date: today
Supervisor System Chemistry

X. OPERATIONS CHECKLIST — INITIALS/DATE

A. PRIOR TO LIQUID RELEASE

1. If the effluent monitor is inoperable and not set to automatically close the discharge valve in accordance with the ODCM, then two independent samples must be analyzed and two qualified individuals must independently verify the release rate calculations. (Mark "N/A" if not applicable.)

N/A 8 / today
Chemist

N/A 8 / today
Supv-System Chemistry

2. Determine that the following recorders are operable.

NOTE: If any of these recorders are inoperable, the appropriate portions of the Discharge Log must be completed every two hours.

a. PROCESS RADIATION MONITORS RECORDER ______ / ______

b. WASTE LIQUID RELEASE FLOWRATE RECORDER ______ / ______

3. Shift Manager notified prior to start of release. ______ / ______
Shift Manager

B. Initiate Liquid Waste Release in accordance with OI-WDL-3 and record initial readings in Table I. ______ / ______

C. Terminate Liquid Waste Release in accordance with OI-WDL-3 and record final readings in Table I. ______ / ______

D. Attach a copy of the completed applicable section of OI-WDL-3 to this liquid release permit. ______ / ______

	Date	Time	Tank Level	X1 Integrator Reading	X4 Integrator Reading
Start			in.		
Stop			in.		
Maximum Release Rate =	gpm (From FR-690)				
Release Duration =	min. (Chemistry)				

(OI-WDL-3, CH-AD-0022)

LIQUID DISCHARGE LOG*

Permit No.__________________ Monitor Tank I.D.______________

Time/Date	Waste Liquid Flow (GPM)	RM-055 Process Monitor Reading (CPM)

***NOTE 1**: This log will normally not be filled out. Only if one or more of the recorders itemized in the Waste Liquid Release Permit are inoperable will this log be used (every 4 hours) to record applicable readings.

***NOTE 2**: If the waste liquid flow indicator is inoperable, calculate the release rate based on change in integrator readings or change in tank level divided by the time between readings.

WASTE LIQUID TANK RELEASE PERMIT

RELEASE NUMBER:

" " MONITOR TANK

XI. Post-Release Summary:

Initial Level: inches

Final Level: inches

Initial X1 Int:
Final X1 Int:

Initial X4 Int:
Final X4 Int:

Dilution Volume: mls
Raw Pumps in Service:
Release Duration: min

Limiting Flowrate: gpm
Maximum Flowrate: gpm
Average Flowrate: gpm

Release Volume by Tank Level: mls (Info)
Release Volume by Totalizer: mls (Calcs)

Nuclide	Max Site Disch Conc (uCi/ml)	Avg Site Disch Conc (uCi/ml)	WEC Limit (uCi/ml)	Site Discharge Fraction	Activity Released (uCi)
Totals:					

Remarks: ____________________

XII. Final Summarization and Approval:

Summarized by: ____________________ Date: __________
Chemistry

Approved by: ____________________ Date: __________
Supervisor-System Chemistry

Facility: FCS JPM # NRC SA5 Task # 1453 K/A # 2.4.41 2.9 / 4.6

Title: Classify an Emergency Plan Event

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______
Time Critical:	X		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- The Site has received an aircraft attack notification from the Nuclear Regulatory Commission Headquarters Operations Officer that was confirmed by Security at 1200.
- Actions of AOP-37, Security Events, are being implemented.
- Aircraft arrival time is estimated at 60 minutes.

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- DETERMINE the Recognition Category and Event Classification per EPIP-OSC-1, Emergency Plan.
  - Emergency Classification Level __________
  - IC/EAL Classification __________

  THIS IS A TIME CRITICAL JPM

Task Standard: Utilizing EPIP-OSC-1 and TDB-EPIP-OSC-1H, determined Recognition Category and classified the event as a Notification of Unusual Event Category HU4.

Required Materials: EPIP-OSC-1, Emergency Plan, Rev. 48b.

TDB-EPIP-OSC-1H, Recognition Category H - Hazards and Other Conditions Affecting Plant Safety, Rev. 3.

Validation Time: 5 minutes Completion Time: ________ minutes

Critical Time limit: 15 minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

CLASSROOM SETUP

EXAMINER:
PROVIDE the examinee with a copy of:

- **EPIP-OSC-1, Emergency Plan.**
- **EPIP-OSC-1A/1C/1E/1F/1H/1S, EPIP Recognition Category Basis Documents.**

NOTE:
PROVIDE the entire EPIP-OSC-1A/1C/1E/1F/1H/1S, EPIP Recognition Category Basis Documents.

√ - Check Mark Denotes Critical Step

CRITICAL START TIME:	

Examiner Note:	**The following steps are from Fort Calhoun Station Emergency Action Levels.**
Examiner Note:	**The Applicant may reference TDB-EPIP-OSC-1H which is the EPIP Bases document for HAZARDS.**
Perform Step: 1	DETERMINE the Event Category.
Standard:	REFERRED to FCS Emergency Action Levels: • Figure 8.1, Recognition Categories That Apply to Operating Modes Greater Than OR Equal to 210°F. • Figure 8.1, Recognition Categories That Apply to Operating Modes Less Than to 210°F.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2	MATCH plant conditions in the Recognition Category.
Standard:	IDENTIFIED EAL Recognition Category **H - Hazards and Other Conditions Affecting Plant Safety**.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√	Declare the event emergency level.
Standard:	IDENTIFIED Emergency level – **NOUE (Notification of Unusual Event)**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Declaration shall be made within 15 minutes of start time of JPM.**
Perform Step: 4√	Classify the event.
Standard:	CLASSIFIED the event as a **NOTIFICATION OF UNUSUAL EVENT (HU4), EAL 3.** Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant, EAL #3: A validated notification from NRC providing information of an aircraft threat.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

CRITICAL STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **The Site has received an aircraft attack notification from the Nuclear Regulatory Commission Headquarters Operations Officer that was confirmed by Security at 1200.**
- **Actions of AOP-37, Security Events, are being implemented.**
- **Aircraft arrival time is estimated at 60 minutes.**

INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **DETERMINE the Recognition Category and Event Classification per EPIP-OSC-1, Emergency Plan.**
  - **Emergency Classification Level** ___________
  - **IC/EAL Classification** ___________

THIS IS A TIME CRITICAL JPM

Facility:	Fort Calhoun Station	Date of Examination:	Dec / 2015
Exam Level:	RO SRO(I) **SRO (U)**	Operating Test No.:	NRC

Control Room Systems: *(8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)

	System / JPM Title	Type Code*	Safety Function
S-1	Control Rod Drive System (N) (RO Only) Perform Control Element Assembly Exercises	A, N, S	1
S-2	Chemical and Volume Control System (N) Align Charging Flow Via the HPSI Header	N, S	2
S-3	**Emergency Core Cooling System (007F)** **Perform HPSI Stop and Throttle**	**A, D, EN, S**	**3**
S-4	**Reactor Coolant Pump System (0612)** **Start a Reactor Coolant Pump**	**D, L, S**	**4P**
S-5	Main Turbine Generator System (N) Perform Control Room Evacuation Required Actions	A, N, S	4S
S-6	Containment Spray System (0369) Reset Containment Spray Actuation Signal	D, EN, S	5
S-7	**Emergency Diesel Generator System (344AF)** **Parallel and Load Emergency Diesel Generator**	**A, M, S**	**6**
S-8	Reactor Protection System (0778) Adjust Reactor Protection System T_{COLD} Calibration	D, S	7

In-Plant Systems: *(3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)

P-1	**Spent Fuel Pool Cooling System (N)** **Spent Fuel Pool Cooling Restoration with SIAS**	**E, N, R**	**8**
P-2	**Auxiliary Feedwater System (0101C)** **Locally Start FW-54, Diesel Driven AFW Pump**	**A, E, M**	**4S**
P-3	Waste Gas Disposal System (N) Terminate Release of Radioactive Gas	N, R	9

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.
* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	≥ 1 / ≥ 1 / ≥ 1 (control room system)
(L)ow Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	1 / 1 / 1
(S)imulator	

NRC JPM Examination
Summary Description

S-1 The applicant will perform Control Element Assembly exercises per OPT-ST-CEA-0003, Control Element Assembly Partial Movement Check. The alternate path requires a Reactor Trip per EOP-00, Standard Post-Trip Actions, when two Control Rods drop into the core during surveillance testing. This is a new JPM under the Control Rod Drive System – Reactivity Control Safety Function. This is a PRA significant action. (K/A 001.A2.11 - IR 4.4 / 4.7)

S-2 The applicant will align Charging flow via the High Pressure Safety Injection (HPSI) header per AOP-33, CVCS Leak, Attachment C, Charging Via the HPSI Header Using Only CH-1C. This is a new JPM under the Chemical and Volume Control System – Reactor Coolant System Inventory Control Safety Function. (K/A 004.A4.08 - IR 3.8 / 3.4)

S-3 The applicant will perform the actions for HPSI Stop and Throttle per EOP/AOP Floating Steps, FS-A, HPSI Stop and Throttle Criteria. The alternate path occurs after HPSI flow is throttled and the leak increases and requires re-initiation of HPSI flow. This is a bank JPM under Emergency Core Cooling System – Reactor Pressure Control Safety Function. This is a PRA significant action. (K/A 009.EA2.34 - IR 3.6 /4.2)

S-4 The applicant will start a Reactor Coolant Pump (RCP) RC-3D per OI-RC-9, Reactor Coolant Pump Operation, Attachment 1, Starting Reactor Coolant Pumps (Coupled). This is a bank JPM under the Reactor Coolant System – Primary System Heat Removal from Reactor Core Safety Function. (K/A 003.A2.02 - IR 3.7 / 3.9)

S-5 The applicant will perform required actions prior to a Control Room Evacuation per AOP-07, Evacuation of Control Room. The alternate path occurs when it is determined the Turbine failed to trip following the Reactor Trip. Turbine Trip is accomplished by stopping the Electrohydraulic Control Pumps (EHC). This is a new JPM under the Main Turbine Generator System– Secondary System Heat Removal from Reactor Core Safety Function. This is a PRA significant action. (K/A G 2.1.19 - IR 3.9 / 3.8)

S-6 The applicant will reset a Containment Spray Actuation Signal per EOP/AOP Floating Steps, FS-F, Containment Spray Termination. This is a bank JPM under the Containment Spray System – Containment Integrity Safety Function. (K/A 026.A4.05 - IR 3.5 / 3.5)

S-7 The applicant will parallel and load the Emergency Diesel Generator (EDG) per OI-DG-1, Diesel Generator Operation, Attachment 1, Idle Speed Start and Loading. The alternate path occurs when the EDG continues to load after the Governor Control Switch is released requiring a manual trip of the EDG output breaker. This is a modified bank JPM under the Emergency Diesel Generator System – Electrical Safety Function. This is a PRA significant action. (K/A 064.A4.06 - IR 3.9 /3.9)

S-8 The applicant will adjust Reactor Protection System (RPS) T_{COLD} calibration on Channel D per OI-RPS-2, Reactor Protective System-TM/LP $T_{COLD\ CAL}$ Calibration. This is a bank JPM under the Reactor Protection System – Instrumentation Safety Function. (K/A 012.A4.02 - IR 3.3 / 3.4)

P-1 The applicant will restore Spent Fuel Pool (SFP) Cooling following a Safety Injection Actuation Signal (SIAS) using Spent Fuel Pool Cooling Pump AC-5A per AOP-36, Loss of Spent Fuel Pool Cooling, Attachment H, Spent Fuel Pool Cooling Restoration with SIAS. This is a new JPM under the Spent Fuel Pool Cooling System – Plant Service Systems Safety Function. (K/A 033.A2.02 - IR 2.7 / 3.0) This is a PRA significant action.

P-2 The applicant will perform a local start of FW-54, Diesel Driven Auxiliary Feedwater (AFW) Pump per EOP/AOP Attachments-HR Heat Removal, HR-16, FW-54 Operation. The alternate path occurs when it is determined Auxiliary Feedwater must be directed through the AFW Nozzles and HCV-1384, FW/AFW Header Cross Connect Valve will not open. This is a modified bank JPM under the Auxiliary Feedwater System – Secondary System Heat Removal from Reactor Core Safety Function. (K/A 061.A2.05 - IR 3.1 / 3.4) This is a PRA significant action.

P-3 The applicant will perform actions to terminate a gaseous radioactive release from WD-29A, Waste Gas Decay Tank per AOP-09, High Radioactivity. The release termination is performed per OI-WDG-2, Waste Gas Disposal System Release, Attachment 3, Manual Waste Gas Release with FE-532 Unavailable. This is a new JPM under the Waste Gas Disposal System – Radioactivity Release Safety Function. (K/A 071 G 2.1.30 - IR 4.4 / 4.0)

Facility: FCS JPM # NRC S-1 Task # 0675 K/A # 001.A2.11 4.4 / 4.7 SF-1

Title: Perform Control Element Assembly Exercises

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Maintenance on Shutdown Group A was just completed.
- A partial movement check of Shutdown Group A is required.
- CEAs are in an All-Rods-Out configuration.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- COMPLETE Control Element Assembly exercise on Shutdown Group A per OP-ST-CEA-0003, Control Element Assembly Partial Movement Check.

Task Standard: Utilizing OP-ST-CEA-0003, exercised Shutdown Group A CEAs then tripped the Reactor when 2 CEAs dropped by opening CEDM Clutch Power Supply Breakers.

Required Materials: OP-ST-CEA-0003, Control Element Assembly Partial Movement Check, Rev. 14.
EOP-00, Standard Post Trip Actions, Rev. 33.

Validation Time: 20 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:
INITIALIZE to IC-112:

- **ENSURE DCS Computer Screen set at "CEA ALL."**

Type	Item	Value	Condition
Event	ATWAS_PLUS		
MALF/CRD	ROD_PWR_A30_1 (Rod 30 clutch failure)	DE-ENERGIZED	When second rod motion is performed
MALF/CRD	ROD_PWR_A33_1 (Rod 33 clutch failure)	DE-ENERGIZED	When second rod motion is performed

BOOTH OPERATOR NOTE:

- **After each JPM, VERIFY all control switches and reactor trip pushbutton cover is restored to normal condition prior to performance by the next examinee.**

EXAMINER:
PROVIDE the examinee with a copy of:

- **OP-ST-CEA-0003, Control Element Assembly Partial Movement Check.**
  - **INITIALED through Prerequisites.**
  - **N/A all CEAs from 14 to 1 on Attachment 1.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OP-ST-CEA-0003.**

NOTE

Step 0 can be performed at anytime and repeated as necessary.

Perform Step: 1 7.1	IF this Surveillance Test is turned over, a prejob briefing must be conducted prior to the continuation of this test.
Standard:	ACKNOWLEDGED a pre-job brief is required prior to continuing.
Comment:	**SAT ☐ UNSAT ☐**

CAUTION

When the Reactor is critical, this Surveillance Test must be performed within the specified Technical Specification time interval regardless of rod configuration or use.

Perform Step: 2 7.2	IF not in an All-Rods-Out configuration, THEN contact the Reactor Engineer prior to commencing this test for guidance to ensure the requirements of Technical Specification 3.2, Table 3-5, Item 2 are met.
Standard:	DETERMINED CEAs in an All-Rods-Out configuration per Initial Conditions.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 7.3	Record Initial Position of all CEAs on Attachment 1.
Standard:	RECORDED Initial Position of Shutdown Group A CEAs #30, #31, #32, #33, #34, #35, #36, and #37 on Attachment 1.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 7.4	Rotate the Mode Selector Switch (M/M) to the Manual Individual (M/I) position.
Standard:	ROTATED Mode Selector Switch (M/M) to Manual Individual (M/I) position.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ 7.5	Rotate the Group Selector Switch (M/G) to the Group containing the CEA to be moved.
Standard:	ROTATED Group Selector Switch (M/G) to Shutdown Group A.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6 7.6	If available, verify on SCEAPIS (DCS) display CEA_ALL that the group button is DARK GREY for the group selected.
Standard:	VERIFIED on Secondary Control Element Assembly Position Indicating System (SCEAPIS) Digital Control System (DCS) display CEA_ALL that Shutdown Group A button is DARK GREY.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Cue:	**If questioned, REPORT the CRS directs you to start with CEA #30.**
Perform Step: 7√ 7.7	Rotate the Rod Selector Switch to the CEA to be moved.
Standard:	ROTATED Rod Selector Switch to any Shutdown Group A CEA.
Comment:	**SAT ☐ UNSAT ☐**

NOTE

If Group 4 CEAs are being used for ASI control, movement of 6 inches in a single direction may be credited. The returned to position may not necessarily be the initial position. Note the time of Group 4 movement for ASI control on the Comment Sheet if applicable.

Examiner Note:	**When the 2nd CEA is exercised, two CEAs will drop into the Core.**
Perform Step: 8√ 7.8	Insert or withdraw the CEA, as applicable, a minimum of six (6) inches, THEN return the CEA to its Initial Position.
Standard:	INSERTED CEA a minimum of six (6) inches, MONITORED Nuclear Instrumentation and T_{AVE} then WITHDREW CEA to its Initial Position.
Examiner Cue:	**If the ROD DRIVE POWER INTERRUPT alarm is received (in the event the CEA is moved 8 inches), REPORT as CRS that permission is granted to use the Rod Block Bypass Switch to move the CEA back to its original position.**
Examiner Note:	**Candidate may reposition the Manual Mode Selector switch to "off" to respond to alarms. If so, the candidate must return the switch to "MI" to move the selected rod.**
Comment:	**SAT ☐ UNSAT ☐**

NOTE
Step 0 may be completed after all CEAs within a Group have been exercised, after all CEAs have been exercised, OR after exercising each CEA.

Examiner Note:	**When the 2nd CEA is exercised, two CEAs will drop into the Core.**
Perform Step: 9	Record Inserted/Withdrawn To AND Return To information on Test Data Sheet, THEN initial Attachment 1.
Standard:	Record Inserted/Withdrawn To AND Return To information on Test Data Sheet, THEN initial Attachment 1.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 10	Determine that 2 CEAs have dropped into the core.
Standard:	OBSERVED Annunciator Alarms and DETERMINED two CEAs have dropped.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps are from EOP-00, Standard Post Trip Actions.**

Perform Step: 11
1
	<u>Verify</u> Reactivity Control is established by performing steps a and b:
	• <u>Verify</u> **ALL** of the following:
	  • No more than one Regulating or Shutdown CEA is **NOT** inserted
	  • Reactor power is lowering
	  • Startup rate is negative

Standard:	DETERMINED Reactor did NOT trip when both CEAs dropped and REFERRED to CONTINGENCY ACTIONS (CA).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 12a 1.1 & 1.1.a CA	**IF** the reactor did **NOT** trip, **THEN** <u>establish</u> Reactivity Control by performing step a, b, c or d: • Manually <u>trip</u> the Reactor (CB-4).
Standard:	DEPRESSED REACTOR TRIP pushbutton on CB-4 and DETERMINED Reactor did NOT trip.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 12b 1.1 & 1.1.b CA	**IF** the reactor did **NOT** trip, **THEN** establish Reactivity Control by performing step a, b, c or d: • Manually trip the Reactor (AI-31).
Standard:	DEPRESSED REACTOR TRIP pushbutton on AI-31 and DETERMINED Reactor did NOT trip.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 12c 1.1 & 1.1.c CA	**IF** the reactor did **NOT** trip, **THEN** establish Reactivity Control by performing step a, b, c or d: • Place the DSS Manual Trip Switches in "TRIP" (AI-66A/B).
Standard:	PERFORMED the following: • PLACED DSS Manual Trip Switch in TRIP position on AI-66A and DETERMINED Reactor did NOT trip. • PLACED DSS Manual Trip Switch in TRIP position on AI-66B and DETERMINED Reactor did NOT trip.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 12d√ 1.1 & 1.1.d CA	**IF** the reactor did **NOT** trip, **THEN** establish Reactivity Control by performing step a, b, c or d: • Manually open the CEDM Clutch Power Supply Breakers (AI-57).
Standard:	PERFORMED the following: • OPENED both CLUTCH POWER SUPPLY BREAKER RPS/CB-A/B on AI-57 (**critical**). • OPENED both CLUTCH POWER SUPPLY BREAKER RPS/CB-C/D on AI-57 (**critical**) • OBSERVED all Rod Bottom lights LIT on SCEAPIS (**NOT critical**).
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **Maintenance on Shutdown Group A was just completed.**
- **A partial movement check of Shutdown Group A is required.**
- **CEAs are in an All-Rods-Out configuration.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **COMPLETE Control Element Assembly exercise on Shutdown Group A per OP-ST-CEA-0003, Control Element Assembly Partial Movement Check.**

Fort Calhoun Station
Unit No. 1

OP-ST-CEA-0003

SURVEILLANCE TEST

CONTROL ELEMENT ASSEMBLY (CEA) PARTIAL MOVEMENT CHECK

Change No.	EC 54172
Reason for Change	Incorporate LAR 10-01 into the station manual for 15 minute surveillance for CEA position indication.
Requestor	J. Willett
Preparer	K. Bessey
Issue Date	02-24-12 3:00pm

CONTROL ELEMENT ASSEMBLY (CEA) PARTIAL MOVEMENT CHECK

SAFETY RELATED

1. PURPOSE

To satisfy, at least quarterly when the Reactor is critical, the requirements of Technical Specification 3.2, Table 3-5, Item 2.

2. REFERENCES/COMMITMENT DOCUMENTS

2.1 Technical Specifications:

- 3.2, Table 3-5, Minimum Frequencies for Equipment Tests
- 2.10.2 (4), Full Length CEA Position During Power Operation
- 2.10.2 (6), Shutdown CEA Insertion Limit During Power Operation
- 2.10.2 (7), Regulating CEA Insertion Limits During Hot Standby and Power Operation
- 2.15, Table 2-5, Instrumentation Operating Requirements for Other Safety Feature Functions

2.2 USAR, Section 7

2.3 SO-G-23, Surveillance Test Program

2.4 Ongoing Commitment Documents

- AR 04922, LIC-87-0013
- AR 08779, LIC-88-0765 (LER-88-018)
- AR 14801, LIC-93-0294 (LER-93-016)

3. DEFINITIONS

None

4. EQUIPMENT LIST

None

5. PRECAUTIONS AND LIMITATIONS

5.1 All anomalies and deficiencies shall be reported immediately to the Shift Manager and noted in the Comment Sheet. An immediate check shall be made to verify Limiting Conditions for Operation, per Technical Specifications, have not been exceeded.

5.2 Manipulation of all controls as defined in 10CFR55 shall be done by, or under the direct supervision of, a NRC licensed operator.

5.3 Only one method of positive reactivity addition shall be used at any one time.

5.4 During all rod motion, nuclear instrumentation and RCS temperature shall be observed for response.

5.5 Be alert for any abnormal noise or indications when CEDMs are in operation.

5.6 Power Dependent Insertion Limits shall not be violated by more than one (1) CEA at a time during this test.

5.7 Shutdown CEAs shall not be inserted below the exercise limit.

5.8 Each CEA shall be returned to its pretest position prior to testing other CEAs.

5.9 CEAs within a Group should be kept within two (2) inches of each other per OI-RR-1, unless specifically required otherwise by this procedure.

5.10 The Reactor shall not be placed in the Automatic mode.

5.11 No maintenance shall be conducted within this Surveillance Test other than that specifically directed by this procedure.

5.12 Any time the Mode Selector Switch is in any position other than OFF, all available rod position indications shall be monitored, Synchros, SCEAPIS, ERF (page 302), and Core Mimic. **[AR 14801]**

5.13 If a CEA position abnormality occurs, verify the position of all CEAs. **[AR 14801]**

5.14 The following annunciators may Alarm during the performance of this Surveillance test:

- **REGULATING GROUP WITHDRAWAL** PROHIBIT (CB-4, A8, A-1L)
- **PDIL GR 1 COMPUTER** (CB-4, A8, A-3U)
- **PPDIL GR 1 COMPUTER** (CB-4, A8, A-4U)
- **PDIL GR 2 COMPUTER** (CB-4, A8, A-3L)
- **PPDIL GR 2 COMPUTER** (CB-4, A8, A-4L)
- **ROD DRIVE POWER INTERRUPT** (CB-4, A8, A-5U)
- **ROD POSITION DEVIATION LOW LIMIT** (CB-4, A8, B-1U)
- **ROD POSITION DEVIATION LOW-LOW LIMIT** (CB-4, A8, B-1L)
- **PDIL GR 3 COMPUTER** (CB-4, A8, B-3U)
- **PPDIL GR 3 COMPUTER** (CB-4, A8, B-4U)
- **VLPM ALARM OR TROUBLE** (CB-1,2,3, A6, D-3)

5.15 The following DCS alarms may occur during the performance of this Surveillance Test:

- ROD BLOCK
- PPDIL
- PDIL
- ISH

5.16 If one channel of CEA position indication is inoperable for one or more CEAs, the requirements of specification 2.15 are modified for CEA Position Indication Systems, which requires the CEA position to be verified within 15 minutes following any CEA motion in that group. OP-ST-CEA-0007, 15 Minute Control Element Assembly (CEA) Channel Check for an Inoperable CEA Position Indication System, needs to be completed.

6. PREREQUISITES — INITIALS/DATE

~~6.1~~ Procedure Revision Verification:

Revision No. 14 — X / today

~~6.2~~ Reactor is critical. — X / today

~~6.3~~ No other test is in progress which could potentially affect this test, or if this test were performed, could have an effect on that test. — X / today

~~6.4~~ A prejob briefing has been conducted prior to the start of this test. — X / today

~~6.5~~ Primary Synchro CEA Position Indicating System is operable. **[AR 14801]** — X / today

~~6.6~~ Shift Manager authorizes performance of this test:

Shift Manager A.S. Manager Date/Time today / 0630

7. PROCEDURE

NOTE: Step 7.1 can be performed at anytime and repeated as necessary.

7.1 IF this Surveillance Test is turned over, a prejob briefing must be conducted prior to the continuation of this test. ____ / ____

CAUTION [AR 04922]

When the Reactor is critical, this Surveillance Test must be performed within the specified Technical Specification time interval regardless of rod configuration or use.

7.2 IF not in an All-Rods-Out configuration, THEN contact the Reactor Engineer prior to commencing this test for guidance to ensure the requirements of Technical Specification 3.2, Table 3-5, Item 2 are met. **[AR 04922] [AR 08779]** ______

7.3 Record Initial Position of all CEAs on Attachment 1. ______

7.4 Rotate the Mode Selector Switch (M/M) to the Manual Individual (M/I) position. ______

7.5 Rotate the Group Selector Switch (M/G) to the Group containing the CEA to be moved. ______

7.6 If available, verify on SCEAPIS (DCS) display CEA_ALL that the group button is DARK GREY for the group selected. ______

7.7 Rotate the Rod Selector Switch to the CEA to be moved. ______

NOTE: If Group 4 CEAs are being used for ASI control, movement of 6 inches in a single direction may be credited. The returned to position may not necessarily be the initial position. Note the time of Group 4 movement for ASI control on the Comment Sheet if applicable.

7.8 Insert or withdraw the CEA, as applicable, a minimum of six (6) inches, THEN return the CEA to its Initial Position. ______

NOTE: Step 7.9 may be completed after all CEAs within a Group have been exercised, after all CEAs have been exercised, OR after exercising each CEA.

7.9 Record Inserted/Withdrawn To AND Return To information on Test Data Sheet, THEN initial Attachment 1. ______

7.10 Repeat Steps 7.5 through 7.9 for all CEAs listed on Attachment 1. ______

7.11 Rotate Mode Selector Switch (M/M) to the OFF position. ______

7.12 Independently verify each CEA returned to its initial position listed on Attachment 1.

______
Ind Verif

Completed By: ______________________________ Date/Time ________/______

8. RESTORATION

8.1 Shift Manager notified this test is completed:

Shift Manager ________________________________ Date/Time ________/______

9. ACCEPTANCE CRITERIA

9.1 All CEAs have been moved a minimum of six (6) inches.

10. TEST RECORD

10.1 This entire procedure.

11. REVIEW

NOTE: The Reactor Engineer shall be notified within 24 hours of the completion of this test of any unexpected results.

11.1 Test data shall be evaluated by the STA and reviewed by the Shift Manager for acceptability within 24 hours of completion of this test.

Evaluated by ________________________________ Date/Time ________/______
STA

Reviewed by ________________________________ Date/Time ________/______
Shift Manager

Attachment 1 - CEA Partial Movement Check Test Data Sheet

CEA No.	Initial Position (inches)	Inserted/ Withdrawn (inches)	Returned to (Inches)	Initials	Ind Verif Initials	Remarks
30						
31						
32						
33						
34						
35						
36						
37						
14						
15						
16						
17						
6						
8						
10						
12						
22						
23						
24						
25						
26						
27						
28						
29						
2						

N/A today

Attachment 1 - CEA Partial Movement Check Test Data Sheet

CEA No.	Initial Position (inches)	Inserted/ Withdrawn (inches)	Returned to (Inches)	Initials	Ind Verif Initials	Remarks
3						
4						
5						
38			N/A	[illegible]		
39						
40						
41						
1						

Completed By: ______________________ Date/Time ________/______

Comment Sheet

Facility: FCS JPM # NRC S-2 Task # 1391 K/A # 004.A4.08 3.8 / 3.4 SF-2

Title: Align Charging Flow Via the HPSI Header

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- AOP-33, CVCS Leak, is in progress.
- Leak isolation has restored Charging Pump CH-1C.
- AOP-33, Step 13.d, directs use of Attachment C.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- ALIGN Charging flow via the High Pressure Safety Injection (HPSI) header per AOP-33, CVCS Leak, Attachment C, Charging Via the HPSI Header Using Only CH-1C.

Task Standard: Utilizing AOP-33, opened HCV-308, opened HCV-312, and started Charging Pump CH-1C to restore Pressurizer level.

Required Materials: AOP-33, CVCS Leak, Rev. 9.

Validation Time: 13 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-112:

- **VERIFY Pressurizer level is lowered to 55%.**
- **ENSURE all Charging Loop Isolation Valves CLOSED per AOP-33.**
- **ENSURE all Auxiliary Spray Valves are CLOSED per AOP-33.**
- **ENSURE all Charging Pumps in PULL-TO-LOCK per AOP-33.**

Type	Item	Value	Condition
Remote/CVC	REM:CVC_CH172	0	
Remote/CVC	REM:CVC_CH173	0	
Remote/CVC	REM:CVC_CH191	0	
Remote/CVC	REM:CVC_CH192	0	
Remote/CVC	REM:CVC_CH193	0	
Remote/CVC	REM:CVC_CH194	0	

EXAMINER:

PROVIDE the examinee with a copy of:

- **AOP-33, CVCS Leak.**
  - **Attachment C, Charging Via the HPSI Header Using Only CH-1C.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from AOP-33, Attachment C.**
NOTE Charging flow can be verified on the associated HPSI flow indicator(s) for the HPSI Loop Valve(s) in use, or on ERF (Page 323).	
Perform Step: 1 1	Ensure all Charging Pumps are in "PULL-TO-LOCK".
Standard:	DETERMINED all Charging Pumps in PULL-TO-LOCK.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 2 & all bullets	Unlock and close **BOTH** of the following valves: • CH-194, "CHARGING PUMPS CH-1A/B/C DISCHARGE HEADER CONTAINMENT OUTBOARD ISOLATION VALVE" (Room 13) • CH-191, "CHARGING PUMPS CH-1A & B DISCHARGE HEADER TO SAFETY INJECTION ISOLATION VLV." (Charging Pump Valve Room)
Standard:	CONTACTED Auxiliary Operator to UNLOCK and CLOSE CH-194 in Room 13 and CH-191 in Charging Pump Valve Room.
Booth Operator:	**When contacted, UNLOCK and CLOSE CH-194 and CH-191. REPORT CH-194 and CH-191 UNLOCKED and CLOSED.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 3 & all bullets	Close **ALL** of the following valves: • CH-192, "CHARGING PUMP CH-1B DISCHARGE VALVE" (Charging Pump Valve Room) • CH-173, "CHARGING PUMP CH-1B SUCTION VALVE" (Charging Pump Valve Room) • CH-193, "CHARGING PUMP CH-1A DISCHARGE VALVE" (Charging Pump Valve Room) • CH-172, "CHARGING PUMP CH-1A SUCTION VALVE" (Charging Pump Valve Room)
Standard:	CONTACTED Auxiliary Operator to UNLOCK and CLOSE CH-192, CH-173, CH-193, and CH-172 in Charging Pump Valve Room.
Booth Operator:	**When contacted, CLOSE CH-192, CH-173, CH-193, and CH-172. REPORT CH-192, CH-173, CH-193, and CH-172 CLOSED.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 4	Open HCV-308, Charging Pump HPSI Header Isolation Valve.
Standard:	PERFORMED the following: • PLACED HCV-308, CHARGING PUMP/HPSI HDR ISOLATION VALVE in OPEN (**critical**). • OBSERVED red OPEN light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 5 & all bullets	Ensure **ALL** of the following valves are open: • HCV-2987, HPSI Header Isolation Valve • HCV-307, HPSI Header Isolation Valve • HCV-305, SI-2A and SI-2C Discharge Cross-Connect Valve • HCV-304, SI-2B and SI-2C Discharge Cross-Connect Valve • HCV-306, HPSI Header Isolation Valve
Standard:	VERIFIED all of the following valves OPEN and red OPEN lights lit: • HCV-2987, HPSI HEADER NUMBER 2 DISCHARGE VALVE • HCV-307, HPSI HEADER NUMBER 2 DISCHARGE VALVE • HCV-305, HPSI PUMPS SI [JUNK] 2A/SI [JUNK] 2C DISCH CROSSCONNECT VLV • HCV-304, HPSI PUMPS SI [JUNK] 2B/SI [JUNK] 2C DISCH CROSSCONNECT VLV • HCV-306, HPSI HEADER NUMBER 1 DISCHARGE VALVE
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**HCV-312 was selected for consistency of Applicants.**
Examiner Cue:	**The CRS directs you to open HCV-312, HPSI Loop Injection Valve.**
Perform Step: 6a 6, 6.a, & 6.a.1)	Open at least **ONE** of the following HPSI Loop Injection Valves: • Open HCV-312 (Loop 1B) by performing the following: • Rotate thumbwheel for PCV-2909, "LEAKAGE CLR SI-4A DISCH VLV CNTRLR" fully clockwise to close "C".
Standard:	PERFORMED the following: • VERIFIED thumbwheel for PCV-2909, LEAKAGE CLR SI-4A DISCHARGE VALVE CONTROLLER fully CLOCKWISE in CLOSE (C position). • OBSERVED needle in C position.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6b√ 6, 6.a, & 6.a.2)	Open at least **ONE** of the following HPSI Loop Injection Valves: • Open HCV-312 (Loop 1B) by performing the following: • Place PCV-2909, "LEAKAGE CLR SI-4A DISCHARGE VALVE" in "MANUAL".
Standard:	PERFORMED the following: • PLACED PCV-2909, LEAKAGE CLR SI-4A DISCHARGE VALVE to MANUAL position (**critical**). • OBSERVED switch in MANUAL (**NOT critical**). • OBSERVED amber light off and green light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6c√ 6, 6.a, & 6.a.3)	Open at least **ONE** of the following HPSI Loop Injection Valves: • Open HCV-312 (Loop 1B) by performing the following: • Open HCV-312, "LOOP 1B HPSI INJECTION VALVE".
Standard:	PERFORMED the following: • TURNED and HELD HCV-312, LOOP 1B HPSI INJECTION VALVE to OPEN position (**critical**). • OBSERVED red OPEN light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

NOTE

Charging flow can be verified on the associated HPSI flow indicator(s) for the HPSI Loop Valve(s) in use, or on ERF (Page 323).

Perform Step: 7√ 7	Operate CH-1C as necessary to maintain PZR level within 4% of programmed level.
Standard:	PERFORMED the following: • PLACED CH-1C, CHRG PUMP in START (**critical**). • OBSERVED red START light lit (**NOT critical**). • OBSERVED flow on HCV-312, HPSI Loop Injection Valve or on ERF Computer Page 323 (**NOT critical**). • OBSERVED ~75 amps on CH-1C ammeter (**NOT critical**).
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **AOP-33, CVCS Leak, is in progress.**
- **Leak isolation has restored Charging Pump CH-1C.**
- **AOP-33, Step 13.d, directs use of Attachment C.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **ALIGN Charging flow via the High Pressure Safety Injection (HPSI) header per AOP-33, CVCS Leak, Attachment C, Charging Via the HPSI Header Using Only CH-1C.**

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Charging flow can be verified on the associated HPSI flow indicator(s) for the HPSI Loop Valve(s) in use, or on ERF (Page 323).

1. Ensure all Charging Pumps are in "PULL-TO-LOCK".

2. Unlock and close **BOTH** of the following valves:

   - CH-194, "CHARGING PUMPS CH-1A/B/C DISCHARGE HEADER CONTAINMENT OUTBOARD ISOLATION VALVE" (Room 13)

   - CH-191, "CHARGING PUMPS CH-1A & B DISCHARGE HEADER TO SAFETY INJECTION ISOLATION VLV." (Charging Pump Valve Room)

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS | CONTINGENCY ACTIONS

3. Close **ALL** of the following valves:

- CH-192, "CHARGING PUMP CH-1B DISCHARGE VALVE" (Charging Pump Valve Room)
- CH-173, "CHARGING PUMP CH-1B SUCTION VALVE" (Charging Pump Valve Room)
- CH-193, "CHARGING PUMP CH-1A DISCHARGE VALVE" (Charging Pump Valve Room)
- CH-172, "CHARGING PUMP CH-1A SUCTION VALVE" (Charging Pump Valve Room)

4. Open HCV-308, Charging Pump HPSI Header Isolation Valve.

5. Ensure **ALL** of the following valves are open:

- HCV-2987, HPSI Header Isolation Valve
- HCV-307, HPSI Header Isolation Valve
- HCV-305, SI-2A and SI-2C Discharge Cross-Connect Valve
- HCV-304, SI-2B and SI-2C Discharge Cross-Connect Valve
- HCV-306, HPSI Header Isolation Valve

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS | CONTINGENCY ACTIONS

6. Open at least **ONE** of the following HPSI Loop Injection Valves:

   a. Open HCV-312 (Loop 1B) by performing the following:

      1) Rotate thumbwheel for PCV-2909, "LEAKAGE CLR SI-4A DISCH VLV CNTRLR" fully clockwise to close "C".

      2) Place PCV-2909, "LEAKAGE CLR SI-4A DISCHARGE VALVE" in "MANUAL".

      3) Open HCV-312, "LOOP 1B HPSI INJECTION VALVE".

   b. Open HCV-315 (Loop 1A) by performing the following:

      1) Rotate thumbwheel for PCV-2929, "LEAKAGE CLR SI-4B DISCH VLV CNTRLR" fully clockwise to close "C".

(continue)

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS	CONTINGENCY ACTIONS

6.b (continued)

2) Place PCV-2929, "LEAKAGE CLR SI-4B DISCHARGE VALVE" in "MANUAL".

3) Open HCV-315, "LOOP 1A HPSI INJECTION VALVE".

c. Open HCV-318 (Loop 2A) by performing the following:

1) Rotate thumbwheel for PCV-2949, "LEAKAGE CLR SI-4C DISCH VLV CNTRLR" fully clockwise to close "C".

2) Place PCV-2949, "LEAKAGE CLR SI-4C DISCHARGE VALVE" in "MANUAL".

3) Open HCV-318, "LOOP 2A HPSI INJECTION VALVE".

(continue)

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS	CONTINGENCY ACTIONS

6. (continued)

   d. Open HCV-321 (Loop 2B) by performing the following:

      1) Rotate thumbwheel for PCV-2969, "LEAKAGE CLR SI-4D DISCH VLV CNTRLR" fully clockwise to close "C".

      2) Place PCV-2969, "LEAKAGE CLR SI-4D DISCHARGE VALVE" in "MANUAL".

      3) Open HCV-321, "LOOP 2B HPSI INJECTION VALVE".

NOTE

Charging flow can be verified on the associated HPSI flow indicator(s) for the HPSI Loop Valve(s) in use, or on ERF (Page 323).

7. Operate CH-1C as necessary to maintain PZR level within 4% of programmed level.

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS	CONTINGENCY ACTIONS
8. **IF** Charging Header repairs are possible with the Plant in its current operating mode, **THEN** direct Maintenance to repair leak.	8.1 **IF** Charging Header repairs are **NOT** possible with the Plant in its current operating mode, **THEN** place the Plant in desired mode PER **ONE** of the following procedures: • AOP-05, Emergency Shutdown • OP-4, Load Change and Normal Power Operations
9. **WHEN** the Charging Header has been repaired, **THEN** return the CVCS to normal operation PER OI-CH-1, Startup of Charging and Letdown.	

Attachment C

Charging Via the HPSI Header Using Only CH-1C

INSTRUCTIONS	CONTINGENCY ACTIONS
10. **WHEN** the CVCS has been returned to normal, **THEN** IMPLEMENT Attachment E, HPSI Piping Flush, to flush all safety injection piping used in this procedure.	
11. **WHEN** HPSI piping flush is completed, **THEN** GO TO Section 5.0, Exit Conditions.	

End of Attachment C

Facility: FCS JPM # NRC S-3 Task # 1129 K/A # 009.EA2.34 3.6 / 4.2 SF-3

Title: Perform HPSI Stop and Throttle

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A Small Break Loss of Coolant Accident is in progress.
- EOP-03, Loss of Coolant Accident, has been entered.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- EVALUATE then EXECUTE actions for HPSI Stop and Throttle per EOP/AOP Floating Step A, HPSI Stop and Throttle Criteria.

Task Standard: Utilizing Floating Step A, stopped all but one HPSI Pump and throttled Loop Injection Valves. Upon leak increase, restarted HPSI Pumps and opened Loop Injection Valves as required.

Required Materials: EOP/AOP Floating Steps, Rev. 7.

Validation Time: 10 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:
INITIALIZE to IC-118:

Type	Item	Value	Condition
MALF/RCS	RCS01B (RCS-01B – RCS Loop Leak – Loop 1B Cold Leg Medium)	0.7 ** Leak is present when you restore this IC	Recall/modify after flow and level are balanced, increase leak rate to value 1.5

EXAMINER:
PROVIDE the examinee with a copy of:

- **EOP/AOP Floating Step A, HPSI Stop and Throttle Criteria.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	The following steps are from Floating Step FS-A.

CAUTIONS

1. If emergency boration is required then at least one charging pump must remain running.
2. As natural circulation develops, the expected rise in T_H will reduce subcooling. This may jeopardize HPSI Stop and Throttle Criteria.
3. Reducing SI flow should be approached cautiously.
4. The purpose of HPSI stop and throttle is to prevent an over pressurization of the RCS and a solid PZR, however, maintaining RCS inventory is more important than pressure control.

Perform Step: 1 1 & all bullets	Verify **ALL** of the following stop and throttle criteria are satisfied: • RCS subcooling is greater than or equal to 20°F • PZR level is greater than or equal to 10% and not lowering • At least one S/G is available for RCS heat removal • RVLMS indicates level is at or above the top of the Hot Leg (43%, ERF "I" display)
Standard:	OBSERVED the following: • RCS subcooling greater than 20°F. • Pressurizer level greater than 10% and not lowering. • Both Steam Generators are available for RCS heat removal. • Reactor Vessel Level Monitoring System is greater than 43%.
Comment:	SAT ☐ UNSAT ☐

Examiner Note:	**Applicant must place any 2 of 3 Charging Pumps in PULL-TO-LOCK otherwise they will AUTO START.**
Perform Step: 2√ 2	Ensure only **ONE** Charging Pump is operating.
Standard:	STOPPED 2 of 3 Charging Pumps by PERFORMING the following: • PLACED CH-1A, CHARGING PUMP in PULL-TO-LOCK (**critical**). • PLACED CH-1B, CHARGING PUMP in PULL-TO-LOCK (**critical**). • PLACED CH-1C, CHARGING PUMP in PULL-TO-LOCK (**critical**). • OBSERVED pump breaker lights off (**NOT critical**).
Comment:	SAT ☐ UNSAT ☐

CAUTIONS 1. During a UHE HPSI stop and throttle should be performed before the expansion of the relatively cold SI water overfills the pressurizer. 2. Operators should closely monitor RCS pressure-temperature limits. Pressurizer spray may be required to prevent exceeding the maximum subcooling limit. 3. Allowing the RCS to repressurize to 1300 psia will effectively stop HPSI flow.	
Perform Step: 3 3	**IF** a UHE is in progress, **THEN** maintain RCS pressure control by performing the following:
Standard:	DETERMINED Uncontrolled Heat Extraction is NOT in progress.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**It is acceptable to stop one or more HPSI Pumps and throttle HPSI Loop Injection Valves to achieve control over Pressurizer level, resulting in stable or slowly rising Pressurizer level. Applicant may stop one HPSI Pump in this step and/or throttle HPSI Loop Injection Valves to achieve this condition.**
CAUTIONS 1. LOCAs pose a significant threat to RCS subcooling. Therefore, full SI Flow should be maintained until subcooled margin is stable and natural circulation has developed. 2. During a SGTR, the depressurization of the RCS to less than 1000 psia should be stopped when HPSI flow is initially being stopped and throttled.	
Perform Step: 4a 4 & 4.a	**IF** a LOCA or SGTR is in progress, **THEN** maintain RCS pressure control by performing the following: • Ensure at least one HPSI Pump is operating.
Standard:	DETERMINED SI-2A and SI-2B, HPSI Pumps are running. May stop one HPSI pump or leave both running.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4b 4 & 4.b	**IF** a LOCA or SGTR is in progress, **THEN** maintain RCS pressure control by performing the following: • Throttle HPSI Loop Injection Valve(s).
Standard:	THROTTLED CLOSE any or all the following: • HCV-314, LOOP 1A HPSI INJECTION VALVE. • HCV-317, LOOP 2A HPSI INJECTION VALVE. • HCV-311, LOOP 1B HPSI INJECTION VALVE. • HCV-321, LOOP 2B HPSI INJECTION VALVE.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Examiner Note:	**Once HPSI flow has been throttled, the break size will increase and Stop and Throttle criteria will no longer be met. HPSI Pumps must be restarted <u>and/or</u> Loop Injection Valves reopened. Applicant must recognize this and begin to take action before Reactor Vessel Level Monitoring System (RVLMS) indicates less than 100%.**
Perform Step: 5a 5 & 5.a	**IF** HPSI stop and throttle criteria can **NOT** be maintained, **THEN** <u>raise</u> HPSI flow by performing the following: • <u>Start</u> either HPSI Pumps, SI-2A/B or SI-2B/C, as necessary.
Standard:	DETERMINED SI-2A and SI-2B, HPSI Pumps are running. If HPSI pumps were secured in step 4, restart HPSI pumps. (**critical** if one of more HPSI pumps were secured)
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Applicant should throttle open valve(s) closed at Perform Step 4b.**
Perform Step: 5b√	**IF** HPSI stop and throttle criteria can **NOT** be maintained, **THEN** <u>raise</u> HPSI flow by performing the following: • Open HPSI Loop Injection Valves, as necessary.
Standard:	OPEN any **<u>or</u>** all the following: • HCV-314, LOOP 1A HPSI INJECTION VALVE. • HCV-317, LOOP 2A HPSI INJECTION VALVE. • HCV-311, LOOP 1B HPSI INJECTION VALVE. • HCV-321, LOOP 2B HPSI INJECTION VALVE.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **A Small Break Loss of Coolant Accident is in progress.**
- **EOP-03, Loss of Coolant Accident, has been entered.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **EVALUATE then EXECUTE actions for HPSI Stop and Throttle per EOP/AOP Floating Step A, HPSI Stop and Throttle Criteria.**

1.0 FLOATING STEPS

F. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. If emergency boration is required then at least one charging pump must remain running.
2. As natural circulation develops, the expected rise in T_H will reduce subcooling. This may jeopardize HPSI Stop and Throttle Criteria.
3. Reducing SI flow should be approached cautiously.
4. The purpose of HPSI stop and throttle is to prevent an over pressurization of the RCS and a solid PZR, however, maintaining RCS inventory is more important than pressure control.

****

1. Verify **ALL** of the following stop and throttle criteria are satisfied:
   - RCS subcooling is greater than or equal to 20EF
   - PZR level is greater than or equal to 10% and not lowering
   - At least one S/G is available for RCS heat removal
   - RVLMS indicates level is at or above the top of the Hot Leg (43%, ERF "I" display)

1.0 FLOATING STEPS

F. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS	CONTINGENCY ACTIONS
2. Ensure only **ONE** Charging Pump is operating.	2.1 **IF** RCS inventory is **NOT** being maintained, **THEN** restart Charging Pump as required.

****

CAUTIONS

1. During a UHE HPSI stop and throttle should be performed before the expansion of the relatively cold SI water overfills the pressurizer.
2. Operators should closely monitor RCS pressure-temperature limits. Pressurizer spray may be required to prevent exceeding the maximum subcooling limit.
3. Allowing the RCS to repressurize to 1300 psia will effectively stop HPSI flow.

****

3. **IF** a UHE is in progress,
   THEN maintain RCS pressure control by performing the following:

   a. Stop all HPSI Pumps.

   b. Throttle HPSI Loop Injection Valves.

1.0 FLOATING STEPS

F. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS	CONTINGENCY ACTIONS

****

CAUTIONS

1. LOCAs pose a significant threat to RCS subcooling. Therefore, full SI Flow should be maintained until subcooled margin is stable and natural circulation has developed.

2. During a SGTR, the depressurization of the RCS to less than 1000 psia should be stopped when HPSI flow is initially being stopped and throttled.

****

4. **IF** a LOCA or SGTR is in progress, **THEN** maintain RCS pressure control by performing the following:

   a. Ensure at least one HPSI Pump is operating.

   b. Throttle HPSI Loop Injection Valves.

1.0 FLOATING STEPS

F. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS	CONTINGENCY ACTIONS
5. **IF** HPSI stop and throttle criteria can **NOT** be maintained, **THEN** raise HPSI flow by performing the following:	
a. Start either HPSI Pumps, SI-2A/B or SI-2B/C, as necessary.	
b. Open HPSI Loop Injection Valves, as necessary.	

Facility: FCS JPM # NRC S-4 Task # 0612 K/A # 003.A2.02 3.7 / 3.9 SF-4P

Title: Start a Reactor Coolant Pump

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Plant Startup is in progress.
- Reactor Coolant Pumps RC-3A, RC-3B, and RC-3C are running.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- START Reactor Coolant Pump RC-3D per OI-RC-9, Reactor Coolant Pump Operation, Attachment 1, Starting Reactor Coolant Pumps (Coupled).
- START at Step 11.

Task Standard: Utilizing OI-RC-9, started RC-3D-1 Oil Lift Pump and RCP RC-3D.

Required Materials: OI-RC-9, Reactor Coolant Pump Operation, Rev. 78.

Validation Time: 7 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:
INITIALIZE to IC-116:

- **ENSURE Reactor Coolant Pump RC-3-D is STOPPED.**
- **ENSURE ERF Computer Page 342 or DCS "RCP Summary" on display.**

EXAMINER:
PROVIDE the examinee with a copy of:

- **OI-RC-9, Reactor Coolant Pump Operation.**
  - **Attachment 1, Starting Reactor Coolant Pumps (Coupled), INITIALED through Step 10.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OI-RC-9, Attachment 1.**
NOTE The Oil Lift Pump shall not be run for longer than 10 minutes before starting the Reactor Coolant Pump.	
Perform Step: 1 11	Announce the Reactor Coolant Pump start on the Gaitronics.
Standard:	ANNOUNCED start of Reactor Coolant Pump RC-3D on Gaitronics.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√ 12	Place the Oil Lift Pump for the selected RCP to AFTER START: • RC-3D-1, Oil Lift Pump
Standard:	PERFORMED the following: • PLACED RC-3D-1, OIL LIFT PUMP handswitch in AFTER START (**critical**). • OBSERVED red indicating light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 13	Verify adequate ARRD Lube Oil Flow for the selected RCP. ERF/DCS indication shall read NORMAL: (ERF page 342 or DCS "RCP Summary") • RC-3D F3190
Standard:	OBSERVED ERF Computer Page 342 or Digital Computer System RCP Summary and VERIFIED ARRD Lube Oil Flow for RC-3D is NORMAL.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 14	Prior to starting RCP, inform the Radiation Protection Department so it can monitor changing radiological conditions.
Standard:	CONTACTED Radiation Protection Department about RCP start.
Examiner Cue:	**Radiation Protection acknowledges start of RCP.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5a 15 & 15.1	Startup sequence for selected RCP: • Run the oil lift pump for the selected RCP a minimum of 2 minutes. • RC-3D-1, Oil Lift Pump
Standard:	DETERMINED RC-3D-1, OIL LIFT PUMP already running.
Examiner Cue:	**If Applicant begins timing, REPORT two minutes have passed.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5b√ 15 & 15.2	Startup sequence for selected RCP: • Place the selected RCP control switch in AFTER START: • RC-3D, RC Pump
Standard:	PERFORMED the following: • PLACED RC-3D, RC PUMP handswitch in AFTER START (**critical**). • OBSERVED red START light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5c 15 & 15.3	Startup sequence for selected RCP: • IF the Reactor Coolant Pump motor amps fail to drop below 425 amps within the time listed below, THEN place the control switch in AFTER STOP: • RC-3D - seventeen (17) seconds
Standard:	DETERMINED RC-3D ammeter reads less than 425 amps in less than 17 seconds.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6a 16 & 16.1	Verify the following for the selected RCP: • Oil Lift Pump stops (Green indicating light ON):
Standard:	OBSERVED RC-3D-1, OIL LIFT PUMP green indicating light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6b 16 & 16.2	Verify the following for the selected RCP: • For the selected RCP verify the following Annunciator is clear: • RC-3D **REACTOR COOLANT PUMP RC-3D REVERSE ROTATION** (CB-1/2/3, A6, D5)
Standard:	OBSERVED CB-1/2/3/A6, Window D-5 – REACTOR COOLANT PUMP RC-3D REVERSE ROTATION is CLEAR.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 6c 16 & 16.3	Verify the following for the selected RCP: • For the selected RCP verify the following Annunciator is clear: • RC-3D **REACTOR COOLANT PUMP RC-3D VIBRATION HI** (CB-1/2/3, A6, D4)
Standard:	OBSERVED CB-1/2/3/A6, Window D-4 – REACTOR COOLANT PUMP RC-3D VIBRATION HI is CLEAR.
Comment:	**SAT** ☐ **UNSAT** ☐

NOTE

At low RCS Pressure, verification of positive Controlled Bleedoff Flow may NOT be possible.

Perform Step: 7 17	Verify positive Controlled Bleedoff flow for the selected RCP: (ERF page 342 or DCS "RCP Summary") • RC-3D F3175
Standard:	OBSERVED ERF Computer Page 342 or Digital Computer System RCP Summary and VERIFIED Control Bleedoff Flow for RC-3D is POSITIVE.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 8 18	Monitor the ERF Computer or DCS and verify all parameters are normal for the selected RCP: • RC-3D
Standard:	MONITORED ERF Computer or Digital Computer System and VERIFIED RCP RC-3D parameters NORMAL.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **Plant Startup is in progress.**
- **Reactor Coolant Pumps RC-3A, RC-3B, and RC-3C are running.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **START Reactor Coolant Pump RC-3D per OI-RC-9, Reactor Coolant Pump Operation, Attachment 1, Starting Reactor Coolant Pumps (Coupled).**
- **START at Step 11.**

Fort Calhoun Station
Unit No. 1

OI-RC-9

OPERATING INSTRUCTION

REACTOR COOLANT PUMP OPERATION

Change No.	EC 63785
Reason for Change	Update procedures based on new pressurizer heater modification.
Requestor	A. Mosier
Preparer	K. Bessey
Issue Date	05-21-15 3:00pm

REACTOR COOLANT PUMP OPERATION

SAFETY RELATED

PRECAUTIONS

1. IF more than one main seal has failed, THEN a RCP should NOT be operated. Action should be taken per associated Annunciator Response Procedure (ARP) and AOP-35, Reactor Coolant Pump Malfunctions (Does not apply to an uncoupled RCP).

2. A Reactor Coolant Pump shall NOT be operated with full system pressure on the fourth seal.

3. WHEN a RCP Motor is cold, THEN do NOT attempt more than two starts in succession, allowing the motor to coast completely to rest between starts. This is to prevent damage to motor windings. A cold motor is a motor at ambient temperature (shutdown 4 hours or longer).

4. WHEN the RCP Motor is hot, THEN only one start shall be attempted following shutdown. The RCP Motor shall run a minimum of forty minutes to return itself to normal operating temperatures and permit another hot start. If the RCP Motor does not run for forty minutes or longer following a hot start, a minimum of two hours shall pass before attempting another start. This is to prevent damage to motor windings.

4.1 A hot motor is:

- A motor which has been shutdown for two hours or less with a prior run time less than forty minutes, or
- A motor that has run greater than forty minutes prior to being shutdown, or
- A motor that was started cold and was shutdown upon reaching rated speed

PRECAUTIONS (continued)

5. IF the shaft is rotating in the reverse direction, THEN a RCP Motor shall NOT be started.

6. When the RCP Mechanical Seal are not vented a RCP shall not be operated for more than five minutes continuously.

7. The Anti-Reverse Rotation Device (ARRD) Lube Oil Flow Pressure Switches for RC-3A, RC-3C and RC-3D provide useful flow information only when the RCP oil lift pump is running. The switch setpoint can not be adjusted low enough to properly monitor flow conditions when the oil lift pump is not running. ARRD lube oil flow indication is valid when starting a RCP and should be used. When the RCP is running (i.e., the oil lift pump is OFF), ARRD temperature monitoring on the ERF or DCS shall be used.

8. The following motor stator temperatures on a running RCP shall not be exceeded. If the Reactor is critical the Reactor shall be tripped and the associated RCP tripped immediately:

   - 130°C continuous stator temperature on RC-3A, 3C and 3D
   - 140°C continuous stator temperature on RC-3B
   - Stator temperature rise of 5°C per minute during steady state operation
   - Maximum idle stator temperature is 100°C
   - Minimum stator temperature for a RCP is 15°C

9. Have a Local Operator verify the RCP pump is ready to start <u>and</u> make an announcement on Gaitronics.

10. If no RCP is running and Tc is less than 350°F, then a RCP shall NOT be started unless the following conditions are satisfied (does not apply to uncoupled RCP):

    - Actual PZR level is less than or equal to 50% per (TDB-III.1.a)
    - RCS pressure must be less than the "Maximum Pressure for First Start RCP" (TDB-III.7.a curve 3)

11. IF no RCP is running, THEN RCS pressure must be less than the TDB-III.7.a Maximum Pressure for First RCP Start curve prior to starting an RCP to minimize the potential of opening a PORV.

12. Component Cooling Water shall be flowing through the RCP seal coolers whenever RCS temperature is over 130°F.

13. No more than three RCP's shall be operated when the Reactor Coolant temperature is less than 500°F to prevent fuel damage from excessive fuel assembly axial stress and core lift.

PRECAUTIONS (continued)

14. During power operations (Mode 1) or Hot Standby (Mode 2), the desired operating band for the RCP Controlled Bleed off Temperature is 135°F to 165°F. However, the RCP Controlled Bleedoff Temperature shall remain between 100°F and 180°F during pump operation.

15. A Reactor Coolant Pump shall NOT be operated without controlled Bleedoff flow established. Normally, Reactor Coolant Pump controlled Bleedoff is established within 24 hrs of starting the first RCP (does not apply to an uncoupled RCP).

16. IF Component Cooling Water is lost to the pump motors, THEN the Reactor Coolant Pump Motor(s) shall NOT be operated for more than five minutes and follow the guidance of AOP-11.

17. RCS pressure of 2500 psia shall NOT be exceeded during RCP operation.

18. A valid temperature indication for the motor ARRD, radial or thrust bearings of 210°F for RC-3A, 3C and 3D or 230°F for RC-3B, OR an abnormal temperature rise of greater than or equal to 10°F per minute during steady state operation shall not be exceeded. If the Reactor is critical, the Reactor shall be tripped and the associated RCP immediately shutdown. The associated bearings shall be inspected (RC-3B motor does not have a ARRD bearing).

19. A running RCPs lower seal (cavity) temperature shall not exceed 200°F. If the Reactor is critical, the Reactor shall be tripped and the RCP stopped. Restart shall not be permitted until all O-rings, seal faces and U-cups have been inspected or replaced. Before starting, the RCP Seal Cavity temperature usually exceeds 200°F, but should cool down shortly after the RCP has been started.

20. The lube oil flow monitoring capability of the ARRD Lube Oil Flow Pressure Switches for RC-3A, RC-3C and RC-3D when these pumps are running (i.e., the oil lift pump is OFF), is accomplished with the ARRD Temperature Indication of the ERF or DCS.

21. IF the Vibration Alarm on an operating RCP is IN ALARM, THEN action shall be taken per the ARP and OI-RC-13. OI-RC-13 provides guidance for operation of the Vibration Monitoring System (AI-270) including response to alarms for both normal RCP operation and pump starts.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specifications:
   - Section 1.2: Safety Limit, Reactor Coolant System Pressure
   - Section 2.1.1: Reactor Coolant System, Operable Components
2. USAR Section 4.3.5: Reactor Coolant Pump
3. Technical Data Book:
   - Figure III.1a: Temperature Correction for Pressurizer Level
   - Figure III.7.a: RCS Pressure and Temperature Limits
   - Figure III.7.d: RCS Pressure and Temperature Limits
   - Section III.8.b, RCS Inventory Based on Pressurizer Level
   - Section III.8.c, RCS Volumes
   - Section VII, Tank Curves
4. I&C Equipment List:
   - EM-3101 through 3196: Reactor Coolant Pump Instrumentation (File No. 21643, 42482-42499, 42564)
   - EM-450: Component Cooling Water Flow and Temperature Indication to Seal Cooler and Lube Oil Cooler (File No. 15380)
5. Vendor Manuals:
   - General Electric: Reactor Coolant Pump Motor Manual, Vol. 1 (Technical Manual No. TD G080.2380)
   - Byron Jackson: Reactor Coolant Pump Instruction Manual, Vol. 2 (Technical Manual No. TD-B580.0020)
   - TD A921.0010, ABB Industrial: Reactor Coolant Pump Motor Manual
6. EA-FC-97-037, Reactor Coolant Pump NPSH
7. EAR No. 96-145
8. Commitments (other than Ongoing)
   - AR 16999, SER 24-95

REFERENCES/COMMITMENT DOCUMENTS (continued)

9. Drawings File Description

Drawings	File	Description
E-23866-210-110	44479	Reactor Coolant System
E-23866-210-111 Sh 1	10473	Reactor Coolant Pump RC-3A
E-23866-210-111 Sh 2	45592	Reactor Coolant Pump RC-3B
E-23866-210-111 Sh 3	45592	Reactor Coolant Pump RC-3C
E-23866-210-111 Sh 4	45593	Reactor Coolant Pump RC-3D
E-23866-210-120	41712	Chemical and Volume Control System
E-23866-210-130	44353	Safety Injection and Containment Spray System

APPENDICES

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PREREQUISITES (✓) INITIALS

NOTES
1. Attachment 6 provides guidance for starting RCPs for sweeping steam generator tubes.
2. Attachment 6 provides recovery guidance from the possibility of a slug of water with reduced boron concentration in an idle RCS loop.

1. Procedure Revision Verification

   Revision No. 78 Date: today — γ

2. The Reactor is Shutdown (Mode 3, Mode 4 or Mode 5). — γ

3. Communications have been established between the Control Room and locally at each Reactor Coolant Pump, as needed, prior to Reactor Coolant Pump start. — γ

4. Applicable sections of Checklist OI-RC-9-CL-A have been completed for each RCP to be run. — γ

5. A minimum Reactor Coolant System pressure per TDB-III.7.a or TDB-III.7.d, RCS Pressure and Temperature Limits, has been established to ensure adequate Reactor Coolant Pump Net Positive Suction Head (NPSH). — γ

6. The Component Cooling Water System is in operation to each Reactor Coolant Pump to be run per OI-CC-1. — γ

7. The Red Motor Heater Lights for each non-operating Reactor Coolant Pump are energized. — γ

8. The ERF Computer is available or DCS is available to monitor RCP parameters. — γ

9. A general visual inspection of the RCP should be performed prior to starting. — γ

10. The Reactor Coolant Pump Motor upper and lower oil reservoirs are filled to levels as specified in Tables 1 through 4. — γ

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PREREQUISITES (continued) (✓) INITIALS

11. Instrumentation and alarms associated with RCPs to be run are operable. ____

12. 4.16 KV Buses for RCPs to be run (1A1, 1A2, 1A3 and 1A4) are energized. ____

13. RCPs to be operated are coupled per MM-RR-RC-0008. ____

PROCEDURE

NOTE
Controlled Bleedoff Flow can be established within 24 hrs of starting the first RCP.

1. Open all the following valves:

- RC-308, RC-3A Reactor Coolant Pump Controlled Bleedoff Isolation Valve ✓
- RC-309, RC-3B Reactor Coolant Pump Controlled Bleedoff Isolation Valve ✓
- RC-310, RC-3C Reactor Coolant Pump Controlled Bleedoff Isolation Valve ✓
- RC-311, RC-3D Reactor Coolant Pump Controlled Bleedoff Isolation Valve ✓ ____

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

CAUTION

IF the Steam Generator secondary side temperature is greater than 30°F above the RCS cold leg temperature, the effects of heat input from the secondary side, and the subsequent effects on heatup rate, must be evaluated/considered.

2. Set up either ERF or DCS Display Screen and monitor the selected Reactor Coolant Pump and Motor parameters during start-up:

- RC-3A page 440 / DCS RCP_A ___
- RC-3B page 441 / DCS RCP_B ___
- RC-3C page 442 / DCS RCP_C ___
- RC-3D page 443 / DCS RCP_D ✓ ___

NOTE

High vibration alarms should be anticipated when starting RCPs.

3. Station an operator at AI-270 (Room 57) to respond to vibration alarms. OI-RC-13 provides alarm response guidance. ___

NOTE

At low RCS Pressure, verification of positive Controlled Bleedoff Flow may NOT be possible.

4. Verify positive Controlled Bleedoff Flow for the selected RCP on ERF (page 342 or DCS "RCP Summary"):

- RC-3A ___
- RC-3B ___
- RC-3C ___
- RC-3D ✓ ___

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

> **NOTE**
> When starting the first Reactor Coolant Pump, use TR-346, Shutdown Cooling Inlet/Outlet Temp, black pen LPSI Suction Temp indication to verify NPSH requirements are met.

~~5.~~ Ensure RCS Pressure is greater than 225 psia for RCP seal protection (ensures RCS Pressure is greater than minimum NPSH Pressure per TDB-III.7.a and TDB-III.7.d, RCS Pressure and Temperature Limits). ४

~~6.~~ Ensure RCS Pressure is less than the Maximum Pressure for First RCP Start curve per TDB-III.7.a, RCS Pressure and Temperature Limits. (NA if an RCP is running). ४

~~7.~~ Ensure actual corrected level per TDB-III.1.a Pressurizer level is less than 50%. (NA if an RCP is running). ४

~~8.~~ IF the following Lockout Relays are NOT RESET,
THEN Electrical Maintenance shall be contacted and the affected relay(s) inspected.

- 86/RC-3A, Pump RC-3A Motor Current Diff ___
- 86/RC-3B, Pump RC-3B Motor Current Diff ___
- 86/RC-3C, Pump RC-3C Motor Current Diff ___
- 86/RC-3D, Pump RC-3D Motor Current Diff ___ N/A ४

~~9.~~ Ensure the Lockout Relay Switch for the selected RCP is in RESET and the Amber indicating light is ON:

- 86/RC-3A, Pump RC-3A Motor Current Diff ___
- 86/RC-3B, Pump RC-3B Motor Current Diff ___
- 86/RC-3C, Pump RC-3C Motor Current Diff ___
- 86/RC-3D, Pump RC-3D Motor Current Diff ✓ ४

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

10. For the selected RCP verify the following Annunciator is clear:

- RC-3A **REACTOR COOLANT PUMP RC-3A REVERSE ROTATION** (CB-1/2/3, A6, A5) ___
- RC-3B **REACTOR COOLANT PUMP RC-3B REVERSE ROTATION** (CB-1/2/3, A6, B5) ___
- RC-3C **REACTOR COOLANT PUMP RC-3C REVERSE ROTATION** (CB-1/2/3, A6, C5) ___
- RC-3D **REACTOR COOLANT PUMP RC-3D REVERSE ROTATION** (CB-1/2/3, A6, D5) ✓ γ

NOTE

The Oil Lift Pump shall not be run for longer than 10 minutes before starting the Reactor Coolant Pump.

11. Announce the Reactor Coolant Pump start on the Gaitronics. ______

12. Place the Oil Lift Pump for the selected RCP to AFTER START:

- RC-3A-1, Oil Lift Pump ___
- RC-3B-1, Oil Lift Pump ___
- RC-3C-1, Oil Lift Pump ___
- RC-3D-1, Oil Lift Pump ___ ______

13. Verify adequate ARRD Lube Oil Flow for the selected RCP. ERF/DCS indication shall read NORMAL: (ERF page 342 or DCS "RCP Summary")

- RC-3A F3181 ___
- RC-3B F3184 ___
- RC-3C F3187 ___
- RC-3D F3190 ___ ______

14. Prior to starting RCP, inform the Radiation Protection Department so it can monitor changing radiological conditions. ______
RP

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

15. Startup sequence for selected RCP:

15.1 Run the oil lift pump for the selected RCP a minimum of two minutes.

- RC-3A-1, Oil Lift Pump ___
- RC-3B-1, Oil Lift Pump ___
- RC-3C-1, Oil Lift Pump ___
- RC-3D-1, Oil Lift Pump ___ ______

15.2 Place the selected RCP control switch in AFTER START:

- RC-3A, RC Pump ___
- RC-3B, RC Pump ___
- RC-3C, RC Pump ___
- RC-3D, RC Pump ___ ______

15.3 IF the Reactor Coolant Pump motor amps fail to drop below 425 amps within the time listed below,
THEN place the control switch in AFTER STOP:

- RC-3A - seventeen (17) seconds ___
- RC-3B - nineteen (19) seconds ___
- RC-3C - seventeen (17) seconds ___
- RC-3D - seventeen (17) seconds ___ ______

16. Verify the following for the selected RCP:

16.1 Oil Lift Pump stops (Green indicating light ON):

- RC-3A-1 ___
- RC-3B-1 ___
- RC-3C-1 ___
- RC-3D-1 ___ ______

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

16.2 For the selected RCP verify the following Annunciator is clear:

- RC-3A **REACTOR COOLANT PUMP RC-3A REVERSE ROTATION** (CB-1/2/3, A6, A5) ____
- RC-3B **REACTOR COOLANT PUMP RC-3B REVERSE ROTATION** (CB-1/2/3, A6, B5) ____
- RC-3C **REACTOR COOLANT PUMP RC-3C REVERSE ROTATION** (CB-1/2/3, A6, C5) ____
- RC-3D **REACTOR COOLANT PUMP RC-3D REVERSE ROTATION** (CB-1/2/3, A6, D5) ____ ______

NOTE

RCP Vibration Hi annunciator may be erratic until pump speed/flow stabilizes.

16.3 For the selected RCP verify the following Annunciator is clear:

- RC-3A **REACTOR COOLANT PUMP RC-3A VIBRATION HI** (CB-1/2/3, A6, A4) ____
- RC-3B **REACTOR COOLANT PUMP RC-3B VIBRATION HI** (CB-1/2/3, A6, B4) ____
- RC-3C **REACTOR COOLANT PUMP RC-3C VIBRATION HI** (CB-1/2/3, A6, C4) ____
- RC-3D **REACTOR COOLANT PUMP RC-3D VIBRATION HI** (CB-1/2/3, A6, D4) ____ ______

NOTE

At low RCS Pressure, verification of positive Controlled Bleedoff Flow may NOT be possible.

17. Verify positive Controlled Bleedoff flow for the selected RCP: (ERF page 342 or DCS "RCP Summary")

- RC-3A F3115 ____
- RC-3B F3135 ____
- RC-3C F3155 ____
- RC-3D F3175 ____ ______

Attachment 1 - Starting Reactor Coolant Pumps (Coupled)

PROCEDURE (continued) (✓) INITIALS

18. Monitor the ERF Computer or DCS and verify all parameters are normal for the selected RCP:

- RC-3A ___
- RC-3B ___
- RC-3C ___
- RC-3D ___ ______

19. IF other RCPs are to be started,
THEN repeat Steps 2 through 18 for each RCP to be started. ______

Completed by ________________________ Date/Time _____/_____

Facility: FCS JPM # NRC S-5 Task # 0033 K/A # G 2.1.19 3.9 / 3.8 SF-4S

Title: Perform Control Room Evacuation Immediate Actions

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A toxic gas leak is requiring evacuation of the Control Room.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- EXECUTE required actions prior to a Control Room Evacuation per AOP-07, Evacuation of Control Room, Section I, Plant to Hot Shutdown.

Task Standard: Utilizing AOP-07, trip the Reactor and Turbine, secured one Main Feedwater, one Condensate, and two Heater Drain Pumps, and started Turbine Lube Oil Pumps.

Required Materials: AOP-07, Evacuation of Control Room, Rev. 17.

Validation Time: 6 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-111:

- **RUN event "No Turbine Trip from CR Edit.evt"**

Type	Item	Value	Condition
Other	THATFS_050A18O_1FREEZE (FREEZE FLAG)	1	
Other	THATFS_050A28O_1FREEZE (FREEZE FLAG)	1	
Other	THATFS_050A18O_2FREEZE (FREEZE FLAG)	1	
Other	THATFS_050A28O_2FREEZE (FREEZE FLAG)	1	
Remote/GEN	REM:86-1/G1-TRP (86-1/G1 Trip Signal)	Tripped	P10_235SD_1 eq 1 (B EHC pump to PTL)
Override	P10_102S1_1	0 (FALSE)	
Override	P10_102S1_1	1 (TRUE)	P10_235SD_1 eq 1 (B EHC pump to PTL)

EXAMINER:

PROVIDE the examinee with a copy of:

- **AOP-07, Evacuation of Control Room.**
  - **Section I, Plant to Hot Shutdown.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from AOP-07, Section I.**
Examiner Note:	**When the Reactor is tripped, an uncontrolled cooldown will begin because the Turbine has <u>NOT</u> tripped.**
Perform Step: 1√ 1 & 1.a	<u>Perform</u> the following steps prior to evacuating the Control Room: • Manually <u>trip</u> the Reactor.
Standard:	PERFORMED the following: • DEPRESSED REACTOR TRIP pushbutton on CB-4 (**critical**). • OBSERVED all Rods inserted, Reactor Power lowering, and Negative Startup Rate. (**NOT critical**).
Examiner Cue:	If cooldown is addressed: **The ATCO will perform any required Emergency Boration.** **The CRS directs you to complete AOP-07 actions.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 1 & 1.b	<u>Perform</u> the following steps prior to evacuating the Control Room: • <u>Verify</u> the Turbine is tripped as indicated by Stop and Intercept Valves indicating closed.
Standard:	OBSERVED Stop and Intercept Valves, DETERMINED Turbine was NOT tripped, and REFERRED to CONTINGENCY ACTIONS (CA).
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 2a b.1 & b.1.1) CA	<u>Trip</u> the Turbine (CB-10, 11)
Standard:	PERFORMED the following: • DEPRESSED TURBINE ST-1 MASTER TRIP PUSHBUTTON A. • DEPRESSED TURBINE ST-1 MASTER TRIP PUSHBUTTON B. • OBSERVED all Stop and Intercept Valves OPEN.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2b√
b.1 & b.1.2) CA
	Trip the Turbine by performing:
	• Stop the EHC pumps by placing **BOTH** of the following control switches in "PULL-TO-LOCK":
	  • EHC-3A
	  • EHC-3B

Standard:	PERFORMED the following: • PLACED EHC-3A, EHC PUMP handswitch in PULL-TO-LOCK (**critical**). • PLACED EHC-3B, EHC PUMP handswitch in PULL-TO-LOCK (**critical**). • OBSERVED all Stop and Intercept Valves CLOSED and DETERMINED Turbine is tripped (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√ 1 & 1.c	Perform the following steps prior to evacuating the Control Room: • Place the "43/FW" Switch in "OFF".
Standard:	PLACED 43/FW Switch in OFF.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 1 & 1.d	Perform the following steps prior to evacuating the Control Room: • Ensure no more than one Feed Pump, FW-4A/B/C is operating.
Standard:	DETERMINED only FW-4C, MFW Pump is running.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ 1 & 1.e	Perform the following steps prior to evacuating the Control Room: • Ensure no more than one Condensate Pump, FW-2A/B/C is operating.
Standard:	PERFORMED the following: • PLACED FW-2A, COND PUMP handswitch in STOP (**critical**). • OBSERVED green STOP light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

<table>
<tr><td>Perform Step: 6√
1 & 1.f</td><td>Perform the following steps prior to evacuating the Control Room:
• Stop ALL operating Heater Drain Pumps, FW-5A/B/C.</td></tr>
<tr><td>Standard:</td><td>PERFORMED the following:
• PLACED FW-5C, HTR DRN PUMP handswitch in STOP (critical).
• OBSERVED green STOP light lit (NOT critical).</td></tr>
<tr><td>Comment:</td><td>SAT ☐ UNSAT ☐</td></tr>
</table>

<table>
<tr><td>Perform Step: 7√
1 & 1.g</td><td>Perform the following steps prior to evacuating the Control Room:
• Ensure ALL of the following Turbine Lube Oil equipment is running:
  • LO-3, Turning Gear Oil Pump
  • LO-8, Motor Suction Oil Pump
  • LO-4, DC Oil Pump,
  • Turbine Lift Pumps, LO-14A/B/C</td></tr>
<tr><td>Standard:</td><td>PERFORMED the following:
• PLACED LO-3, TURNING GEAR OIL PUMP handswitch in START (critical).
• PLACED LO-8, MOTOR SUCTION OIL PUMP handswitch in START (critical).
• PLACED LO-4, EMGY BRG OIL PUMP handswitch in START (critical).
• PLACED LO-14A, TURBINE BEARING LUBE OIL LIFT OIL PUMP handswitch in START (critical).
• PLACED LO-14B, TURBINE BEARING LUBE OIL LIFT OIL PUMP handswitch in START (critical).
• PLACED LO-14C, TURBINE BEARING LUBE OIL LIFT OIL PUMP handswitch in START (critical).
• OBSERVED red START lights lit (NOT critical).
• OBSERVED six white DISCH PRESS lights lit (NOT critical).</td></tr>
<tr><td>Terminating Cue:</td><td>This JPM is complete.</td></tr>
<tr><td>Comment:</td><td>SAT ☐ UNSAT ☐</td></tr>
</table>

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **A toxic gas leak is requiring evacuation of the Control Room.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **EXECUTE required actions prior to a Control Room Evacuation per AOP-07, Evacuation of Control Room, Section I, Plant to Hot Shutdown.**

Fort Calhoun Station
Unit No. 1

AOP-07

EVACUATION OF CONTROL ROOM

Change No.	ECs 45086, 66076
Reason for Change	EC 45086 - 1A2/1A4 breaker replacement under. EC 66076 - Correct labels
Initiator	C. Smith, A. Hedges
Preparer	S. Lindquist
Issued	05-29-15 3:00 pm

TABLE OF CONTENTS

Section I - Plant to Hot Shutdown

1.0 PURPOSE

This procedure provides the steps necessary to shutdown the Plant in the event of the evacuation of the Control Room for reasons other than a fire. AOP-06, Fire Emergency, addresses Control Room evacuation due to fire. AOP-07, Evacuation of Control Room, is written to ensure the Control Room Supervisor at the Alternate Shutdown Panel maintains control and coordinates this abnormal evolution.

2.0 ENTRY CONDITIONS

Any condition, except fire, requiring Control Room evacuation.

Section I - Plant to Hot Shutdown

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. **CAUTIONS**

1. Charging to the RCS may cause overpressurization due to the isolation of Letdown and RCS Heatup.

B. **NOTES**

1. The LO's chief objective is to gain access to the Control Room as soon as possible so that normal Reactor control can be re-established.
2. HCV-1107B and HCV-1108B can be positioned when Instrument Air is available using the Hand Load Controllers on panel AI-179 when their control switches are in throttle. If these switches are in open, or Instrument Air is lost, the valves will remain open.

Section I - Plant to Hot Shutdown

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Perform the following steps prior to evacuating the Control Room:	
a. Manually trip the Reactor.	a.1 **IF** the Reactor will **NOT** trip, **THEN** open **BOTH** Clutch Power Supply Circuit Breakers (AI-57): • "CB-AB" • "CB-CD"
b. Verify the Turbine is tripped as indicated by Stop and Intercept Valves indicating closed. (continue)	b.1 Trip the Turbine by performing Steps 1), 2), or 3): 1) Trip the Turbine (CB-10,11). 2) Stop the EHC pumps by placing **BOTH** of the following control switches in "PULL-TO-LOCK": • EHC-3A • EHC-3B (continue)

Section I - Plant to Hot Shutdown

INSTRUCTIONS

1. (continued)

c. Place the "43/FW" Switch in "OFF".

d. Ensure no more than one Feed Pump, FW-4A/B/C is operating.

e. Ensure no more than one Condensate Pump, FW-2A/B/C is operating.

f. Stop **ALL** operating Heater Drain Pumps, FW-5A/B/C.

(continue)

CONTINGENCY ACTIONS

b.1 (continued)

3) **(LOCAL)** Trip the Turbine using IA-3/ST-1 and IB-3/ST-1 "TURBINE TRIP PUSHBUTTONS" (Turbine Bldg stairway to Cable Spread Room).

Section I - Plant to Hot Shutdown

INSTRUCTIONS | **CONTINGENCY ACTIONS**

1. (continued)

   g. Ensure **ALL** of the following Turbine Lube Oil equipment is running:

   - LO-3, Turning Gear Oil Pump
   - LO-8, Motor Suction Oil Pump
   - LO-4, DC Oil Pump
   - Turbine Lift Pumps, LO-14A/B/C

   h. Announce and repeat the following over the Plant communications system:

   "The Control Room has been evacuated. All operators report to the Alternate Shutdown Panel."

   i. Direct each Operator to obtain a transceiver.

   j. Ensure extra transceivers are taken to AI-185, "ALTERNATE SHUTDOWN PANEL" (West Upper Electrical Penetration Room).

(continue)

Section I - Plant to Hot Shutdown

INSTRUCTIONS	CONTINGENCY ACTIONS
1. (continued)	
k. (**CRS**) Obtain the AOP-06 Keys.	
l. (**STA**) Perform the Safety Function Status Check (TSC).	
m. (**LO**) Attempt to regain access to the Control Room.	
n. IMPLEMENT the Emergency Plan.	
o. (**CR Communicator**) Assist the Shift Manager in the TSC.	
p. Direct the EONA to perform Attachment B, EONA Responsibilities.	

Facility: FCS   JPM # NRC S-6   Task # 0369   K/A # 026.A4.05   3.5 / 3.5   SF-5

Title: Reset Containment Spray Actuation Signal

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A Main Steam Line Break has occurred inside Containment.
- EOP- 05, Uncontrolled Heat Extraction, is in progress.
- Containment Spray Pumps SI-3A and SI-3B are running.
- Containment Pressure is less than 3 psig.
- All Containment Cooling and Filtering Units are in service.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- RESET the Containment Spray Actuation Signal and return Containment Spray Pumps to Standby per EOP/AOP Floating Step F, Containment Spray Termination.

Task Standard: Utilizing Floating Step F, reset CPHS, CSAS, and SGLS lockout relays and secured Containment Spray Pumps.

Required Materials: EOP/AOP Floating Steps, Rev. 7.

Validation Time: 15 minutes   Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:
INITIALIZE to IC-120:

Type	Item	Value	Condition
MALF/SGN	SGN01B (Main Steam Line B Leak Inside Containment)	0.25	**Simulator is frozen >30 minutes into a steam header rupture in containment

EXAMINER:
PROVIDE the examinee with a copy of:

- **EOP/AOP Floating Step F, Containment Spray Termination.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from EOP/AOP Floating Steps, FS-F.**

NOTE

Stopping SI-3A or SI-3B will result in closure of one spray valve, HCV-344 or HCV-345 by interlock which will extend the time to RAS.

CAUTION

Containment Spray may affect proper operation of RCPs, non-qualified equipment, Containment Sump, and instrumentation inside the Containment. When the termination criterion is satisfied, Containment Spray should be promptly secured

Perform Step: 1 1	**IF** Containment Spray has been initiated **AND ALL** of the following conditions are satisfied: • Two CS pumps are operating • Containment pressure is less than 60 psig and **NOT** rising • At least one VA-3A/B in service • At least one VA-7C/D in service **THEN** perform the following:
Standard:	DETERMINED all conditions are met per Initial Conditions.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√ 1.a	Ensure only **ONE** CS pump is operating.
Standard:	PERFORMED the following: • PLACED SI-3A or SI-3B, CNTMT SPRAY PUMP in PULL-TO-LOCK (**critical**). • OBSERVED pump indicating lights off (**NOT critical**).
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Applicant may place HCV-344 <u>or</u> HCV-345, CNTMT SPRAY VLV CONTROL SWITCH in OVERRIDE, but is not required.**
Perform Step: 3 1.b	<u>Ensure</u> only **ONE** of the following valves is open: • HCV-344 • HCV-345

Standard:	PERFORMED the following:
	• OBSERVED that red OPEN light lit for one valve and green CLOSED light lit for the other valve.
	  • HCV-344 is open when SI-3B is running
	  • HCV-345 is open when SI-3A is running

Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 1.c	<u>Ensure</u> total CS flow is at least 1800 gpm.
Standard:	OBSERVED approximately 2400 gpm of combined flow on FI-343 **<u>and</u>** FI-342 SPRAY FLOW meters.
Comment:	**SAT ☐ UNSAT ☐**

<u>NOTE</u>

Terminating Containment Spray prior to resetting actuation relays will require increased monitoring of containment parameters.

Perform Step: 5 2	**IF** CS pump(s) are operating, **AND ALL** of the following conditions are satisfied: • Containment pressure is less than 30 psig and stable or lowering • Containment Spray is **NOT** required for Containment cooling • At least one VA-3A/B in service • At least one VA-7C/D in service **THEN** <u>terminate</u> Containment Spray by performing the following:
Standard:	DETERMINED all conditions are met per Initial Conditions.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Applicant will place valve left open at Perform Step 3 in OPEN.**
Perform Step: 6√ 2.a	Place the control switches for the open valve(s) in "OPEN": • HCV-344 • HCV-345
Standard:	PERFORMED the following for the valve that is open: • PLACED HCV-344 **or** HCV-345, CNTMT SPRAY VLV CONTROL SWITCH in OPEN (**critical**). • OBSERVED red OPEN light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Applicant will place SI-3A or SI-3B and SI-3C in PULL-TO-LOCK.**
Perform Step: 7√ 2.b	Place all CS pumps in "PULL-TO-LOCK": • SI-3A • SI-3B • SI-3C
Standard:	PERFORMED the following: • PLACED SI-3A, CNTMT SPRAY PUMP in PULL-TO-LOCK (**critical**). • PLACED SI-3C, CNTMT SPRAY PUMP in PULL-TO-LOCK (**critical**) • OBSERVED all pump indicating lights off (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Applicant will place valve left OPEN at Perform Step 6 in CLOSE.**
Perform Step: 8√ 2.c	Close **BOTH** Containment Spray Valves:
Standard:	PERFORMED the following: • PLACED HCV-344 **or** HCV-345, CNTMT SPRAY VLV CONTROL SWITCH in OVERRIDE **or** AUTO (**critical**). • OBSERVED green CLOSE light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9√ 2.d	Place **BOTH** control switches in "AUTO": • HCV-344 • HCV-345
Standard:	PERFORMED the following: • PLACED HCV-344, CNTMT SPRAY VLV CONTROL SWITCH in AUTO (**critical**). • PLACED HCV-345, CNTMT SPRAY VLV CONTROL SWITCH in AUTO (**critical**) • OBSERVED white AUTO light off (**NOT critical**).
Comment:	**SAT** ☐ **UNSAT** ☐

NOTES

1. Resetting CPHS and SGLS Lockout Relays may reset SGIS. HCV-1105 and HCV-1106 may reopen.
2. Resetting PPLS, CPHS or SGLS Lockout Relays will reset Containment Spray.

Perform Step: 10 3	**IF** resetting actuation relays, **THEN** perform the following:
Standard:	DETERMINED actuation relays will be RESET.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 11√ 3.a	**IF** Containment pressure less than or equal to 3 psig, **THEN** reset all of the following relays: • 86A/CPHS • 86B/CPHS
Standard:	PERFORMED the following: • TURNED 86A/CPHS relay in CLOCKWISE direction until LATCHED (**critical**). • TURNED 86B/CPHS relay in CLOCKWISE direction until LATCHED (**critical**) • OBSERVED black relay flag and amber light lit (**NOT critical**).
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 12√ 3.b	<u>Reset</u> **ALL** of the following relays: • 86A1/CPHS • 86B1/CPHS
Standard:	PERFORMED the following: • TURNED 86A1/CPHS relay in CLOCKWISE direction until LATCHED (**critical**). • TURNED 86B1/CPHS relay in CLOCKWISE direction until LATCHED (**critical**) • OBSERVED black relay flag and amber light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 13√ 3.c	<u>Reset</u> **ALL** of the following CSAS relays: • 86A/CSAS • 86B/CSAS • 86A1/CSAS • 86B1/CSAS
Standard:	PERFORMED the following: • TURNED 86A/CSAS relay in CLOCKWISE direction until LATCHED (**critical**). • TURNED 86B/CSAS relay in CLOCKWISE direction until LATCHED (**critical**) • TURNED 86A1/CSAS relay in CLOCKWISE direction until LATCHED (**critical**). • TURNED 86B1/CSAS relay in CLOCKWISE direction until LATCHED (**critical**) • OBSERVED black relay flag and amber light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following actions are performed at CB-4.**

Perform Step: 14√
3.d
	Reset SGLS by performing the following:
	• Block SGLS-A and SGLS-B by performing the following:
	• Place the SGLS Block key into the SGLS Block key switch.
	• Block SGLS-A and SGLS-B by turning key to "BLOCK".
	• Verify at least one of the following SGLS Blocked alarms annunciates (CB-4; A8):
	  • "SGLS "A" BLOCKED"
	  • "SGLS "B" BLOCKED"
	Time _____

Standard:	• DETERMINED Steam Generator Low Pressure Signal will be RESET. (**NOT critical**) • REMOVED SGLS Block Key from Key Holder and PLACED SGLS Block Key into SGLS Block Key Switch. (**critical**) • TURNED Key in SGLS Block Key Switch to BLOCK position. (**critical**) • VERIFIED Annunciator Panel A8 SGLS "A" BLOCKED (Window D-4L) **or** SGLS "B" BLOCKED (Window D-5U) in alarm and RECORDED time. (**NOT critical**)
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 15√ 3.d.1).c).2)	Reset **BOTH** of the following SGLS relays: • 86A/SGLS • 86B/SGLS
Standard:	PERFORMED the following: • TURNED 86A/SGLS relay in CLOCKWISE direction until LATCHED (**critical**). • TURNED 86B/SGLS relay in CLOCKWISE direction until LATCHED (**critical**). • OBSERVED black relay flag (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 16√ 3.e.1)	**IF** returning CS to standby, **THEN** <u>perform</u> the following: • <u>Place</u> CS Pumps SI-3A/B/C to "AFTER STOP".
Standard:	PERFORMED the following: • PLACED SI-3A, CNTMT SPRAY PUMP in AFTER STOP (**critical**). • PLACED SI-3B, CNTMT SPRAY PUMP in AFTER STOP (**critical**). • PLACED SI-3C, CNTMT SPRAY PUMP in AFTER STOP (**critical**) • OBSERVED green STOP lights lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 17 3.e.2)	<u>Place</u> **BOTH** Containment Spray Valves in "AUTO": • HCV-344 • HCV-345
Standard:	DETERMINED HCV-344 **<u>and</u>** HCV-345 already in AUTO.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **A Main Steam Line Break has occurred inside Containment.**
- **EOP- 05, Uncontrolled Heat Extraction, is in progress.**
- **Containment Spray Pumps SI-3A and SI-3B are running.**
- **Containment Pressure is less than 3 psig.**
- **All Containment Cooling and Filtering Units are in service.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **RESET the Containment Spray Actuation Signal and return Containment Spray Pumps to Standby per EOP/AOP Floating Step F, Containment Spray Termination.**

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Stopping SI-3A or SI-3B will result in closure of one spray valve, HCV-344 or HCV-345 by interlock which will extend the time to RAS.

****

CAUTION

Containment Spray may affect proper operation of RCPs, non-qualified equipment, Containment Sump, and instrumentation inside the Containment. When the termination criterion is satisfied, Containment Spray should be promptly secured.

****

1. **IF** Containment Spray has been initiated **AND** **ALL** of the following conditions are satisfied:

   - Two CS pumps are operating
   - Containment pressure is less than 60 psig and **NOT** rising
   - At least one VA-3A/B inservice
   - At least one VA-7C/D inservice

   THEN perform the following:

   a. Ensure only **ONE** CS pump is operating.

   (continue)

1.1 **IF** Containment pressure can not be maintained less than 60 psig, **THEN** perform the following:

****

CAUTION

Do **NOT** run SI-3B and SI-3C at the same time.

****

   a. Ensure at least **TWO** CS pumps are operating:

   - SI-3A
   - SI-3B
   - SI-3C

   (continue)

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS	CONTINGENCY ACTIONS
1. (continued)	1.1 (continued)
b. Ensure only **ONE** of the following valves is open: • HCV-344 • HCV-345	b.1 Ensure HCV-344 and HCV-345 are open.
c. Ensure total CS flow is at least 1800 gpm.	

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

NOTE

Terminating Containment Spray prior to resetting actuation relays will require increased monitoring of containment parameters.

INSTRUCTIONS

2. **IF** CS pump(s) are operating,

   AND ALL of the following conditions are satisfied:

   - Containment pressure is less than 30 psig and stable or lowering
   - Containment Spray is **NOT** required for Containment cooling
   - At least one VA-3A/B inservice
   - At least one VA-7C/D inservice

   THEN terminate Containment Spray flow by performing the following:

   a. Place the control switches for the open valve(s) in "OPEN":

      - HCV-344
      - HCV-345

(continue)

CONTINGENCY ACTIONS

2.1 **IF** containment pressure can not be maintained less than 40 psig,

   THEN initiate Containment Spray by performing the following:

   a. Start **ONE** CS Pump:

      - SI-3A
      - SI-3B

   b. Open **ONE** Containment Spray Valve:

      - HCV-344
      - HCV-345

   c. Ensure total CS flow is at least 1800 gpm.

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS	CONTINGENCY ACTIONS
2. (continued)	
b. Place all CS pumps in "PULL-TO-LOCK": • SI-3A • SI-3B • SI-3C	
c. Close **BOTH** Containment Spray Valves: • HCV-344 • HCV-345	c.1 **IF** either Containment Spray Valve fails to close, **THEN** manually close the open valve(s) using the handwheel (Room 59).
d. Place **BOTH** control switches in "AUTO": • HCV-344 • HCV-345	

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTES

1. Resetting CPHS and SGLS Lockout Relays may reset SGIS. HCV-1105 and HCV-1106 may reopen.
2. Resetting PPLS, CPHS or SGLS Lockout Relays will reset Containment Spray.

3. **IF** resetting actuation relays,

   THEN perform the following:

   a. **IF** Containment pressure less than or equal to 3 psig, **THEN** reset all of the following relays:

      - 86A/CPHS
      - 86B/CPHS

   b. Reset **ALL** of the following relays:

      - 86A1/CPHS
      - 86B1/CPHS

   c. Reset **ALL** of the following CSAS relays:

      - 86A/CSAS
      - 86B/CSAS
      - 86A1/CSAS
      - 86B1/CSAS

      (continue)

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS	CONTINGENCY ACTIONS
3. (continued)	
d. Reset SGLS by performing the following:	
1) Block SGLS-A and SGLS-B by performing the following:	
a) Place the SGLS Block key into the SGLS Block key switch.	
b) Block SGLS-A and SGLS-B by turning key to "BLOCK".	
c) Verify at least one of the following SGLS Blocked alarms annunciates (CB-4; A8): • "SGLS "A" BLOCKED" • "SGLS "B" BLOCKED" Time: ____________	c).1. **IF** neither SGLS Blocked alarm annunciates, **THEN** continue attempts to block SGLS until at least one alarm annunciates.

(continue)

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS

CONTINGENCY ACTIONS

3. (continued)

2) Reset **BOTH** of the following SGLS relays:

- 86A/SGLS
- 86B/SGLS

e. **IF** returning CS to standby, **THEN** perform the following:

1) Place CS Pumps SI-3A/B/C to "AFTER STOP".

2) Place **BOTH** Containment Spray Valves in "AUTO":

- HCV-344
- HCV-345

2.0 FLOATING STEPS

A. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS	CONTINGENCY ACTIONS
4. **IF** Containment temperature is less than 120°F, **AND NOT** required for cooling, **THEN** stop **BOTH** of the following Containment Vent Fans: • VA-7C • VA-7D	
5. **IF** cooled SI flow is required, **AND** RAS has occurred, **THEN** IMPLEMENT Attachment HR-29, Cooled SI Flow With RAS.	

Facility: FCS JPM # NRC S-7 Task # 0344 K/A # 064.A4.06 3.9 / 3.9 SF-6

Title: Parallel and Load Emergency Diesel Generator

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- DG-1 has been manually started and is at idle speed.
- YCV-871 G/H/E Inlet and Exhaust Dampers have been verified OPEN.
- Jacket water temperature is 128°F.
- All Prerequisites are met.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- PARALLEL and LOAD Emergency Diesel Generator DG-1 with Bus 1A3 per OI-DG-1, Diesel Generator Operation, Attachment 1, Idle Speed Start and Loading.
- START at Step 4.b.
- LOAD DG-1 to 2000 KW.

Task Standard: Utilizing OI-DG-1, raised DG-1 speed, paralleled and loaded to Bus 1A3, then tripped DG-1 Breaker when load rose uncontrollably.

Required Materials: OI-DG-1, Diesel Generator Operation, Rev. 63.

Validation Time: 15 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-111. Load and execute scenario "NRC JPM S7"

Type	Item	Value	Condition
Override	P20_185_3 (DG/1 Governor Sel SW Raise Position)	1	Condition: H_P20_033_1 ge 800 (When DG Watts are greater than 800, override governor switch to raise)

BOOTH OPERATOR NOTE:

- **After each JPM, VERIFY Synchroscope Switch is moved from the D1/BUS 1A3 Sync Switch position prior to performance by the next examinee.**

EXAMINER:

PROVIDE the examinee with a copy of:

- **OI-DG-1, Diesel Generator Operation.**
    - **Attachment 1, Idle Speed Start and Loading, INITIALED through Step 4.a.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OI-DG-1, Attachment 1.**
Perform Step: 1√ 4.b	Place CS-65/D1, Diesel Generator D1 Governor, to Raise until the Diesel Speed is 900 rpm.
Standard:	PERFORMED the following: • PLACED CS-65/D1, DIESEL GENERATOR D1 GOVERNOR, to RAISE position until Diesel Speed is 900 rpm. (**critical**). • OBSERVED Diesel speed rising to 900 rpm on Diesel Generator DG-1 Engine Tachometer (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 4.c	Verify the Generator Field flashed by performing one of the following: • Ready to Load light is ON (AI-30A) **OR** • Generator frequency is responding
Standard:	OBSERVED the following: • Ready to Load red light is ON at Panel AI-30A. • Generator frequency is 60 Hz at 900 rpm.
Examiner Note:	**Automatic field flashing occurs at approximately 700 rpm.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 4.d	**(LOCAL)** Inspect field flash circuitry by performing the following: • Verify that Control Relay 2CR in Panel AI-133A is not energized. • Verify that Field Flash Current Limiting Resistors (1R4, 1R5, 1R6, and 1R7) in Panel AI-133A are not damaged due to overheating. Contact System Engineer if damage is suspected.
Standard:	CONTACTED Auxiliary Operator at DG-1 to verify Field Flash Circuitry.
Examiner Cue:	**Auxiliary Operator reports Field Flash Circuitry is satisfactory.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 4.e	**(LOCAL)** Place AI-133A-S4, Diesel Generator DG-1 Electronic Droop Control Switch, in ENABLED. (AI-133A)
Standard:	CONTACTED Auxiliary Operator at DG-1 to PLACE AI-133A-S4, Diesel Generator DG-1 Electronic Droop Control Switch, in ENABLED position.
Examiner Cue:	**Auxiliary Operator reports Electronic Droop Control Switch in ENABLED position.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The panel holding the Synchroscope and Running and Incoming Volts indications is "hinged" and can be rotated for better viewing.**
Perform Step: 5√ 4.f	Place D1/BUS 1A3 Sync Switch to ON.
Standard:	PERFORMED the following: • LOCATED and INSERTED Synchroscope Switch into D1/BUS 1A3 SYNC SWITCH position and TURNED to ON (**critical**). • OBSERVED Synchroscope rotation (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 4.g	Adjust CS-90/D1, Diesel Generator D1 Voltage Regulator, until the RUNNING VOLTS is approximately matched to the INCOMING VOLTS on the Synchroscope or the ERF DGD Display.
Standard:	PERFORMED the following: • ADJUSTED CS-90/D1, DIESEL GENERATOR D1 VOLTAGE REGULATOR switch until RUNNING VOLTS is approximately MATCHED (within ~100 volts) to INCOMING VOLTS (**critical**).
Comment:	**SAT ☐ UNSAT ☐**

NOTE Recommended synchroscope speed is less than 1 revolution per 10 seconds.	
Perform Step: 7√ 4.h	Adjust CS-65/D1 until the Synchroscope is rotating slowly in the **FAST** direction.
Standard:	ADJUSTED CS-65/D1, DIESEL GENERATOR D1 GOVERNOR until Synchroscope is ROTATING SLOWLY in FAST direction and less than 1 revolution per 10 seconds.
Comment:	**SAT ☐ UNSAT ☐**

NOTE
Steps 4.i and 4.j may be performed without the procedure in hand. Sign-offs may be completed after these steps are performed.

CAUTIONS
1. Load must be immediately picked up following closure of 1AD1 to prevent motorizing the Diesel Generator. 2. Governor controls are extremely sensitive.

Perform Step: 8√ 4.i	WHEN the Synchroscope is between 11 and 12 **O'CLOCK**, THEN close 1AD1 BREAKER.
Standard:	PERFORMED the following: • When Synchroscope was between 11 and 12 o'clock, PLACED 1AD1 BREAKER in CLOSE position (**critical**). • OBSERVED red CLOSE light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9√ 4.j	Place CS-65/D1 to Raise to pick up 250-350 KW. • Time _____
Standard:	PERFORMED the following: • PLACED CS-65/D1, DIESEL GENERATOR D1 GOVERNOR in RAISE and PICKED UP 250 KW to 350 KW load (**critical**). • OBSERVED load rising on DG-1 WATT METER (**NOT critical**). • RECORDED Time at Step 4.j (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 10 4.k	Place D1/BUS 1A3 Sync Switch to OFF.
Standard:	PLACED D1/BUS 1A3 SYNC SWITCH in OFF.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 11 4.l	IF the Diesel is loaded AND Y3287A, ERF 1A3 Bus Voltage, is greater than 4375 VAC, THEN immediately notify the System Engineer.
Standard:	DETERMINED Bus 1A3 voltage is normal.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**When DG-1 load is > 800 KW and CS-65/D1 is placed in RAISE, DG-1 load will continue to rise to the maximum value.**
Examiner Note:	**The following steps represent the Alternate Path of this JPM.**

NOTE

1. Load should be maintained below the 2000 hr Rating vs Ambient Temp curve per TDB-III.26A Figure 1, DG-1 Output Power Rate.
2. Power factor may be determined by using TDB-III.26, Diesel Generator Capability Curve.
3. Current is normally limited to 400 amps at 2500 KW.
4. Diesel Generator manual loading and unloading rates should be maintained at less than 500 KW per minute.
5. Steps 4.m and 4.n may be repeated as necessary while the diesel is loaded. Sign-offs may be completed after these steps are performed.

Perform Step: 12√ 4.m	Place CS-65/D1 to RAISE picking up the required DG-1 Load.
Standard:	PERFORMED the following: • PLACED CS-65/D1, DIESEL GENERATOR D1 GOVERNOR in RAISE and PICKED UP load (**critical**). • DETERMINED DG-1 load rising out of control and PLACED 1AD1 BREAKER in TRIP position (**critical**). • OBSERVED green TRIP light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 13	Inform Control Room Supervisor of problem.
Standard:	INFORMED Control Room Supervisor DG-1 Output Breaker tripped due to excessive Diesel loading.
Terminating Cue:	**The CRS has been notified. This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **DG-1 has been manually started and is at idle speed.**
- **YCV-871 G/H/E Inlet and Exhaust Dampers have been verified OPEN.**
- **Jacket water temperature is 128°F.**
- **All Prerequisites are met.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **PARALLEL and LOAD Emergency Diesel Generator DG-1 with Bus 1A3 per OI-DG-1, Diesel Generator Operation, Attachment 1, Idle Speed Start and Loading.**
- **START at Step 4.b.**
- **LOAD DG-1 to 2000 KW.**

Fort Calhoun Station
Unit No. 1

OI-DG-1

OPERATING INSTRUCTION

DIESEL GENERATOR NO. 1

Change No.	EC 60488, EC 60326, EC 58621, EC 57709, EC 59439, EC 58800, EC 57087, EC 56453, EC 56460, EC 58551
Reason for Change	Apply human factors for procedure use upgrade. (EC 60488) Include EC number for Tracking these procedures. (EC 60326) Clarify step 3 of Attachment 4. (EC 58621) Add step to declare diesel operable after barring over if desired. (EC 57709) Clarification of independent sign off's in step 1 and 2 of Attachment 4. (EC 59439) Add a note to the prerequisites. (EC 58800) Add proper equipment designator for SA-1-1 and SA-1-2. (EC 57087) Match precaution with DG-1. (EC 56453) Change Attachment 5 Step 5 to be a subset of Step 4. (EC 56460) Enhancement to conform with FCSG-9. (EC 58551) This is a Major Revision as defined in FCSG-8 and rev bars are not used.
Requestor	H. Hartwell, K. Bessey, C. Smith, Ca. Hayes, D. Gautreau, D. Hoffine, N. Vassios, B. Godfrey, T. Bettcher
Preparer	Ca. Hayes
Issue Date	09-12-13 3:00 pm

DIESEL GENERATOR NO. 1

SAFETY RELATED

ATT PURPOSE PAGE

PRECAUTIONS

1. Pulling the engine fuel rack lever out away from the engine will shut down the engine.

2. The Turbo Oil Circulating Pump and Lube Oil Circulation Pump must remain in service after shutdown to remove heat from the Turbo-Charger.

3. 183/MES, Master Emergency Switch, (AI-133A) out of NORMAL will deenergize the associated 4160 VAC Bus.

4. Air Start Motor System D1-163 must be in NO. 1 for the Air Start System to alternate when the Diesel is in Emergency Standby.

5. Diesel Generators should not be started or loaded for an anticipated loss of Offsite Power.

6. The Diesel will consume lube oil at two gallons per hour. Oil addition will be required on an extended run.

7. Lube oil should always be visible in the bottom sightglass and only visible in the upper sightglass when the engine is running.

8. Primary or Secondary Air Banks must have a minimum pressure of 190 psig. Pressures between 190 psig and 150 psig require Tech Spec 2.7 LCO entry. Pressure at or below 150 psig renders the system inoperable.

9. In AUTO, VA-759A will remain energized for ten minutes after DG-1 is shutdown.

   - VA-759A will automatically start after the Diesel Generator Breaker is closed.
   - Compressor Thermal Overload Protection will reset automatically. The Evaporator Blower will continue to run when the compressor has tripped on Thermal Overload.
   - VA-759A will operate in HAND regardless of the Diesel Generator Breaker position.

PRECAUTIONS (continued)

10. Operation at Fast Speed (900 rpm) with No Load or Light Load (less than 500 kw) should be minimized due to the following:

    - Excessive oil consumption
    - Build up of carbon deposits in the Exhaust Manifold
    - Oil accumulation in the Exhaust System and Air Box creating a fire hazard
    - Accelerated Turbo Charger Gear Train wear
    - Oil will be accumulated in the Exhaust Manifold and Air Box after four hours.

11. Extended operation at Idle Speed (500 rpm) is allowed.

12. Running the Diesel at less than 900 rpm after Field Flash will damage the Static Exciter.

13. Due to high velocity air flow near the generator and radiator, maintain positive control over any hand held or loose objects to avoid accidental ingestion of objects into the generator blower, inlet air filters, or the radiator fan.

14. Use caution while the Diesel Engine is running due to hot piping, hot surfaces, and rotating equipment.

15. Prior to declaring Diesel Generator No. 1 inoperable, consider posting Protected Equipment Signs on Diesel Generator No. 2. If posted, remove signs when Diesel Generator No. 1 is operable.

16. At the completion of Attachment 2, Unloading and Stopping, the diesel governor will be set at a nominal 907.5 rpm (60.5 Hz) via CS-65/D1. No further adjustments to this switch are permissible. The DG-1 governor switch MUST remain in this "As Left" position to ensure DG operability.

17. Coolant leaks from jacket water pumps may occur during engine barring, start-up, and warm-up periods. The longer an engine has been shut down, even with cooling system drained, the more prone it is to develop this type of leak. Therefore, such leaks can be considered normal provided that they cease within 15 minutes after engine warm-up.

18. Opening of the (Primary or Secondary) air receiver drain valves (SA-111, SA-112, SA-109, or SA-110) results in the associated starting air subsystem being inoperable. If this is performed on the "required" (typically the Primary) starting air subsystem, the emergency diesel generator, DG-1 is inoperable. Remaining stationed at the air receiver drain valve (SA-111, SA-112, SA-109, or SA-110) until the valve is closed administratively replaces the log entry required for the inoperable Diesel Generator.

PRECAUTIONS (continued)

19. Diesel exhausts have the ability, under certain wind conditions, to be drawn into plant ventilation supply systems. It is not uncommon for these exhaust fumes to be conductive enough in nature to affect fire protection systems, particularly air sampling, and ion detector devices. A rapid response could be necessary to avoid unanticipated actuation of fire suppression systems.

20. Operation of DG-1 with AI-133A-S4 in the ENABLED position causes DG-1 to be INOPERABLE.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specification:
   - 2.7: Electrical Systems
2. USAR:
   - 8.4.1: Diesel - Generators
3. Ongoing Commitments:
   - AR 10546, LER 90-010
   - AR 16852, LER 95-006
   - AR 36462, LIC 04-0129
4. Technical Manuals:
   - TD G100.0020: Diesel Generator Technical Manual
   - TD G100.1670: Power Pointer 1P-93 for Electromotive Diesel Engines
   - Kaeser Compressors Vendor Manual
5. Other:
   - AR 11562 (LIC-91-0058): Barring Over the Diesel Generators
   - FC-OPS-092-96: Management Expectations Concerning Diesel Generator Inoperability
   - AR 12033, PED-SYE-92-0001

6. Drawings

Drawings	File	Description
B120F07001, Sh.1	17390	DG-1 Air Starting Air System
B120F04002, Sh.1	17388	Jacket Water
B120F03001, Sh.1	48725	Lube Oil System
11405-M-262, Sh.1	16303	Fuel Oil
B120F15502, Sh.1	17409	480 VAC 125 VDC Dist. Panel and 120 VAC Dist.
B120F15503,. Sh.1	17410	480 VAC Auxiliary Systems
B120F15503, Sh.2	17411	480 VAC Auxiliary Systems
D-4665	57226	DG-1 Diesel Generator One Line Diagram

APPENDICES

Attachment 1 - Idle Speed Start and Loading

PREREQUISITES

(✓) INITIALS

1. Procedure Revision Verification:

   Revision No. 63 Date: today — γ

NOTE
The vendor recommends barring over the engine if the Diesel has not run in the last 24 hours.

2. Record the date and time the diesel was last run.

   Date/Time yesterday / 0800 — γ

3. IF greater than 24 hours,
   THEN perform Step 4. — N/A γ

4. IF Barring Over the engine is required,
   THEN Attachment 6 has been completed. — N/A γ

5. Lube oil is visible in the bottom sightglass (Engine Control Panel). — γ

6. Ensure LO-40-1, DG-1 Turbo Oil Circulating Pump is running (Local). — γ

7. Ensure the **LUBE OIL PRESSURE LOW** (DG1, D1L, A1) alarm is clear. — γ

8. Primary or Secondary Air Bank Pressure(s) is greater than or equal to 200 psig. (PI-3358, PI-3359) — γ

9. Record 1A3 Bus Voltage from the ERF:

   - Y3287A = 4220 VAC — γ

NOTE
If bus voltage is greater than 4300 volts, placing the DG on the bus may increase bus voltage high enough to cause the DG Voltage Regulator control to reach maximum voltage.

10. IF Y3287A, ERF 1A3 Bus Voltage, is greater than 4300 VAC,
    THEN contact the System Engineer for further directions. — N/A γ

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (✓) INITIALS

NOTES

1. After failure of DG-2, DG-1 must be started to verify operability within 8 hours OR ensure the absence of a Common Mode Failure within 8 hours (T.S. 2.7(2)j). The 8 hours should be utilized to determine the cause of the failure.
2. After a Common Mode Failure is found, DG-1 must be started immediately to confirm operability even when time remains in the 8 hour provision.
3. If running the only operable diesel below 907.5 RPM to confirm operability, it is not necessary to enter the Emergency Plan due to both Diesel Generators being considered inoperable. EPIP-OSC-1 states "Emergency Action Levels are not intended to be used during approved testing situations where abnormal equipment status is expected".
4. DG-1 is inoperable if its Speed Governor is not set to 907.5 rpm. **[AR 16852]**
5. This procedure will result in the inoperability of DG-1.

1. IF DG-1 is the only operable Diesel Generator, THEN ensure the following:

   a. DG-1 is declared inoperable and Technical Specifications 2.0.1 entered for inoperability of both Diesel Generators. N/A

   b. Reference SO-O-1, Section 5.4 for T.S. 2.0.1 guidance.

   c. All other Plant Work is stopped.

   d. All Switchyard Work is stopped and personnel are evacuated.

   e. Continue to work to restore operability of DG-1 and DG-2. N/A US
   CRS

2. Declare DG-1, Emergency Diesel Generator #1, inoperable and enter Technical Specification 2.7. US
   CRS

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued)

(✓) INITIALS

> **NOTE**
> When a diesel generator is running, AI-230, Room 81 Fire Detection Panel may go into alarm.

3. Perform the following for an Idle Speed start:

   a. Station an Operator at DG-1 to locally observe rundown of the governor. ʃ

   > **NOTE**
   > Steps 3.b and 3.c must be performed concurrently.

   b. Place AND hold CS-65/D1 Diesel Generator D1 Governor in the LOWER position until amber light is ON. (CB-20) ʃ

   > **NOTE**
   > Full rundown of the governor consists of approximately 8 to 9 revolutions of the Speed Setting knob in the counterclockwise direction.

   c. **(LOCAL)** Observe rundown of the Diesel Generator Governor. ʃ

   d. **(LOCAL)** Set the governor speed droop dial to the scribe mark on the scale. ʃ

   M
   Ind Verif

   e. Select the required starting air motors by placing the D1-163 Air Start Motor System to required position:

   - No. 1 (Normal Operation) ✓
   - No. 2 (Secondary Start Only)

   f. Push the Diesel Normal Start pushbutton (AI-30A-D1). ✓

   g. Verify DG-1 accelerates to Idle Speed (500 rpm). ✓

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

3. h. Verify the following dampers are open:

- YCV-871G, Inlet ✓
- YCV-871H, Inlet ✓
- YCV-871E, Exhaust ✓

i. Notify the Local Operator to perform the following:

- Visual inspection of DG-1 ✓
- Verify oil visible in the Upper Sightglass ✓

j. Record the Start Time in the Control Room Log and FC-1046, Diesel Generator Demand Record. ✓ X

4. IF Loading DG-1,
THEN perform the following:

a. **(LOCAL)** Verify Jacket Water Inlet Temperature is greater than 120°F. (TI-6024) ✓

b. Place CS-65/D1, Diesel Generator D1 Governor, to Raise until the Diesel Speed is 900 rpm. ___

NOTE

Should it be determined that the field flash circuitry is continually energized, the Diesel Generator may have problems which would prevent the field flash circuitry from performing its function on the next diesel run. The Diesel Generator should be shut down and a thorough examination of the field flash circuitry be performed. Proceed as directed by the Shift Manager to exit procedure.

c. Verify the Generator Field flashed by performing one of the following:

- Ready to Load light is ON (AI-30A) ___

OR

- Generator frequency is responding ___

4.

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

d. **(LOCAL)** Inspect field flash circuitry by performing the following (AR12033)

   1) Verify that Control Relay 2CR in Panel AI-133A is not energized. ______ EM

   2) Verify that Field Flash Current Limiting Resistors (1R4, 1R5, 1R6, and 1R7) in Panel AI-133A are not damaged due to overheating. Contact System Engineer if damage is suspected. ______ EM

e. **(LOCAL)** Place AI-133A-S4, Diesel Generator DG-1 Electronic Droop Control Switch, in ENABLED. (AI-133A) ___

f. Place D1/BUS 1A3 Sync Switch to ON. ___

g. Adjust CS-90/D1, Diesel Generator D1 Voltage Regulator, until the RUNNING VOLTS is approximately matched to the INCOMING VOLTS on the Synchroscope or the ERF DGD Display. ___

NOTE
Recommended synchroscope speed is less than 1 revolution per 10 seconds.

h. Adjust CS-65/D1 until the Synchroscope is rotating slowly in the **FAST** direction. ___

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

NOTE

Steps 4.i and 4.j may be performed without the procedure in hand. Sign-offs may be completed after these steps are performed.

CAUTIONS

1. Load must be immediately picked up following closure of 1AD1 to prevent motorizing the Diesel Generator.
2. Governor controls are extremely sensitive.

4.

i. WHEN the Synchroscope is between 11 and 12 **O'CLOCK**, THEN close 1AD1 BREAKER. ___

j. Place CS-65/D1 to Raise to pick up 250-350 KW.

Time ______________ ___

k. Place D1/BUS 1A3 Sync Switch to OFF. ___

l. IF the Diesel is loaded AND Y3287A, ERF 1A3 Bus Voltage, is greater than 4375 VAC,
THEN immediately notify the System Engineer. ___

4.

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

NOTES

1. Load should be maintained below the 2000 hr Rating vs Ambient Temp curve per TDB-III.26A Figure 1, DG-1 Output Power Rate.
2. Power factor may be determined by using TDB-III.26, Diesel Generator Capability Curve.
3. Current is normally limited to 400 amps at 2500 KW.
4. Diesel Generator manual loading and unloading rates should be maintained at less than 500 KW per minute.
5. Steps 4.m and 4.n may repeated as necessary while the diesel is loaded. Sign-offs may be completed after these steps are performed.

m. Place CS-65/D1 to RAISE picking up the required DG-1 Load. ___

n. IF DG-1 voltage OR amperage need adjustment, THEN adjust CS-90/D1. ___

o. Record the Loading Time in the Control Room Log and FC-1046. ___ ________

Completed by ______________________________ Date/Time ______/____

Facility: FCS JPM # NRC S-8 Task # 0778 K/A # 012.A4.02 3.3 / 3.4 SF-7

Title: Adjust Reactor Protection System T_{COLD} Calibration

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	____	Classroom:	____
Actual Performance:	X	Simulator:	X
Alternate Path:	____	Plant:	____
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- Plant is operating at 100% power.
- Channel D T_{COLD} calibration is indicating high.

Initiating Cue: The Shift Manager directs you to PERFORM the following:

- ADJUST Reactor Protection System (RPS) T_{COLD} calibration on Channel D per OI-RPS-2, Reactor Protective System-TM/LP $T_{COLD\ CAL}$ Calibration.

Task Standard: Utilizing OI-RPS-2, bypassed Channel D TM/LP Trip Unit, adjusted $T_{COLD\ CAL}$ Calibration, then returned Channel D TM/LP Trip Unit to service.

Required Materials: OI-RPS-2, Reactor Protective System-TM/LP $T_{COLD\ CAL}$ Calibration, Rev. 10.
TM/LP Trip Unit # 9 Bypass Key

Validation Time: 19 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

SIMULATOR SETUP

BOOTH OPERATOR:
INITIALIZE to IC-111 or any at power Initial Condition:

- **ENSURE T_{COLD} Calibrate Potentiometer on Channel D is set to greater than 5.20.**

BOOTH OPERATOR NOTE:

- **After each JPM, VERIFY Channel D TM/LP Trip Unit # 9 Bypass Key is removed from AI-31D prior to performance by the next examinee.**

EXAMINER:
PROVIDE the examinee with a copy of:

- **OI-RPS-2, Reactor Protective System-TM/LP $T_{COLD\ CAL}$ Calibration.**
  - **INITIALED through Prerequisites.**
- **PROVIDE the TM/LP Trip Unit # 9 Bypass Key.**

EXAMINER NOTE: Only SROs can check out keys from the Key Locker at FCS.

√ - **Check Mark Denotes Critical Step**

START TIME:	

Examiner Note:	**The following steps are from OI-RPS-2.**
Perform Step: 1 1.a	Record T_{cold} DVM readings on all four RPS channels. • AI-31A ____°F • AI-31B ____°F • AI-31C ____°F • AI-31D ____°F
Standard:	SELECTED Channel A/B/C/D T_{COLD} on Digital Voltmeters (DVM) and RECORDED temperatures at Step 1.a.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2 1.b	Record $T_{cold\ cal}$ DVM readings on all four RPS channels. • AI-31A ____°F • AI-31B ____°F • AI-31C ____°F • AI-31D ____°F
Standard:	SELECTED Channel A/B/C/D $T_{COLD\ CAL}$ on Digital Voltmeters (DVM) and RECORDED temperatures at Step 1.b.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3 1.c	Record $T_{cold\ cal}$ POT settings. • AI-31A _____ • AI-31B _____ • AI-31C _____ • AI-31D _____
Standard:	RECORDED Channel A/B/C/D $T_{COLD\ CAL}$ POT settings at Step 1.c.
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**PROVIDE RPS TM/LP Trip Unit # 9 Bypass Key.**
Perform Step: 4 1.d	Obtain the RPS TM/LP Trip Unit # 9 Bypass Key.
Standard:	OBTAINED RPS TM/LP Trip Unit # 9 Bypass Key from Key Locker.
Comment:	**SAT** ☐ **UNSAT** ☐

CAUTION
Only **ONE** channel shall be adjusted at a time.

Perform Step: 5 1.e	Log into Technical Specification 2.15.1(1) 48 hour LCO for selected channel: • AI-31D
Standard:	INFORMED CRS of entry into Technical Specification LCO 2.15.1 for Reactor Protection System Channel D.
Examiner Cue:	**CRS logs entry into Technical Specification LCO.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 1.f	Bypass TM/LP trip unit # 9 on the selected channel using Bypass Key: • AI-31D
Standard:	PERFORMED the following: • INSERTED key into Channel D TM/LP Trip Unit # 9 and TURNED to BYPASS Channel D (**critical**). • OBSERVED Channel D Trip Unit amber light lit (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7√ 1.g	Adjust $T_{cold\ cal}$ POT on the selected channel until the $T_{cold\ cal}$ DVM reading equals highest RPS channel T_{cold} recorded in Step a. • AI-31D
Standard:	ADJUSTED $T_{COLD\ CAL}$ POT on Channel D until $T_{COLD\ CAL}$ DVM reading equals highest RPS channel T_{COLD} recorded in Step a.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 8 1.h	Ensure selected TM/LP Trip Unit #9 is RESET by depressing T/U Alarm Reset. • AI-31D
Standard:	PERFORMED the following: • DEPRESSED T/U Alarm Reset on Channel D TM/LP Trip Unit #9. • OBSERVED Channel D Trip Unit #9 alarm RESET light off.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9 1.i	Verify TM/LP Trip Unit #9 Lights are reset.
Standard:	VERIFIED Channel D Trip Unit #9 alarm RESET light off.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 10√ 1.j	Remove Bypass Key for selected TM/LP Trip Unit. • AI-31D
Standard:	PERFORMED the following: • REMOVED Channel D Trip Unit #9 Bypass Key (**critical**). • OBSERVED Channel D Trip Unit amber light off (**NOT critical**).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 11 1.k	Exit Technical Specification 2.15.1(1) for the selected channel. ◦ AI-31D
Standard:	INFORMED CRS of exit from Technical Specification LCO 2.15.1 for Reactor Protection System Channel D.
Examiner Cue:	**CRS logs exit from Technical Specification LCO.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 12 1.l	Repeat Steps "e" through "k" for any remaining channels out of specification.
Standard:	DETERMINED there are NO remaining Channels out of specification.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 13 1.m	Record $T_{cold\ cal}$ DVM readings. • AI-31A ____°F • AI-31B ____°F • AI-31C ____°F • AI-31D ____°F
Standard:	SELECTED Channel A/B/C/D $T_{COLD\ CAL}$ on Digital Voltmeters (DVM) and RECORDED temperatures at Step 1.m.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 14 1.n	Record $T_{cold\ cal}$ POT settings. • AI-31A _____ • AI-31B _____ • AI-31C _____ • AI-31D _____
Standard:	RECORDED Channel A/B/C/D $T_{COLD\ CAL}$ POT settings at Step 1.n.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **Plant is operating at 100% power.**
- **Channel D T_{COLD} calibration is indicating high.**

INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **ADJUST Reactor Protection System (RPS) T_{COLD} calibration on Channel D per OI-RPS-2, Reactor Protective System-TM/LP $T_{COLD\ CAL}$ Calibration.**

Fort Calhoun Station
Unit No. 1

OI-RPS-2

OPERATING INSTRUCTION

REACTOR PROTECTIVE SYSTEM – TM/LP $T_{COLD\ CAL}$ CALIBRATION

Change No.	EC 59995
Reason for Change	Correct Technical Specification reference as a result of Amendment 270.
Requestor	S. Lindquist
Preparer	K. Bessey
Issue Date	08-27-13 3:00 pm

REACTOR PROTECTIVE SYSTEM - TM/LP $T_{COLD\ CAL}$ CALIBRATION

SAFETY RELATED

PRECAUTIONS

1. The selected RPS TM/LP Trip Unit is placed in BYPASS prior to adjusting the $T_{cold\ cal}$ Pot to prevent an inadvertent channel trip.

2. T_{cold} DVM readings while at a steady state power are normally kept within close tolerances. Hence, if while at steady state power readings differ by greater than 1.0°F, the System Engineer should be consulted prior to adjustment to ensure operability of the affected channel.

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specifications:

   - 1.1.1: Reactor Core SL_S
   - 2.13: Table 2-11: RPS Limiting Safety System Settings
   - 2.15: Instrumentation and Control Systems
   - 2.15: Table 2-2: Instrument Operating Requirements for Reactor Protective System
   - 3.1: Table 3-3: Item 17: Reactor Coolant Inlet Temperature
   - 3.1: Table 3-1: Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls
   - 3.10(7)a: DNB Parameters

2. USAR:

   - 7.2: Reactor Protective System
   - 7.5: Instrumentation Systems

3. Others:

   - OP-ST-SHIFT-0001
   - SO-G-100: Operability Dispositions When Calibrating, Testing or Operating Safety Related Equipment
   - EAR 94-130, Appropriate Acceptance Criteria or Requirements to T_{COLD} Cal

Attachment 1 - Adjust the $T_{cold\ cal}$ POT(s)

PREREQUISITES (✓) INITIAL

1. Procedure Revision Verification

   Revision Number 10 Date: today — 8

2. Reactor is at steady state conditions. — 8

3. Reactor power is greater than 15%. — 8

4. Adjustment is requested by the Shift Manager OR the difference between the highest T_{cold} DVM reading and any $T_{cold\ cal}$ DVM reading for the applicable Reactor power level is as follows:

   - greater than 75% to 100% power greater than or equal to 0.2°F
   - greater than 50% to less than or equal to 75% power greater than or equal to 0.5°F
   - greater than or equal to 15% to less than or equal to 50% power greater than or equal to 1.0°F — 8

5. IF T_{cold} DVM readings differ by more than 1.0°F, THEN contact the System Engineer prior to adjustment. — 8

6. Shift Manager notified prior to adjustment. — SM
   Shift Mgr

PROCEDURE

1. Perform the following:

   a. Record T_{cold} DVM readings on all four RPS channels.

      - AI-31A ______°F ___
      - AI-31B ______°F ___
      - AI-31C ______°F ___
      - AI-31D ______°F ___ ______

Attachment 1 - Adjust the $T_{cold\ cal}$ POT(s)

PROCEDURE (continued) (✓) INITIAL

1.

b. Record $T_{cold\ cal}$ DVM readings on all four RPS channels.

- AI-31A ______°F ___
- AI-31B ______°F ___
- AI-31C ______°F ___
- AI-31D ______°F ___ ________

c. Record $T_{cold\ cal}$ POT settings.

- AI-31A ______ ___
- AI-31B ______ ___
- AI-31C ______ ___
- AI-31D ______ ___ ________

d. Obtain the RPS TM/LP Trip Unit # 9 Bypass Key. ________

CAUTION
Only **ONE** channel shall be adjusted at a time.

e. Log into Technical Specification 2.15.1(1) 48 hour LCO for selected channel:

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

f. Bypass TM/LP trip unit # 9 on the selected channel using Bypass Key:

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

Attachment 1 - Adjust the $T_{cold\ cal}$ POT(s)

PROCEDURE (continued) (✓) INITIAL

1.

g. Adjust $T_{cold\ cal}$ POT on the selected channel until the $T_{cold\ cal}$ DVM reading equals highest RPS channel T_{cold} recorded in Step a.

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

h. Ensure selected TM/LP Trip Unit #9 is RESET by depressing T/U Alarm Reset.

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

i. Verify TM/LP Trip Unit #9 Lights are reset. ________

j. Remove Bypass Key for selected TM/LP Trip Unit.

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

k. Exit Technical Specification 2.15.1(1) for the selected channel

- AI-31A ___
- AI-31B ___
- AI-31C ___
- AI-31D ___ ________

l. Repeat Steps e. through k. for any remaining channels out of specification. ________

Attachment 1 - Adjust the $T_{cold\ cal}$ POT(s)

PROCEDURE (continued) (✓) INITIAL

1.

m. Record $T_{cold\ cal}$ DVM readings:

- AI-31A ______°F ___
- AI-31B ______°F ___
- AI-31C ______°F ___
- AI-31D ______°F ___ ______

n. Record new $T_{cold\ cal}$ POT settings:

- AI-31A ______ ___
- AI-31B ______ ___
- AI-31C ______ ___
- AI-31D ______ ___ ______

Completed by________________________________Date/Time__________/________

Facility: FCS JPM # NRC P-1 Task # 1398 K/A # 033.A2.02 2.7 / 3.0 SF-8

Title: Spent Fuel Pool Cooling Restoration with SIAS

Examinee (Print): ______________________

Testing Method:

Simulated Performance:	X	Classroom:	
Actual Performance:		Simulator:	
Alternate Path:		Plant:	X
Time Critical:			

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- AOP-36, Loss of Spent Fuel Pool Cooling, has been entered following a Safety Injection Actuation Signal.
- SIAS and CIAS have been RESET.
- 480 Volt Bus 1B4C is energized and available for Spent Fuel Pool Cooling Pump AC-5B.
- HC-478, STORAGE POOL HX AC-8 AC OUTL HCV-478 at AI-45 is OPEN.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- RESTORE Spent Fuel Pool Cooling using Spent Fuel Pool Cooling Pump AC-5B per AOP-36, Loss of Spent Fuel Pool Cooling, Attachment H, Spent Fuel Pool Cooling Restoration with SIAS.
- START at Step 6.

Task Standard: Utilizing AOP-36, Attachment H, aligned valves, started Spent Fuel Pool Cooling Pump AC-5B, and restored SFP cooling flow.

Required Materials: AOP-36, Loss of Spent Fuel Pool Cooling, Rev. 11.

Validation Time: 15 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________ Date: ____________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **AOP-36, Loss of Spent Fuel Pool Cooling.**
    - **Attachment H, Spent Fuel Pool Cooling Restoration with SAIS.**
    - **INITIALED/PLACE KEEPING through Step 5.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from AOP-36, Attachment H, Step 6.**
Examiner Note:	**MCC 4C2 is located in Corridor 4, 989' elev. of Auxiliary Building.**
Examiner Note:	**MCC-4C2-F05 is a "bucket style" breaker. Turn to the left to RESET, then turn to right to ON.**
Perform Step: 1 6	Verify breaker MCC-4C2-F05, "AC-5B FUEL STORAGE POOL CIRC PUMP" is reset and closed (Corridor 4)
Standard:	PERFORMED the following: • OBSERVED breaker in the ON position with Green light LIT • Candidate *may* (not required) reset breaker by: TURN handle on MCC-4C2-F05, AC-5B FUEL STORAGE POOL CIRC PUMP to LEFT to RESET breaker, then TURN handle on MCC-4C2-F05, AC-5B FUEL STORAGE POOL CIRC PUMP to RIGHT to ON.
Examiner Cue:	**Breaker handle pointing to "ON." Green light is lit.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Room 5 is located in Corridor 4, 989' elev. of Auxiliary Building.**
Examiner Note:	**Radiation levels are somewhat elevated in Room 5. By selecting SFP AC-5B, the Applicant can remain in a "LOW DOSE WAITING AREA" while describing manipulations to be performed.**
Perform Step: 2 7	Contact Shift RP for Room 5 entry.
Standard:	CONTACTED Shift Radiation Protection for Room 5 entry.
Examiner Cue:	**Shift Radiation Protection has been contacted and approves entry.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**AC-191 is a 90-degree butterfly valve. The operating handle has "notches" to keep the valve in position. The specific throttle position in this step is not critical, but throttling the valve is critical to prevent pump runout on start. Candidate may determine appropriate throttle position, or the Examiner may provide a cue.**
Examiner Cue:	**If requested: The CRS directs you to place AC-191 at approximately 50% open.**
Perform Step: 3 √ 8	Throttle <u>open</u> the selected SFP pump discharge valve (Room 5): • AC-191, "SPENT FUEL POOL CIRC PUMP AC-5B DISCHARGE VALVE"
Standard:	PERFORMED the following: • SQUEEZED handle on AC-191, SPENT FUEL POOL CIRC PUMP AC-5B DISCHARGE VALVE and PLACED in a throttled position less than full open. • OBSERVED valve handle at 45° from piping.
Examiner Cue:	**Valve handle is 45° offset from piping (or throttled appropriately).**
Comment:	**SAT ☐ UNSAT ☐**

CAUTION

If the Discharge Valve for the non-running Spent Fuel Pool Circ Pump is Open, the pump will windmill backwards. The discharge valve must be Closed while starting the pump to prevent excessive starting current.

Perform Step: 4√ 9	<u>Close</u> the non-selected SFP pump discharge valve: • AC-192, AC-5A
Standard:	PERFORMED the following: • SQUEEZED handle on AC-192, SPENT FUEL POOL CIRC PUMP AC-5A DISCHARGE VALVE and PLACED in CLOSED position then RELEASED (**critical**). • OBSERVED valve handle perpendicular to piping (**NOT critical**).
Examiner Cue:	**Valve handle is perpendicular to piping.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**START pushbutton is located on wall behind pump.**
Perform Step: 5√ 10	<u>Start</u> the selected SFP pump (Room 5): • AC-5B, "SPENT FUEL POOL COOLING PUMP"
Standard:	PERFORMED the following: • DEPRESSED black START pushbutton for AC-5B, SPENT FUEL POOL COOLING PUMP (**critical**). • OBSERVED red START light lit and GREEN stop light off (**NOT critical**).
Examiner Cue:	**RED light is lit and GREEN light is off. Noise emanating from pump.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 11	<u>Open</u> the selected SFP pump discharge valve: • AC-191
Standard:	PERFORMED the following: • SQUEEZED handle on AC-191, SPENT FUEL POOL CIRC PUMP AC-5B DISCHARGE VALVE and PLACED in OPEN position then RELEASED (**critical**). • OBSERVED valve handle parallel with piping (**NOT critical**).
Examiner Cue:	**Valve handwheel is parallel with piping.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Pressure gauge ranges from 0 to 300 psig.**
Perform Step: 7 12	<u>Verify</u> SFP pump discharge pressure 40-60 psig.
Standard:	OBSERVED pressure gauge on discharge of AC-5B between 40 psig and 60 psig.
Terminating Cue:	**Pressure gauge needle positioned 1/5th upscale. Another operator will throttle CCW flow to minimize dose rate. This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **AOP-36, Loss of Spent Fuel Pool Cooling, has been entered following a Safety Injection Actuation Signal.**
- **SIAS and CIAS have been RESET.**
- **480 Volt Bus 1B4C is energized and available for Spent Fuel Pool Cooling Pump AC-5B.**
- **HC-478, STORAGE POOL HX AC-8 AC OUTL HCV-478 at AI-45 is OPEN.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **RESTORE Spent Fuel Pool Cooling using Spent Fuel Pool Cooling Pump AC-5B per AOP-36, Loss of Spent Fuel Pool Cooling, Attachment H, Spent Fuel Pool Cooling Restoration with SIAS.**
- **START at Step 6.**

Attachment H

Spent Fuel Pool Cooling Restoration with SIAS

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTES

The following steps may override Engineered Safeguard signals.

****

CAUTION

The restoration of the Spent Fuel Pool Cooling system will place more heat load on the CCW system. Containment and other heat loads will be affected.

****

1. Reset the following PER AOP-23, Reset of Engineered Safeguards:

   - SIAS
   - CIAS

1.1 **IF** safeguards can **NOT** be reset **AND** is required for restoration, **THEN** perform the following:

   a. Contact the TSC.

   b. IMPLEMENT EM-RR-SFP-0100, Spent Fuel Pool Cooling Restoration.

Attachment H

Spent Fuel Pool Cooling Restoration with SIAS

INSTRUCTIONS	CONTINGENCY ACTIONS
2. Ensure HC-478, "STORAGE POOL HX AC-8 AC OUTL HCV-478" is open (AI-45).	2.1 **IF** HCV-478 is **NOT** open, **THEN** locally open by performing the following: a. Contact Shift RP for entrance to Room 5. b. Close IA-HCV-478-B, "IA VALVE TO HCV-478" (Room 5). c. Open IA-HCV-478-V, "HCV-478 INSTRUMENT AIR VENT VALVE" (Room 5).
3. Ensure at least one Spent Fuel Pool Cooling Pump, AC-5A/B, is available.	3.1 **IF NO** Spent Fuel Cooling Pumps are available, **THEN** GO TO Step 12 in the body of the procedure.

Attachment H

Spent Fuel Pool Cooling Restoration with SIAS

INSTRUCTIONS

CONTINGENCY ACTIONS

4. Ensure power is available to the selected pump:

- 1B3A, AC-5A
- 1B4C, AC-5B

PER AOP-32, Loss of 4160 Volt or 480 Volt Bus Power.

N/A

5. Verify breaker MCC-3A2-F05, "AC-5A FUEL STORAGE POOL CIRC PUMP" is reset and closed (Corridor 4).

6. Verify breaker MCC-4C2-F05, "AC-5B FUEL STORAGE POOL CIRC PUMP" is reset and closed (Corridor 4).

7. Contact Shift RP for Room 5 entry.

8. Throttle open the selected SFP pump discharge valve (Room 5):

- AC-192, "SPENT FUEL POOL CIRC PUMP AC-5A DISCHARGE VALVE"
- AC-191, "SPENT FUEL POOL CIRC PUMP AC-5B DISCHARGE VALVE"

Attachment H

Spent Fuel Pool Cooling Restoration with SIAS

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

If the Discharge Valve for the non-running Spent Fuel Pool Circ Pump is Open, the pump will windmill backwards. The discharge valve must be Closed while starting the pump to prevent excessive starting current.

****

9. Close the non-selected SFP pump discharge valve:

   - AC-192, AC-5A
   - AC-191, AC-5B

10. Start the selected SFP pump (Room 5):

    - AC-5A, "SPENT FUEL POOL COOLING PUMP"
    - AC-5B, "SPENT FUEL POOL COOLING PUMP"

11. Open the selected SFP pump discharge valve:

    - AC-192
    - AC-191

Attachment H

Spent Fuel Pool Cooling Restoration with SIAS

INSTRUCTIONS	CONTINGENCY ACTIONS
12. Verify SFP pump discharge pressure 40-60 psig.	12.1 Throttle AC-194, "SPENT FUEL POOL HT EXCH AC-8 INLET VALVE" to maintain SFP pump discharge pressure 40-60 psig.

****

CAUTION

Component Cooling Water temperature needs to be maintained less than 120°F to ensure all other required loads are not adversely affected.

****

INSTRUCTIONS	CONTINGENCY ACTIONS
13. Throttle AC-117,"SPENT FUEL POOL HT EXCH AC-8 CCW INLET VALVE" as necessary to maintain **BOTH** of the following (Room 5): • CCW temperature less than 120°F (TE-2800) • SFP temperature less than 120°F (TI-2845)	13.1 **IF** CCW temperature can **NOT** be maintained less than 120°F, **THEN** stop **BOTH** SFP Pumps (Room 5): • AC-5A • AC-5B

End of Attachment H

Facility: FCS   JPM # NRC P-2   Task # 0809   K/A #061.A2.05   3.1 / 3.4   SF-4S

Title: Locally Start FW-54, Diesel Driven AFW Pump

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	X	Classroom:	
Actual Performance:		Simulator:	
Alternate Path:	X	Plant:	X
Time Critical:			

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- EOP-06, Loss of All Feedwater, is in progress.
- FW-54, Diesel Driven Auxiliary Feedwater Pump, failed to start from the Control Room.
- Feedwater flow CANNOT be aligned to the Feed Ring.
- Control Room has provided AI-114, FW-54 Control Panel keys.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- Locally START FW-54, Diesel Driven AFW Pump per EOP/AOP Attachments-HR Heat Removal, HR-16, FW-54 Operation.

Task Standard: Utilizing HR-16, started FW-54 then deenergized and opened HCV-1384.

Required Materials: EOP/AOP Attachments-HR Heat Removal, Rev. 1.

Validation Time: 15 minutes   Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **EOP/AOP Attachments-HR Heat Removal.**
    - **HR-16, FW-54 Operation.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from EOP/AOP Attachments, HR-16.**
Examiner Note:	**FW-54 Room is located in Southeast corner of Turbine Building basement.**
Perform Step: 1 1, 1.a, & 1.b	Start FW-54, Diesel AFW Pump, by performing step a or b. • Start FW-54 from the Control Room by placing HC/FW-54, "AFW PUMP FW-54" in "START". • **(LOCAL)** Start FW-54 by performing the following:
Standard:	DETERMINED from Initial Conditions that Local Start of FW-54 is required.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 b.1)	Obtain AI-114, FW-54 Control Panel keys, from one of the following: • EONT key ring • Control Room
Standard:	DETERMINED from Initial Conditions that Control Room has provided AI-114, FW-54 Control Panel keys.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 b.2)	Using key, place HC/FW-54-1, "LOCAL CONTROL SWITCH" in "STOP".
Standard:	INSERTED key into 2 position switch and TURNED HC/FW-54-1, FW-54 LOCAL CONTROL SWITCH to STOP (left) position.
Examiner Cue:	**Key is in STOP position.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ b.3)	Using key, place 43/FW-54, "CONTROL TRANSFER SWITCH" in "RESET".
Standard:	INSERTED key into 3 position switch and TURNED 43/FW-54, FW-54 CONTROL TRANSFER SWITCH to RESET (center) position.
Examiner Cue:	**Key is in RESET position.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ b.4)	Using key, place "43/FW-54" in "LOCAL".
Standard:	INSERTED key into 3 position switch and TURNED 43/FW-54, FW-54 CONTROL TRANSFER SWITCH to LOCAL (right) position.
Examiner Cue:	**Key is in LOCAL position.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ b.5)	Using key, place "HC/FW-54-1" in "RUN".
Standard:	PERFORMED the following: • INSERTED key into 2 position switch and TURNED HC/FW-54-1, FW-54 LOCAL CONTROL SWITCH to RUN (right) position. (**critical**). • OBSERVED engine speed rising (**NOT critical**). • OBSERVED engine noise rising (**NOT critical**). • OBSERVED Diesel Room area for leaks (**NOT critical**).
Examiner Cue:	**Key is in RUN position. Engine noise and speed are rising.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7 2 & 2.1 CA	Feed through the Feed Ring by performing the following: • **IF** the Feed Ring is **NOT** available, **THEN** GO TO Step 0.
Standard:	DETERMINED Feed Ring is NOT available per Initial Conditions and TRANSITIONED to Step 4.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 8 4 & 4.a	Feed through the AFW Nozzles by performing the following: • Open HCV-1384, FW/AFW Header Cross-Connect Valve.
Standard:	CONTACTED Control Room to open HCV-1384, FW/AFW Header Cross-Connect Valve.
Examiner Cue:	**Control Room reports HCV-1384, FW/AFW Header Cross-Connect Valve will NOT open.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**West Upper Electrical Penetration Room is located in the Auxiliary Building directly below Control Room.**
Perform Step: 9√ 4.a.1 CA & 4.a.1.1) CA	**(LOCAL) IF** valve will **NOT** open, **THEN** perform the following: • Place Breaker MCC-4C1-E03, "FW AND AUX FEED WATER CROSS CONNECTION VALVE" in "OFF" (West Upper Electrical Penetration Room).
Standard:	ROTATED switch for MCC-4C1-E03, HCV-1384, FW AND AUX FEEDWATER CROSSCONNECTION VALVE breaker COUNTERCLOCKWISE to OFF position.
Examiner Cue:	**Breaker switch rotated then stopped.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Room 81 is located on 1036' elev. of the Auxiliary Building and is accessed from the Turbine Deck.**
Perform Step: 10√ 4.a.1.2) CA	Manually open HCV-1384 (Room 81).
Standard:	DEPRESSED clutch arm and ROTATED handwheel for HCV-1384, FW-AFW MAIN AND AUXILIARY FEEDWATER CROSSCONNECT VALVE in COUNTERCLOCKWISE direction until stopped.
Examiner Cue:	**Valve handwheel rotated then stopped.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 11 4.b & 1st bullet	Close **BOTH** of the Feed Header Isolation Valves: • HCV-1386 • HCV-1385
Standard:	CONTACTED Control Room to close the Feed Header Isolation Valves.
Terminating Cue:	**Control Room reports Feed Header Isolation Valves are closed. This JPM is complete.**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **EOP-06, Loss of All Feedwater, is in progress.**
- **FW-54, Diesel Driven Auxiliary Feedwater Pump, failed to start from the Control Room. In and in and in and in and**
- **Feedwater flow CANNOT be aligned to the Feed Ring.**
- **Control Room has provided AI-114, FW-54 Control Panel keys.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **Locally START FW-54, Diesel Driven AFW Pump per EOP/AOP Attachments-HR Heat Removal, HR-16, FW-54 Operation.**

Attachment HR-16

FW-54 Operation

INSTRUCTIONS

CONTINGENCY ACTIONS

1. Start FW-54, Diesel AFW Pump, by performing step a or b.

   a. Start FW-54 from the Control Room by placing HC/FW-54, "AFW PUMP FW-54" in "START".

   b. **(LOCAL)** Start FW-54 by performing the following:

      1) Obtain AI-114, FW-54 Control Panel keys, from one of the following:

         - EONT key ring
         - Control Room

      2) Using key, place HC/FW-54-1, "LOCAL CONTROL SWITCH" in "STOP".

      3) Using key, place 43/FW-54, "CONTROL TRANSFER SWITCH" in "RESET".

(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
1.b. (continued)	
4) Using key, place "43/FW-54" in "LOCAL".	
5) Using key, place "HC/FW-54-1" in "RUN".	
2. Feed through the Feed Ring by performing the following:	2.1 **IF** the Feed Ring is **NOT** available, **THEN** GO TO Step 4.
a. Verify DCS is controlling feedwater in automatic.	a.1 **IF** DCS is **NOT** controlling in automatic, **THEN** manually control feed flow via **BOTH** of the Feed Reg Bypass Valves PER Attachment HR-11, Manual Feed Control (DCS).
(continue)	(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
2. (continued)	2.1 (continued)
(continue)	a.2 **IF** the normal flowpath is **NOT** available, **THEN** perform the following: 1) Ensure HCV-1384, FW/AFW Header Cross-Connect Valve, is closed. 2) Ensure **BOTH** Feed Reg Block Valves are closed: • HCV-1103 • HCV-1104 3) **IF** SGIS has occurred, **THEN** IMPLEMENT Attachment HR-14, SGIS Override. (continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
2. (continued)	2.1a.2 (continued)
	4) Ensure **BOTH** Feed Header Isolation Valves are open: • HCV-1386 • HCV-1385
	5) Manually control feed flow to the least affected S/G via the selected Feed Reg Bypass Valve(s) • HCV-1105 • HCV-1106 PER Attachment HR-11, Manual Feed Control (DCS).
b. Record time feedwater restored. Time: ____________	

Attachment HR-16

FW-54 Operation

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

If Instrument Air to valves HCV-1105, HCV-1106, is not available, throttling of these valves is not possible from the Control Room. Open or close operation of these valves is possible for a minimum of three cycles.

3. **IF BOTH** of the following conditions are satisfied:

   - The Feed Ring is available as a flow path
   - Instrument Air is **NOT** available

   THEN open the valve on the least affected S/G by performing the following:

   a. Ensure the selected Feed Header Isolation Valve(s) are open:

      - HCV-1386
      - HCV-1385

(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
3. (continued)	
b. Place the selected SGIS Override Switch(es) in "OPEN": RC-2A • HC-1105 RC-2B • HC-1106	
c. Record time feedwater restored. Time: ________	
4. Feed through the AFW Nozzles by performing the following:	
a. Open HCV-1384, FW/AFW Header Cross-Connect Valve. (continue)	a.1 **(LOCAL) IF** valve will **NOT** open, **THEN** perform the following: 1) Place Breaker MCC-4C1-E03, "FW AND AUX FEED WATER CROSS CONNECTION VALVE" in "OFF" (West Upper Electrical Penetration Room). (continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
4. (continued)	a.1 (continued)
	2) Manually open HCV-1384 (Room 81).
b. Close **BOTH** of the Feed Header Isolation Valves: • HCV-1386 • HCV-1385	b.1 **IF** power is not available to the valves, **THEN** restore power PER Attachment MVA-11, 480 Volt Bus Transfer. • HCV-1386, MCC-4C1 • HCV-1385, MCC-3A1
(continue)	(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS
4. (continued)	
	b.2 **(LOCAL) IF** valve(s) will **NOT** close electrically, **THEN** perform the following:
	1) Place the following breakers in "OFF"
	• MCC-4C1-E04, "HCV-1386 S/G RC-2A FEEDWATER ISOLATION VALVE" (West Upper Electrical Penetration Room).
(continue)	• MCC-3A1-E04, "HCV-1385 S/G RC-2B ISOLATION VALVE" (East Upper Electrical Penetration Room)
	(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS

CONTINGENCY ACTIONS

4. (continued)

b.2 (continued)

2) Manually close selected Valve(s) (Room 81):

- HCV-1386, "STEAM GENERATOR RC-2A INLET ISOLATION VALVE"
- HCV-1385, "STEAM GENERATOR RC-2B INLET ISOLATION VALVE"

****

CAUTION

Placing the AFW Isolation Valve control switches in any position but "AUTO" will override AFAS.

****

c. Open the selected AFW Isolation Valve(s) (AI-66A/B):

- HCV-1107A
- HCV-1108A

(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS	CONTINGENCY ACTIONS

4. (continued)

d. Verify **BOTH** AFW Isolation Valves are in "AUTO" (AI-66A/B).

- HCV-1107B
- HCV-1108B

NOTE

Open and close operation of HCV-1107B and HCV-1108B is possible for a minimum of three cycles if Instrument Air is lost.

e. Manually control the selected AFW Isolation Valve(s) (CB-10,11):

- HIC-1107B
- HIC-1108B

e.1 **IF** Instrument Air is lost, **THEN** manually control AFW flow by cycling selected AFW Isolation Valve(s) (AI-66A/B):

- HCV-1107B
- HCV-1108B

f. Record time feedwater established.

Time: _______

(continue)

Attachment HR-16

FW-54 Operation

INSTRUCTIONS | CONTINGENCY ACTIONS

4. (continued)

g. **(LOCAL) IF** required for level control,
THEN manually throttle **BOTH** AFW Isolation Vavles (Room 81):

- HCV-1107B, "STEAM GENERATOR RC-2A AUXILIARY FEEDWATER INLET VALVE"
- HCV-1108B, "STEAM GENERATOR RC-2B AUXILIARY FEEDWATER INLET VALVE"

End of Attachment HR-16

Facility: FCS JPM # NRC P-3 Task # 0735 K/A # 071 G2.1.30 4.4 / 4.0 SF-9

Title: Terminate Release of Radioactive Gas

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	X	Classroom:	____
Actual Performance:	____	Simulator:	____
Alternate Path:	____	Plant:	X
Time Critical:	____		

READ TO THE EXAMINEE

I will explain the Initial Conditions, which steps to simulate or discuss, and provide an Initiating Cue. When you complete the task successfully, the objective for this JPM will be satisfied.

Initial Conditions: Given the following conditions:

- A release from WG-29A, Waste Gas Decay Tank is in progress.
- RM-052, Auxiliary Building Ventilation Stack Radiation Monitor has gone into HIGH alarm.
- AOP-09, High Radioactivity is in progress.
- When HC-532, Waste Gas Release Control Switch was placed in CLOSE, FCV-532A and FCV-532C on AI-100 did NOT close.
- Independent Verification has been waived by the Shift Manager due to AOP-09 entry and high radiation in the Auxiliary Building.

Initiating Cue: The Control Room Supervisor directs you to PERFORM the following:

- TERMINATE release from WG-29A per OI-WDG-2, Waste Gas Disposal System Release, Attachment 3, Manual Waste Gas Release with FE-532 Unavailable.
- START at Step 2.17.

Task Standard: Utilizing OI-WDG-2, closed WD-158, isolated WG-29A via WD-132 terminating release from Waste Gas Decay Tank WG-29A.

Required Materials: AOP-09, High Radioactivity, Rev. 11.
OI-WDG-2, Waste Gas Disposal System Release, Rev. 30.

Validation Time: 15 minutes Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **OI-WDG-2, Waste Gas Disposal System Release.**
  - **Attachment 3, Manual Waste Gas Release with FE-532 Unavailable.**
  - **Attachment 3 is INITIALED through Step 2.16.**

√ - Check Mark Denotes Critical Step

START TIME:

Examiner Note:	**The following steps are from OI-WDG-2, Attachment 3.**
Examiner Note:	**Panel AI-100 and Room 16 are adjacent to each other and located in Corridor 4, 989' elev. of Auxiliary Building.**
Perform Step: 1 2.17	Verify the following Gas Release Control Valves closed: • FCV-532A (AI-100) • FCV-532C (AI-100) • FCV-532B (Room 16)
Standard:	NOTED the following: • DETERMINED FCV-532A and FCV-532C did NOT close from Initial Conditions, **or** • OBSERVED red OPEN lights lit and green CLOSE lights off at AI-100, **and** • OBSERVED FCV-532B open in Room 16.
Examiner Cue:	**Red lights are lit on AI-100 for FCV-532A & FCV-532C. In Room 16, FCV-532B indicates mid position (between open and close position discs).**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**All valves are located on the East wall of Room 16.**
Perform Step: 2√ 2.18	Close WD-158.
Standard:	ROTATED WD-158, WASTE GAS RELEASE HEADER FLOW ELEMENT FE-532 BYPASS LINE ISOLATION VALVE handwheel in CLOCKWISE direction until stopped.
Examiner Cue:	**Valve handwheel rotated then stopped.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Valve located about 6 feet above the floor.**
Perform Step: 3 2.19	Close the selected WGDT Outlet to Gas Release Header Valve (Rm 16): • WD-132, WD-29A
Standard:	ROTATED WD-132, GAS DECAY TANK WD-29A OUTLET VALVE handwheel in CLOCKWISE direction until stopped.
Examiner Cue:	**Valve handwheel will NOT move.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 2.20 & 1st bullet	Close and lock the following Gas Release Header Isolation Valves (Rm 16): • WD-150
Standard:	PERFORMED the following: • ROTATED WD-150, WASTE GAS DECAY TANKS WD-29A, B, C & D GAS RELEASE HEADER ISOLATION VALVE handwheel in CLOCKWISE direction until stopped (**critical**). • INSTALLED chain and LOCKED valve (**NOT critical**). • INFORMED Control Room WD-150 is LOCKED per SO-O-44 (**NOT critical**).
Examiner Cue:	**Valve handwheel rotated then stopped.** **If Control Room is contacted, ACKNOWLEDGE locking of WD-150.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Valve located > 8 feet above the floor. May require a ladder that can be obtained from the Corridor (West) just beyond AI-100, and notification to RP that they are working above 7 ft.**
Perform Step: 5 2.20 & 2nd bullet	Close and lock the following Gas Release Header Isolation Valves (Rm 16): • WD-167
Standard:	PERFORMED the following: • If needed, OBTAINED a ladder, simulated notifying RP • ROTATED WD-167, WASTE GAS DECAY TANKS WD-29A, B, C & D GAS RELEASE HEADER ISOLATION VALVE handwheel in CLOCKWISE direction until stopped.
Examiner Cue:	**Valve handwheel will NOT move.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Valve located at floor level.**
Perform Step: 6√ 2.21	Close WD-165, Gas Release Header Bypass Valve (Rm 16).
Standard:	ROTATED WD-165, GAS RELEASE HEADER BYPASS VALVE handwheel in CLOCKWISE direction until stopped.
Terminating Cue:	**Valve handwheel rotated then stopped. This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions:**

- **A release from WG-29A, Waste Gas Decay Tank is in progress.**
- **RM-052, Auxiliary Building Ventilation Stack Radiation Monitor has gone into HIGH alarm.**
- **AOP-09, High Radioactivity is in progress.**
- **When HC-532, Waste Gas Release Control Switch was placed in CLOSE, FCV-532A and FCV-532C on AI-100 did NOT close.**
- **Independent Verification has been waived by the Shift Manager due to AOP-09 entry and high radiation in the Auxiliary Building.**

INITIATING CUE: **The Control Room Supervisor directs you to PERFORM the following:**

- **TERMINATE release from WG-29A per OI-WDG-2, Waste Gas Disposal System Release, Attachment 3, Manual Waste Gas Release with FE-532 Unavailable.**
- **START at Step 2.17.**

Fort Calhoun Station
Unit No. 1

AOP-09

HIGH RADIOACTIVITY

Change No.:	EC 61295, 60326
Reason for Change:	Revised as part of Operations Procedure Upgrade in response to NRC 0350 concern.
Initiator:	M. Sandhoefner, J. McClintic, D. Pier
Preparer:	A. Peters
Issued:	09-20-13 3:00 pm

1.0 PURPOSE

This procedure provides guidance in the event of unplanned or uncontrolled high radiation levels in any area of the plant.

2.0 ENTRY CONDITIONS

Unplanned or uncontrolled high radiation levels exist in an area of the plant which may be indicated by any of the following:

A. Area Radiation Monitor alarm.

B. "RM-050 CNTMT PARTICULATE HIGH RADIATION" alarm (AI-33C; A33C).

C. "RM-051 CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

D. "RM-052 STACK/CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

E. "RM-062 AUX BLDG VENT STACK HIGH RADIATION" alarm (AI-33C; A33C).

F. "RM-043 LAB/RAD WASTE BLDG STACK HIGH RADIATION" alarm (AI-33C; A33C).

G. Containment Air Particle Monitor indicates upscale.

H. Ventilation Isolation Actuation Signal.

I. Rupture of a Gas Decay Tank or the Volume Control Tank.

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. CAUTIONS

None

B. NOTES

1. AOP-08, Fuel Handling Incident, provides guidance in the event an irradiated fuel assembly is dropped or otherwise damaged.

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS		CONTINGENCY ACTIONS	
1.	Announce and repeat the following over the plant communications system:		

"Attention all personnel. Attention all personnel. High radiation levels exist in __(location)___. All non-essential personnel should immediately evacuate the area".

2.	Direct the Radiation Protection Technician to assist in validating and locating the high activity.		
3.	Direct the Shift Chemist to terminate primary sampling PER CH-SMP-PR-0007, Reactor Coolant System Sampling.		
4.	Terminate any radioactive releases.		
5.	IMPLEMENT the Emergency Plan.		
6.	Verify PZR level is in normal program band PER OI-RC-8, Reactor Coolant System Level Control Normal Operation.	6.1	**IF** PZR level trend is abnormal, **THEN** IMPLEMENT AOP-22, Reactor Coolant Leak.

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

1. PREREQUISITES (✓) INITIALS

1.1 Procedure Revision Verification

Revision Number 30 Date today γ

1.2 Auxiliary Building Ventilation System is in operation per OI-VA-2. γ

1.3 Ensure one of the following Radiation Monitors is in operation monitoring the Ventilation Stack per OI-RM-1 (ODCM Section 3.2.1):

- RM-062
- RM-052 ✓ γ

1.4 Verify one of the four following sets of CRHS/VIAS lockout relays is reset AND amber lights are on:

1.4.1

- 86A/CRHS ✓
- 86A/VIAS ✓

1.4.2

- 86A/CRHS ✓
- CHAN "A" DERIVED SIG CUTOFF SWITCH CS-A1/SP-A INEMERGENCY STANDBY ✓
- 86A1/CRHS ✓
- 86A1/VIAS ✓

1.4.3

- 86B/CRHS ✓
- 86B/VIAS ✓

1.4.4

- 86B/CRHS ✓
- CHAN "B" DERIVED SIG CUTOFF SWITCH CS-B1/SP-B IN EMERGENCY STANDBY ✓
- 86B1/CRHS ✓
- 86B1/VIAS ✓ γ

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

1. PREREQUISITES (continued) (✓) INITIALS

NOTES

1. The radiation monitor is considered inoperable during the Check Source. Remaining stationed at the monitor AND ensuring the monitor returns to normal before leaving the area administratively replaces the log entry.
2. Steps 1.5.1 through 1.5.3 are repeated for RM-052 and RM-062 as required.

1.5 Perform Check Source on all operational Auxiliary Building Exhaust Stack Noble Gas Radiation Monitors by completing the following:

1.5.1 With the keypad switch in the ON position, record the background reading:

- RM-052 70 cpm ✓
- RM-062 50 cpm ✓

NOTE

The check source is only energized for a 2 minute period.

1.5.2 Depress the CHECK SOURCE pushbutton and verify the meter reading rises above the background reading.

- RM-052 ✓
- RM-062 ✓

1.5.3 WHEN the check source deenergizes, THEN verify the meter returns to its background reading.

- RM-052 ✓
- RM-062 ✓

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

1. PREREQUISITES (continued) (✓) INITIALS

1.6 Verify all operational Auxiliary Building Exhaust Stack Noble Gas Radiation Monitors (RM-062 and/or RM-052) Alert and High Alarm Setpoints are per TBD-IV.7. ꝩ

1.7 Attachment 4 of OI-WDG-2 has been completed within the previous 90 days. ꝩ

NOTE

The most recently completed Checklist, OI-WDG-1-CL-B, with deviations maintained on file, may be used for alignment verification.

1.8 Waste Gas Disposal System is aligned for normal operation per OI-WDG-1-CL-B. ꝩ

1.9 Verify the following recorders are operable:

- RR-049A, Process Radiation Monitor Recorder (AI-31E) ✓
- FR-758, Stack Total Flowrate Recorder (AI-44) ✓ ꝩ

1.10 IF one or more of the recorders is inoperable or a manual release through the orifice is used,
THEN take applicable readings per ODCM on FC-213, Gas Discharge Log. NA ꝩ

1.11 Record the maximum release flowrate specified in the Waste Decay Tank Release Permit, FC-213:

550 SCFH ꝩ

1.12 Record the recommended release flowrate:

495 SCFH ꝩ

1.13 Authorization has been given to perform a Waste Gas Decay Tank Release. SM
Shift Mgr

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

1. PREREQUISITES (continued) (✓) INITIALS

1.14 At least one of the following conditions is met:

- Condenser Evacuation is in service per OI-CE-1 ✓
- VA-412, Condenser Evacuation Stack Discharge Isolation Valve, is closed ___ ___

1.15 Ensure the Δp readings from VA-82 are logged on Table 1 every 2 hours for the duration of the release. ___

2. PROCEDURE

> **NOTE**
>
> The following steps are located in the Auxiliary Building or as designated.

2.1 Record the following information:

Permit No. 2015007 ___

WGDT to be released:

- WD-29A ✓
- WD-29B ___
- WD-29C ___
- WD-29D ___ ___

> **NOTE**
>
> When completing OI-WDG-2-CL-A ensure that WD-156 and WD-157 remain closed.

2.2 Complete Checklist OI-WDG-2-CL-A. ___

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

2. PROCEDURE (continued) (✓) INITIALS

2.3 Ensure the following valves are closed

- WD-156, Gas Decay Tanks WD-29A, B, C & D Gas Release Instrument Loop Root Valve ✓
- WD-157, Gas Decay Tanks WD-29A, B, C & D Gas Release Instrument Loop Root Valve ✓ γ

2.4 Verify the following:

- Flow Orifice is installed upstream of WD-165 ✓
- Diameter of the Flow Orifice matches the Release Permit requirement ✓ γ

2.5 Record the Installed Orifice Diameter:

0.70 inches γ

2.6 Unlock and open the following Gas Release Header Isol Valves (Rm 16):

- WD-150, Waste Gas Decay Tanks WD-29A, B, C & D Gas Release Header Isolation Valve ✓
- WD-167, Waste Gas Decay Tanks WD-29A, B, C & D Gas Release Header Isolation Vlv ✓ γ

2.7 Open WD-165, Gas Release Header Bypass Isolation Valve (Rm 16). γ

NOTE
Since there is no flow at this time, A50/A-4, WASTE GAS RELEASE THRU FCV-532C HI-LO Annunciator should alarm when HC-532 is placed in AUTO.

2.8 Place HC-532, Waste Gas Release Control Switch, in AUTO and verify FCV-532C opens (Red Light on) (AI-100). γ

2.9 Record the Date, Start Time and Permit No. on the following:

- RR-049A, Process Radiation Monitor Recorder (AI-31E) ✓

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

- FR-758, Stack Total Flowrate Recorder (AI-44) ✓
- Narrative Log ✓

NOTE

N/A any readings associated with FIC-532.

2.10 Record Start Data on Table 2.

2.11 Open WD-158, Waste Gas Release Header Flow Element FE-532 Bypass Line Isolation Valve.

2.12 Open the selected WGDT Outlet Valve (Rm 16).

- WD-132, Gas Decay Tank WD-29A Outlet Valve ✓
- WD-143, Gas Decay Tank WD-29B Outlet Valve
- WD-163, Gas Decay Tank WD-29C Outlet Valve
- WD-177, Gas Decay Tank WD-29D Outlet Valve

2.13 Calculate and log the flow rate every hour as specified on FC-213.

2.14 WHEN the selected WGDT has dropped to approximately 2.0 psig or as directed by the Shift Manager,
THEN place HC-532 in CLOSE (AI-100).

2.15 Record the Date, Termination Time and Permit No. on the following

- RR-049A, Process Radiation Monitor Recorder (AI-31E) ✓
- FR-758, Stack Total Flowrate Recorder (AI-44) ✓
- Narrative Log ✓

2.16 Record Stop Data on Table 2.

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

2. PROCEDURE (continued) (✓) INITIALS

2.17 Verify the following Gas Release Control Valves closed:

- FCV-532A (AI-100) ____
- FCV-532C (AI-100) ____
- FCV-532B (Rm 16) ____ ________

2.18 Close WD-158. ________

2.19 Close the selected WGDT Outlet to Gas Release Header Valve (Rm 16):

- WD-132, WD-29A ____
- WD-143, WD-29B ____
- WD-163, WD-29C ____
- WD-177, WD-29D ____ ________

2.20 Close and lock the following Gas Release Header Isolation Valves (Rm 16):

- WD-150 ____
- WD-167 ____ ________

____
Ind Verif

2.21 Close WD-165, Gas Release Header Bypass Valve (Rm 16). ________

2.22 Open the selected WGDT Drain Valve (Rm 16):

- WD-136, Gas Decay Tank WD-29A Drain Valve ____
- WD-149, Gas Decay Tank WD-29B Drain Valve ____
- WD-169, Gas Decay Tank WD-29C Drain Valve ____
- WD-180, Gas Decay Tank WD-29D Drain Valve ____ ________

2.23 Slowly open WD-181, Gas Decay Tanks WD-29A, B, C & D Drain Header Outlet (Rm 16). ________

2.24 When no water is observed flowing by FI-531, Flow Indicator (Rm 16), close WD-181. ________

Attachment 3 – Manual Waste Gas Release with FE-532 Unavailable

2. PROCEDURE (continued) (✓) INITIALS

2.25 Close the selected WGDT Drain Valve (Rm 16):

- WD-136, Gas Decay Tank WD-29A Drain Valve ____
- WD-149, Gas Decay Tank WD-29B Drain Valve ____
- WD-169, Gas Decay Tank WD-29C Drain Valve ____
- WD-180, Gas Decay Tank WD-29D Drain Valve ____ ________

2.26 Attach the completed OI-WDG-2 and this attachment to the FC-213. ________

2.27 Complete FC-213. ________

Completed by ______________________________ Date/Time __________/_____

Facility:	Fort Calhoun Station	Scenario No.:	1	Op Test No.:	Dec 2015 NRC
Examiners:	______			Operators:	______
	______				______
	______				______

Initial Conditions: 100% power MOL - RCS Boron is 482 ppm (by sample).

Turnover: Maintain steady-state power conditions. Chemistry requests two Charging Pumps be placed in service per OI-CH-1, CVCS System Normal Operation.

Critical Tasks:
- Manually Actuate Pressurizer Pressure Low Signal (PPLS) when RCS Pressure ≤ 1600 psia and before 1350 psia to ensure SIAS / VIAS / CIAS Activation. (Event 8)
- Stop All Reactor Coolant Pumps (RCPs) when Subcooling is less than 20°F due to Loss of Net Positive Suction Head (NPSH) per RCP NPSH Curve. (Event 6)
- Commence a Cooldown and Depressurization of the Reactor Coolant System to Reestablish RCS Inventory Control while maintaining RCS Heat Removal. (Event 6)

Event No.	Malf. No.	Event Type*	Event Description
1 +15 min		N (ATCO)	Raise Charging and Letdown Flow per OI-CH-1, CVCS System Normal Operation, Attachment 3.
2 +25 min		C (ATCO, CRS) TS (CRS)	Component Cooling Water (CCW) Pump Trip. Start Either Standby CCW Pump.
3 +35 min		C (BOPO, CRS)	Plant Air System Leak. Start Instrument Air Compressors.
4 +45 min		I (ATCO, CRS) TS (CRS)	Pressurizer Pressure Control Channel PT-103X Fails to 2150 psia on 15 Minute Ramp. Transfer Pressure Control to PT-103Y.
5 +55 min		R (ATCO) C (BOPO, CRS)	Condenser Evacuation Pump Trip with Auto Start Failure. Partial Loss of Condenser Vacuum. Reduce Turbine Load.
6 +55 min		M (ATCO, BOPO, CRS)	Inadvertent Main Turbine Trip. Pressurizer Safety Valve Fails 50% Open on Reactor Trip.
7 +55 min		C (BOPO)	Total Loss of Condenser Vacuum. Place HCV-1040, Atmospheric Dump Valve in Service.
8 +65 min		I (ATCO)	Pressurizer Pressure Low Signal Actuation Failure. Manually Initiate Safety Injection.
9 +65 min		C (ATCO)	Low Pressure Safety Injection (LPSI) Pumps Start Failure. Manually Start LPSI Pumps.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	**Target Quantitative Attributes**
3	Malfunctions after EOP entry (1-2)
5	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC 1

The crew will assume the shift at 100% power per OP-4, Load Change and Normal Power Operation. The scheduled activity is to start a second Charging Pump per OI-CH-1, CVCS System Normal Operation, Attachment 3, Raising Charging and Letdown Flows per Chemistry request.

The next event is a Component Cooling Water Pump Trip with auto start failure of the standby pumps. The crew enters AOP-11, Loss of Component Cooling Water, and restores flow by starting either CCW Pump AC-3A or AC-3B. The SRO will refer to Technical Specification LCO 2.4(1) – Component Cooling Water Pump.

The next event is a Plant Air System leak and entry into AOP-17, Loss of Instrument Air, is required. Crew should recognize that the Control Room Standby Instrument Air Compressor is not loading (ammeter at 0) and start a 3rd Air Compressor. Procedure exit occurs when the Plant Air System is locally isolated from the Instrument Air System.

When plant conditions are stable, Pressurizer Pressure Control Channel, PT-103X, will fail to 2150 psia over 15 minutes. Operator actions are per ARP-CB-1/2/3/A4, Windows A-4 & B-4, PRESSURIZER PRESSURE OFF NORMAL HI-LO CHANNEL X & CHANNEL Y. The crew will transfer to the standby channel PT-103Y and restore Reactor Coolant System (RCS) pressure. The SRO will refer to Technical Specification LCO 2.10.4 – DNBR Margin during Power Operation above 15% of Rated Power.

The next event is a partial Loss of Condenser Vacuum. The crew enters AOP-26, Turbine Malfunctions, Section I, Loss of Vacuum. Actions include starting a Condenser Evacuation Pump and transitioning to AOP-05, Emergency Shutdown, to lower Turbine load and restore Condenser vacuum. When power has been reduced 3% to 5%, an inadvertent Main Turbine trip will occur.

The inadvertent Main Turbine trip results in lifting of a Pressurizer Safety Valve resulting in a Small Break Loss of Coolant Accident (Vapor Space LOCA). The crew enters EOP-00, Standard Post Trip Actions, and manually actuates Safety Injection when it is determined that a Pressurizer Pressure Low Signal Actuation failure has occurred. When Diagnostic Actions are completed at the end of EOP-00, a transition will be made to EOP-03, Loss of Coolant Accident. Two Reactor Coolant Pumps are secured while in EOP-00 when pressure drops to 1350 psia. Eventually all RCPs will be secured due to a loss of subcooling (< 20°F). Upon entry into EOP-03, Containment Cooling Fans VA-7C and VA-7D will need to be started. Containment pressure remains less than 3 psig throughout the event.

The event is complicated by total Loss of Condenser Vacuum which will require placing the Atmospheric Steam Dump Valve, HCV-1040 in service and manual starting of the Low Pressure Safety Injection Pumps due to an automatic start failure.

This scenario is terminated when a cooldown and depressurization is commenced while in EOP-03 using HR-12, Secondary Heat Removal Operation, and PC-11, Pressure Control.

Risk Significance:

- Failure of risk important system prior to trip: Loss of CCW Pump
- Risk significant core damage sequence: Small Break LOCA
  Safety Injection Actuation Failure
- Risk significant operator actions: Manually Actuate Safety Injection
  Stop RCPs Upon Loss of Subcooling
  Cooldown and Depressurize RCS

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

RESET to IC- #1 (or any 100% MOL IC) and LOAD & EXECUTE NRC 1.sce for NRC Scenario 1.

Preset Item – Event 2 – Block Autostart of Non-running CCW Pumps

Type	Item	Value	Condition
Expert	CCAAFU_STDBY_AC_3BCC (AC-3B standby fuse failure) CCBPFU_STDBY_AC_3ACC (AC-3A standby fuse failure)	1 1	Scenario Event: "AC-3B Stbyfuse blown" Scenario Event: "AC-3A Stby Fuse blown"

Preset Item – Event 3 – Block Autostart of CA-1B

Type	Item	Value	Condition
Remote	REM:CA1B_3SS (CA-1B control selector switch)	Off (value = 3)	Scenario Event: "Block start of CA-1B"

Preset Item – Event 5 – Block Auto Start of Condenser Evacuation Pump FW-8C

Type	Item	Value	Condition
Expert	CEACWL_CLTVSP	Triggered	Scenario Event: "block start FW-8C"

Preset Item – Event 8 – PPLS Fail to Actuate

Type	Item	Value	Condition
Malfunction	ESF07 (PPLS Actuation – Train A) ESF08 (PPLS Actuation – Train B)	Block Block	Scenario Event: "PPLS auto fail"

Preset Item – Event 9 – LPSI Pumps Fail to Automatically Start

Type	Item	Value	Condition
Expert	ESEARL62_2_1X_SI_1BTVSP ESEBRL62_2_2X_SI_1BTVSP ESCBRL62_1_2X_SI_1ATVSP ESCARL62_1_1X_SI_1ATVSP	Deenergized Deenergized Deenergized Deenergized	Scenario Event: "LPSI fail to start"

Event 2 – CCW Pump AC-3C Trips

Type	Item	Value	Condition
Malfunction	BUS_1B3C_4C_4_BKR_TRIP (CCW pump AC-3C breaker fail to the trip position)	trip	When directed by examiner, trigger/activate this event. Scenario Event: "CCW Pump AC-3C Trip"

Event 3 – Plant Air Leak

Type	Item	Value	Condition
Malfunction	CAS02C (Plant Air Leak)	25	When directed by examiner, trigger/activate this event. Scenario Event: "Plant Air Leak"
Remote	REM:CAS_CA630 REM:CAS_PCV1753	0 0	When directed to close CA-121 to isolate the instrument air leak, trigger/activate this event. Scenario Event: "When directed to close CA121"

Event 4 – Pressurizer Pressure Transmitter PT-103X Fails High

Type	Item	Value	Condition
Transmitter	RCS_PT103X	2150 Ramp: 900 seconds	When directed by examiner, trigger/activate this event. Scenario Event: "PT-103x fail high"

Event 5 – Running Condenser Evacuation Pump Trips, Degrading Condenser Vacuum

Type	Item	Value	Condition
Malfunction	CES06 (Condenser Evacuation FW-8B Pump trips) CND01 (Loss of Main Condenser Vacuum)	Trip 3%, ramp = 60 sec	When directed by examiner, trigger/activate this event. Scenario Event: "Cond Evac trip"

Event 6 – Inadvertent Trip, Pressurizer Safety Valve Opens

Type	Item	Value	Condition
Remote	REM:86-1/G1-TRP (relay 86-1/G1 fail to trip position) REM: 86-2/G1-TRP (relay 86-2/G1 fail to trip position)	Trip Trip	When directed by examiner, trigger/activate this event. Scenario Event: "Trip, safety valve open"
Malfunction	RCS_RC141 (safety valve RC-141)	After reactor trip, value = 50, ramp = 15 seconds, delay = 5 seconds	

Event 7 – Total Loss of Condenser Vacuum

Type	Item	Value	Condition
Malfunction	CND01 (Loss of Main Condenser Vacuum)	100%, 300 second ramp	60 seconds after reactor trip, automatically trigger/activate event: "Complete Loss of Cond Vacuum"

<u>Booth Operator</u>:	**INITIALIZE to IC-1 and LOAD NRC 1.sce.** **ENSURE all Simulator Annunciator Alarms are ACTIVE.** **ENSURE all Control Board Tags are removed.** **ENSURE CH-1B, Charging Pump is running.** **ENSURE AC-3C, Component Cooling Water Pump running.** **ENSURE Channel X Pressurizer Pressure and Level selected.** **ENSURE FW-8B, Condenser Evacuation Pump running.** **ENSURE Reactivity Briefing Sheet printout provided with Turnover.** **ENSURE Middle-of-Life Thumb Rule Sheet provided with Turnover.** **ENSURE Containment Pressure Relief (CPR) is secured.** **ENSURE procedures in progress provided to crew in Briefing Room:** **- COPY of OI-CH-1, Chemical and Volume Control System Normal Operation, Attachment 3, Raising Charging and Letdown Flows, INITIALED through Step 2.i.**
<u>Control Room Annunciators in Alarm</u>: **NONE**	

<u>Procedure List</u>
Event 1: OP-4, Load Change and Normal Power Operation
Event 1: OI-CH-1, CVCS System Normal Operation, Attachment 3, Raising Charging and Letdown Flows
Event 2: AOP-11, Loss of Component Cooling Water
Event 3: AOP-17, Loss of Instrument Air
Event 4: ARP-CB-1/2/3/A4, Windows A-4 & B-4, PRESSURIZER PRESSURE OFF NORMAL HI-LO CHANNEL X & CHANNEL Y
Event 5: AOP-26, Turbine Malfunctions, Section I, Loss of Vacuum
Event 6: EOP-00, Standard Post Trip Actions
Event 6: EOP-03, Loss of Coolant Accident
Event 6: HR-12, Secondary Heat Removal Operation
Event 6: PC-11, Pressure Control

Operating Test :	NRC	Scenario #	1	Event #	1	Page	6	of	29
Event Description:	Raise Charging and Letdown Flow								

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When directed, RESPOND to requests from Control Room. Report back that plant conditions requested are normal unless otherwise scripted.		
<u>Indications Available</u>: **NONE**		
<u>Examiner Note</u>: The following steps are from OI-CH-1, Chemical and Volume Control System Normal Operation, Attachment 3, Raising Charging and Letdown Flows.		
+1 min	ATCO	START the selected Charging Pump CH-1B. [Step 3]
		• PLACE CH-1B switch to START.
<u>NOTES</u> 1. PIC-210 Letdown Press Cntrlr should be continuously monitored while adjusting letdown flow. 2. Steps 4 and 5 may be performed concurrently without the procedure in hand. Sign-offs may be completed after these steps are performed.		
	ATCO	RAISE bias on HIC-101-1/101-2, Letdown Throttle Valves Controller, and OBSERVE an increase in Letdown flow. [Step 4]
		• ROTATE HIC-101-1/101-2 in COUNTERCLOCKWISE direction to increase Letdown flow.
<u>Examiner Note:</u> It is acceptable to place letdown pressure control and flow control in manual or automatic control during rotation of charging pumps.		
	ATCO	ADJUST PIC-210, Letdown Press Controller as necessary to maintain Letdown pressure approximately 300 psig. [Step 5]
	ATCO	Continue to ADJUST bias on HIC-101-1/101-2 until Pressurizer level is STABILIZED at the programmed setpoint. [Step 6]
When Letdown flow is stable, PROCEED to Event 2.		

Operating Test :	NRC	Scenario #	1	Event #	2	Page	7	of	29
Event Description:	Component Cooling Water Pump Trip								

Time	Position	Applicant's Actions or Behavior
Booth Operator:		**When directed, EXECUTE Event 2.** **When contacted to report pump conditions, Auxiliary Building Operator reports normal conditions. Water Plant Operator reports breaker tripped on overcurrent**
Indications Available:		
CB-1/2/3/A2 – CCW PUMPS TRIP **CB-1/2/3/A2 – CC WATER FROM DISCH HEADER FLOW LO** **CB-1/2/3/A2 – CCW PUMPS AC-3A/B/C STANDBY START** **CB-1/2/3/A2 – AUXILIARY COOLANT FROM CRDM FLOW LO** **CCW Pump AC-3C white TRIP and green STOP lights lit** **Multiple loss of CCW flow alarms**		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
	ATCO	INFORM CRS of CCW Pump AC-3C trip with NO auto start of standby pump.
Examiner Note:		**ATCO may "Operate to Mitigate" per OPD 3-01 and START a CCW Pump.**
	CRS	REFER to AOP-11, Loss of Component Cooling Water.
Examiner Note:		**The following steps are from AOP-11, Loss of Component Cooling Water.**
	ATCO	VERIFY normal CCW/RW System operation: [Step 4.1]
		• START CCW Pump AC-3A or AC-3B. [Step 4.1.a]
		• VERIFY CCW System pressure ≥ 60 psig. [Step 4.1.b]
		• DETERMINE AC-1B, Raw Water CCW Heat Exchanger in service. [Step 4.1.c]
		• DETERMINE RCP Coolers CCW Valves, HCV-438A/B/C/D all OPEN. [Step 4.1.d]
	ATCO	VERIFY Raw Water Pump operating. [Step 4.2]
	ATCO	If CCW Surge Tank level < 42 inches, FILL the CCW Surge Tank: [Step 4.3]
		• OPEN LCV-2801, CCW Surge Tank Makeup Valve, to refill CCW Surge Tank. [Step 4.3.a]
		• PLACE LCV-2801 in CLOSE or AUTO. [Step 4.3.b]

Operating Test :	NRC	Scenario #	1	Event #	2	Page	8	of	29
Event Description:	Component Cooling Water Pump Trip								

Time	Position	Applicant's Actions or Behavior
	CRS	EVALUATE Technical Specification LCO 2.4, Containment Cooling
		• LCO 2.4.(1).a – Component Cooling Water Pump AC-3C
		• CONDITION 2.4.(1).a – Component Cooling Water Pump AC-3C inoperable. • ACTION 2.4.(1).b – RESTORE Component Cooling Water Pump AC-3C within 7 days <u>OR</u> PLACE Reactor in HOT SHUTDOWN condition within 12 hours.
When Technical Specifications are addressed, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	1	Event #	3	Page	9	of	29
Event Description:	Plant Air System Leak								

Time	Position	Applicant's Actions or Behavior
Booth Operator:		**When directed, EXECUTE Event 3.** **- Plant Air System leak @ 25%.**
Indications Available: **CB-10,11/A21 – PLANT AIR PRESS LO** **PI-1700, Plant Air Press lowering on CB-10,11**		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS of Plant Air System pressure less than 96 psig and lowering.
Examiner Note:		**BOPO may "Operate to Mitigate" per OPD 3-01 and START an Air Compressor.**
	CRS	REFER to AOP-17, Loss of Instrument Air.
Examiner Note:		**The following steps are from AOP-17, Loss of Instrument Air.**
	BOPO	ENSURE all available Air Compressors start. [Step 4.1]
		• START Air Compressor CA-1A.
		• START Air Compressor CA-1B.
Booth Operator:		**If contacted, REPORT Compressors, Dryers, and Filters appear to be operating normally.**
Booth Operator:		**If contacted, PLACE standby Air Compressor CA-1B in service.**
	BOPO	CONTACT Equipment Operator to ensure proper operation of Instrument Air Compressors, Dryers, and Filters. [Step 4.2]
	CREW	ANNOUNCE and REPEAT message using Plant Communication System: [Step 4.3]
		• "Attention all personnel, attention all personnel; there is a plant air leak in progress. Report any large air usage to the Control Room."
	CRS	DIRECT available operators to search for source of air leakage. [Step 4.4]

Operating Test : NRC	Scenario # 1	Event # 3	Page 10 of 29
Event Description: Plant Air System Leak			

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When contacted, REPORT leak is downstream of PCV-1753. When directed, execute simulator operation to isolate leak and report CA-121, Service Air Supply System Manual Isolation Valve is CLOSED.		
	BOPO	DETERMINE Instrument Air pressure is < 80 psig, and CONTACT Equipment Operator to VERIFY PCV-1753, Service Air System Automatic Isolation Valve CLOSED. [Step 4.5]
	CRS	DETERMINE Instrument Air pressure slowly returning to normal after service air was isolated. [Step 4.6]
		• VERIFY CA-121, Service Air Supply System Manual Isolation Valve is closed. [Step 4.6.a]
		• GO TO Section 5.0, Exit Conditions. [Step 4.6.b]
<u>Examiner Note</u>: Plant Air System remains isolated for the duration of the Scenario.		
When Instrument Air pressure returns to normal, PROCEED to Event 4.		

Operating Test : NRC	Scenario # 1	Event # 4	Page 11 of 29
Event Description: Pressurizer Pressure Control Channel Failure			

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When directed, EXECUTE Event 4.		
- Pressurizer Pressure Control Channel PT-103X fails to 2150 psia on 15 minute ramp.		
<u>Indications Available</u>:		
CB-1/2/3/A4 – PRESSURIZER PRESSURE OFF NORMAL HI-LO CHANNEL Y (1st alarm)		
CB-1/2/3/A4 – PRESSURIZER PRESSURE OFF NORMAL HI-LO CHANNEL X (2nd alarm ~ 2 min later)		
<u>Examiner Note</u>: Due to the nature of this failure, Channel Y alarm comes in 1st as it senses PZR pressure < 2080 psia (alarm setpoint) even though Channel X is the Controlling Channel. As the Channel X setpoint failure ramps in and reaches > 2145 psia (alarm setpoint), Channel X annunciator will alarm.		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
<u>Examiner Note</u>: ATCO may "Operate to Mitigate" per OPD 4-09 and TRANSFER to Channel Y.		
<u>Examiner Note</u>: The following steps are from ARP-CB-1/2/3/A4, Window A-4 for Channel X.		
	ATCO	VERIFY RCS pressure using all available indications. [Step 1]
		• MONITOR Pressurizer Pressure and operation of PC-103X. [Step 1.1]
		• DETERMINE PC-103X is <u>not</u> controlling pressure and PLACE HC-103, Pressurizer Pressure Channel Selector Switch to CHAN Y position. [Step 1.1.1]
	CRS	• MAY REFER to OI-RC-7, Reactor Coolant System Pressure Control Normal Operation. [Step 1.1.2]
	ATCO	PERFORM the following for the low pressure condition: [Step 2]
	CRS	• REFER to Technical Specification LCO 2.10.4.(5) if pressure ≤ 2075 psia. [Step 2.1]
	ATCO	• DETERMINE Pressurizer Spray Valves PCV-103-1 and PCV-103-2 are CLOSED. [Step 2.2]
	ATCO	• ENSURE <u>all</u> Pressurizer Heater Control Switches in AUTO or ON. [Step 2.3]
	ATCO	• ENERGIZE additional Pressurizer Heaters as required. [Step 2.4]
	ATCO	• DETERMINE Pressurizer level NOT lowering on LR-101X/LR-101Y. [Step 2.5]

Operating Test : NRC	Scenario # 1	Event # 4	Page 12 of 29
Event Description: Pressurizer Pressure Control Channel Failure			

Time	Position	Applicant's Actions or Behavior
	ATCO	• VERIFY VCT level trend on LI-219. [Step 2.6]
	CRS	EVALUATE Technical Specification LCO 2.10.4, Power Distribution Limits
		• LCO 2.10.4.(5) – DNBR Margin during Power Operation above 15% of Rated Power
		• CONDITION 2.10.4.(5).(a).(ii) – Pressurizer Pressure < 2075 psia. • ACTION 2.10.4.(5).(b) – RESTORE Pressurizer Pressure within its limit within 2 hours **<u>or</u>** REDUCE power to less than 15% of rated power within the next 8 hours.
When Technical Specifications have been addressed, PROCEED to Event 5.		

Operating Test :	NRC	Scenario #	1	Event #	5	Page	13	of	29
Event Description:	Partial Loss of Condenser Vacuum / Condenser Evacuation Pump Trip With Auto Start Failure								

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Event 5 - Partial Loss of Condenser vacuum @ 5% on 3 minute ramp. - Condenser Evacuation Pump FW-8B trip. - Condenser Evacuation Pump FW-8C Auto Start failure.**		
Examiner Note: rate of lowering condenser vacuum may be modified at your discretion to advance or retard the pace of this and the next event.		
Indications Available: **CB-10,11/A9 – VACUUM PUMP B STOPPED OR SEAL WATER TEMP HI Emergency Response Facility Computer System (ERFCS) Alarm on Low Condenser Vacuum Condenser Evacuation Pump FW-8B green STOP light lit Lowering Condenser Vacuum on PI-925A/B or P0976A/B**		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS of lowering Condenser vacuum and Condenser Evacuation Pump FW-8B trip.
Examiner Note: **BOPO may "Operate to Mitigate" per OPD 4-09 and START FW-8C.**		
	CRS	REFER to AOP-26, Turbine Malfunctions, Section I, Loss of Vacuum.
Examiner Note: **The following steps are from AOP-26, Turbine Malfunctions, Section I, Loss of Vacuum.**		
	BOPO	MONITOR Condenser vacuum on ERF Computer System/PI-925A/ PI-925B/P0976A/P0976B. [Step 4.1]
	BOPO	ENSURE all Condenser Evacuation Pumps are running. [Step 4.2]
		• START FW-8C, Condenser Evacuation Pump.
CAUTION The Turbine should not be operated with a Generator load of less than 150 MW when vacuum is less than or equal to 23.85" Hg (ERF, P0976A/B) or 6.07" Hg absolute (PI-925A/B) due to possible overheating of final stage blades.		
	CRS	If Condenser vacuum is < 25" Hg **or** 4.92" Hg Absolute, COMMENCE a plant shutdown to restore vacuum per AOP-05 Emergency Shutdown. [Step 4.3]

Operating Test :	NRC	Scenario #	1	Event #	5	Page	14	of	29
Event Description:	Partial Loss of Condenser Vacuum / Condenser Evacuation Pump Trip With Auto Start Failure								

Time	Position	Applicant's Actions or Behavior

Examiner Note: The following steps are from AOP-05, Emergency Shutdown.

NOTE

TDB-III-23a and the Power Ascension/Power Reduction Strategy (PAPRs) provide guidance for the shutdown.

	CRS	CONTACT Reactor Engineer if additional guidance is required. [Step 4.1]

NOTE

Operation of more than one Charging Pump will raise the rate of the power reduction.

Examiner Note: Unless directed, boration will occur from the Safety Injection Refueling Water Tank (SIRWT) when in AOP-05 to avoid time constraints.

	CRS	If borating from SIRWT, COMMENCE boration by performing the following: [Step 4.2]
	ATCO	• DETERMINE Charging Pump CH-1A is RUNNING. [Step 4.2.a]
	ATCO	• OPEN LCV-218-3, Charging Pump Suction SIRWT Isolation Valve. [Step 4.2.b]
	ATCO	• CLOSE LCV-218-2, VCT Outlet Valve. [Step 4.2.c]

	CRS	DETERMINE Boration alignment from CVCS NOT required. [Step 4.3]

	CRS	NOTIFY Energy Marketing of power reduction. [Step 4.4]

NOTE

During the power reduction, maintain T_C PER TDB Figure III.1, T_{ave} Program.

	BOPO	MAINTAIN RCS Temperature Control via Turbine Load per HR-12, Secondary Heat Removal Operation: [Step 4.5]
		• MAINTAIN T_{COLD} 527°F to 547°F AND
		• MAINTAIN T_{COLD} +0°F to -1°F of program.

Time	Position	Applicant's Actions or Behavior
	ATCO	MAINTAIN Pressurizer Level via Charging and Letdown per IC-11, Inventory Control: [Step 4.6]
		• MAINTAIN Pressurizer Level 45% to 60% AND
		• MAINTAIN Pressurizer Level within 4% of program.
When power has been lowered 3% to 5%, PROCEED to Events 6, 7, 8, and 9.		

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	16	of 29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure							

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Events 6, 7, 8, and 9. **- Inadvertent Turbine Trip.** **- Pressurizer Safety Valve fails 50% open on Reactor Trip.** **- Loss of Condenser Vacuum @ 100%.** **- Pressurizer Pressure Low Signal (PPLS) Actuation failure.** **- Low Pressure Safety Injection Pumps start failure.**		
Indications Available: **Numerous Reactor Trip and Turbine Trip Alarms.**		
+10 sec	ATCO	RECOGNIZE Reactor Trip due to Turbine Trip.
	CRS	DIRECT performance of EOP-00, Standard Post Trip Actions.
Examiner Note: The following steps are from EOP-00, Standard Post Trip Actions.		
	ATCO	VERIFY Reactivity Control: [Step 1]
		• VERIFY ALL of the following: [Step 1.a]
		• VERIFY no more than one Regulating or Shutdown CEA NOT inserted.
		• VERIFY Reactor Power is LOWERING.
		• VERIFY Startup Rate is NEGATIVE.
		• MONITOR plant for an uncontrolled RCS Cooldown. [Step 1.b]
	CRS	DETERMINE Reactivity Control criteria SATISFIED.
	BOPO	VERIFY Turbine Trip: [Step 2]
		• VERIFY HP & LP Stop and Intercept Valves CLOSED.
	BOPO	ENSURE all Generator Breakers are tripped: [Step 3]
		• DETERMINE Generator Output Breaker 3451-4 tripped.
		• DETERMINE Generator Output Breaker 3451-5 tripped.
		• DETERMINE Generator Field Breaker 41E/G1F tripped.
	BOPO	VERIFY 4160 V Safeguards Buses 1A3 and 1A4 are ENERGIZED. [Step 4]

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	17	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE Maintenance of Vital Auxiliaries criteria SATISFIED.
Examiner Note: Diesel Generators only start after safeguards (PPLS) actuation.		
	BOPO	VERIFY both Diesel Generators RUNNING on Safety Injection Actuation Signal. [Step 5]
	BOPO	VERIFY 4160 V Non-Safeguards Buses 1A1 and 1A2 are ENERGIZED. [Step 6]
	BOPO	VERIFY 125 VDC Buses 1 and 2 are ENERGIZED. [Step 7]
	BOPO	VERIFY Instrument Air is AVAILABLE: [Step 8]
		• DETERMINE Instrument Air pressure ≥ 90 psig.
		• DETERMINE Instrument Air Compressor CA-1A RUNNING.
	ATCO	VERIFY Component Cooling Water System operation NORMAL: [Step 9]
		• DETERMINE at least one CCW pump RUNNING. [Step 9.a]
		• DETERMINE CCW Pump discharge pressure ≥ 60 psig. [Step 9.b]
		• DETERMINE HCV-438A/B/C/D, CCW to RCP Coolers OPEN. [Step 9.c]
		• DETERMINE at least one Raw Water Pump RUNNING. [Step 9.d]
	CRS	VERIFY RCS Inventory Control criteria satisfied: [Step 10]
	ATCO	• DETERMINE PZR level NOT between 30% and 70% and NOT TRENDING to between 45% and 60%.
		• DETERMINE RCS subcooling ≥ 20°F:
		• [CA] RESTORE Inventory Control by manually controlling Charging and Letdown. [Step 10.1.a]
	CRS	DETERMINE RCS Inventory Control criteria NOT SATISFIED.
	CRS	VERIFY RCS Pressure Control criteria satisfied: [Step 11]

	ATCO	• DETERMINE RCS pressure NOT between 1800 psia and 2300 psia and NOT trending to between 2050 psia and 2150 psia.
		• [CA] DETERMINE RCS pressure < 2300 psia and PORV NOT open. [Step 11.1]
		• [CA] DETERMINE RCS pressure ≤ 1350 psia and TRIP one RCP in each loop. [Step 11.2]
		• [CA] DETERMINE RCS pressure ≤ 1600 psia and ENSURE PPLS actuated. [Step 11.3]
CRITICAL TASK STATEMENT		**Manually Actuate Pressurizer Pressure Low Signal (PPLS) when RCS Pressure ≤ 1600 psia and before 1350 psia to ensure SIAS / VIAS / CIAS Activation.** **Pressure at Time of PPLS Trip ______ psia.**
CRITICAL TASK	ATCO	DETERMINE PPLS relays NOT tripped and manually ACTUATE PPLS.
	ATCO	• [CA] INSERT and TURN keys at 86A/PPLS Test Switch & 86B/PPLS Test Switch on AI-30A & AI-30B. [Step 11.3.a]
		• [CA] DETERMINE PPLS relays 86A/PPLS / 86B/PPLS / 86A1/PPLS / 86B1/PPLS have TRIPPED. [Step 11.3.b]
		• [CA] DETERMINE VIAS relays 86A/VIAS / 86B1/VIAS / 86B/VIAS / 86A1/VIAS have TRIPPED. [Step 11.3.c]
		• [CA] DETERMINE SIAS relays 86A/SIAS / 86AX/SIAS / 86B1/SIAS / 86B1X/SIAS / 86B/SIAS / 86BX/SIAS / 86A1/SIAS / 86A1X/SIAS have TRIPPED. [Step 11.3.d]
		• [CA] DETERMINE CIAS relays 86A/CIAS / 86B1/CIAS / 86B/CIAS / 86A1/CIAS have TRIPPED. [Step 11.3.e]
		• [CA] ENSURE HPSI / LPSI / Charging Pumps RUNNING. [Step 11.3.f]
		• DETERMINE HPSI Pumps SI-2A & SI-2B **or** SI-2B & SI-2C RUNNING.
	ATCO	• DETERMINE LPSI Pumps NOT RUNNING and manually START SI-1A and SI-1B.
		• DETERMINE Charging Pumps CH-1A, CH-1B, & CH-1C RUNNING.
		• [CA] ENSURE adequate SI flow per IC-13, Safety Injection Flow vs. Pressurizer Pressure. [Step 11.3.g]
		• [CA] DETERMINE Emergency Boration in progress. [Step 11.3.h]

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE RCS Pressure Control criteria NOT SATISFIED.
Examiner Note:		**The following steps are from RC-11, Emergency Boration Verification.**
	ATCO	ENSURE the following valves are CLOSED: [Step 1]
		• FCV-269X, Demin Water Makeup Valve
		• FCV-269Y, Boric Acid Makeup Valve
		• HCV-264, CH-4A Recirc Valve
		• HCV-257, CH-4B Recirc Valve
	ATCO	VERIFY all the following valves OPEN: [Step 2]
		• HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
		• HCV-265, CH-11A Gravity Feed Valve
		• HCV-258, CH-11B Gravity Feed Valve
	ATCO	ENSURE all available Boric Acid Pumps RUNNING: [Step 3]
		• CH-4A, Boric Acid Pump
		• CH-4B, Boric Acid Pump
	ATCO	ENSURE all available Charging Pumps RUNNING: [Step 4]
		• CH-1A, Charging Pump
		• CH-1B, Charging Pump
		• CH-1C, Charging Pump
	ATCO	ENSURE the following valves are CLOSED: [Step 5]
		• LCV-218-2, VCT Outlet Valve
		• LCV-218-3, Charging Pump Suction SIRWT Isolation Valve
Examiner Note:		**The following steps continue from EOP-00, Standard Post Trip Actions.**
	CRS	VERIFY Core Heat Removal criteria satisfied: [Step 12]

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	20 of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure							

Time	Position	Applicant's Actions or Behavior
	ATCO	• DETERMINE RCP NPSH requirements met.
		• DETERMINE at least one RCP operating.
		• DETERMINE Core $\Delta T \leq 10°F$.
Examiner Note:		**Depending on Crew actions, RCS subcooling will be lost in either EOP-00, SPTAs or EOP-03, LOCA.**
CRITICAL TASK STATEMENT		**Stop All Reactor Coolant Pumps (RCPs) when Subcooling is approaching or is less than 20°F, before 0°F due to Loss of Net Positive Suction Head (NPSH) per RCP NPSH Curve.**
CRITICAL TASK	ATCO	DETERMINE RCP subcooling < 20°F and PERFORM the following:
	ATCO	• STOP all RCPs.
	BOPO	• [CA] PLACE TCV-909, Temperature Controller in MANUAL on DCS. [Step 12.2.a]
	BOPO	• [CA] ENSURE TCV-909, Temperature Controller OUTPUT is zero (0). [Step 12.2.b]
	CRS	• [CA] VERIFY Natural Circulation in at least one Loop. [Step 12.2.c]
		• [CA] DETERMINE Core $\Delta T \leq 50°F$.
		• [CA] DETERMINE difference between CETs and RCS T_{HOT} is ≤ 10°F on ERF "CHR" display.
		• [CA] DETERMINE RCS subcooling is ≥ 20°F.
		• [CA] DETERMINE T_{HOT} and T_{COLD} are stable or lowering.
	CRS	DETERMINE Core Heat Removal criteria NOT SATISFIED.
NOTE If Instrument Air to valves HCV-1105, HCV-1106, HCV-1107A/B and HCV-1108A/B is not available, throttling of these valves is not possible. Open or close operation of these valves is possible for a minimum of three cycles.		
	CRS	VERIFY RCS Heat Removal criteria satisfied:

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	21	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

Time	Position	Applicant's Actions or Behavior
	BOPO	VERIFY Main Feedwater is restoring SG levels to 35% to 80% NR and 73% to 94% WR. [Step 13]
		• DETERMINE FCV-1101 & FCV-1102 Feed Regulating Valves CLOSED. [Step 13.a]
		• DETERMINE HCV-1105 & HCV-1106 Feed Regulating Bypass Valves ramped to between 40% & 45% OPEN. [Step 13.b]
	BOPO	• PLACE 43/FW switch in OFF. [Step 13.c]
		• ENSURE no more than one Main Feedwater Pump RUNNING. [Step 13.d]
		• ENSURE no more than one Condensate Pump RUNNING. [Step 13.e]
	BOPO	• STOP running Heater Drain Pumps FW-5A, FW-5B, and/or FW-5C. [Step 13.f]
		• ENSURE both sets of SG Blowdown Isolation Valves CLOSED. [Step 13.g]
	BOPO	VERIFY Steam Dump and Bypass Valves controlling both of the following: [Step 14]
		• VERIFY RCS T_{COLD} between 525°F and 535°F.
		• VERIFY Steam Generator pressure between 850 psia & 925 psia.
	BOPO	• [CA] DETERMINE loss of Condenser vacuum and PLACE HCV-1040, Atmosphere Dump Valve in service.
		• SELECT HCV-1040 on DCS Secondary Screen.
	CRS	DETERMINE RCS Heat Removal criteria SATISFIED.
	CRS	VERIFY Containment Isolation criteria satisfied:
	ATCO	VERIFY Normal Containment conditions exist: [Step 15]
		• DETERMINE unexpected rise in Containment Sump level. [Step 15.a]
	ATCO	• DETERMINE Containment Area Radiation Monitors in ALARM. [Step 15.b]
	ATCO	• DETERMINE Containment Ventilation Radiation Monitors RM-050 and RM-051 in ALARM. [Step 15.c]
		• [CA] ENSURE VIAS has ACTUATED and 86A/VIAS, 86A1/VIAS, 86B/VIAS, & 86B1/VIAS relays TRIPPED.

Time	Position	Applicant's Actions or Behavior
		• [CA] ENSURE RM-050 & RM-051 Containment Radiation Monitor Sample Pump STOPPED.
		• [CA] ENSURE RM-065, Post Accident Control Room Iodine Monitor RUNNING.
		• DETERMINE SG Blowdown & Condenser Off Gas Radiation Monitors (RM-054A / RM-054B / RM-057) NOT in alarm. [Step 15.d]
		• DETERMINE SG Blowdown & Condenser Off Gas Radiation Monitors (RM-054A / RM-054B / RM-057) NOT trending to alarm. [Step 15.e]
	ATCO	• VERIFY Containment conditions: [Step 15.f]
		• DETERMINE Containment pressure < 3 psig.
		• DETERMINE Containment temperature > 120°F.
	CRS	DETERMINE Containment Integrity criteria NOT SATISFIED.
	CRS	DIAGNOSE event in progress: [Step 16]
		• DETERMINE Reactivity Control requirements met.
		• DETERMINE both DC buses energized.
		• DETERMINE at least one Vital 4160 V Bus energized.
		• DETERMINE at least one Non-Vital 4160 V Bus energized.
		• VERIFY at least one RCP running.
		• If not, CONSIDER EOP-02, Loss of Offsite Power/Forced Circulation.
		• DETERMINE adequate Feedwater flow to at least one SG.
		• VERIFY Pressurizer pressure > 1800 psia with high subcooled margin, normal SG pressure, and no indications of primary to secondary leakage.
		• If not, CONSIDER EOP-03, Loss of Coolant Accident.

NOTE

Certain events (i.e., LOCA, SGTR, UHE and Loss of All Feedwater) do not require offsite power in order to adequately, mitigate the effects of the accident. For this reason, the LOCA, SGTR, UHE or Loss of All Feedwater procedure may be implemented even if a Loss of Offsite Power has also occurred.

		• DETERMINE all Safety Function Acceptance Criteria NOT SATISFIED.

Operating Test : NRC	Scenario # 1	Event # 6, 7, 8, & 9 Page 23 of 29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure	
Time	Position	Applicant's Actions or Behavior

Time	Position	Applicant's Actions or Behavior
		• DETERMINE single event in progress and transition to EOP-03, Loss of Coolant in Accident.
<u>Examiner Note</u>: The following steps are from EOP-03, Loss of Coolant Accident.		
	CRS	CONFIRM Standard Post Trip Actions have been performed. [Step 1]
	CRS	CONFIRM Loss of Coolant Accident Diagnosis: [Step 2]
		• VERIFY Safety Function Status Check Acceptance Criteria being satisfied. [Step 2.a]
		• VERIFY CIAS is NOT present and SAMPLE both SGs. [Step 2.b]
	CRS	IMPLEMENT the Emergency Plan. [Step 3]
		• Time: __________
<u>NOTE</u> Floating Step BB, <u>Minimizing DC Loads</u>, requires operator action within 15 minutes of loss of either battery charger.		
	CREW	MONITOR the Floating Steps. [Step 4]
	CRS	DETERMINE RCS pressure ≤ 1600 psia and Containment pressure ≤ 5 psig and CSAS NOT present. [Step 5]
	ATCO	DETERMINE RCS pressure ≤ 1600 psia and VERIFY Engineered Safeguards Actuation: [Step 6]
		• DETERMINE PPLS relays 86A/PPLS / 86B/PPLS / 86A1/PPLS / 86B1/PPLS have TRIPPED. [Step 6.a]
		• DETERMINE VIAS relays 86A/VIAS / 86B1/VIAS / 86B/VIAS / 86A1/VIAS relays TRIPPED. [Step 6.b]
		• DETERMINE SIAS relays 86A/SIAS / 86AX/SIAS / 86B1/SIAS / 86B1X/SIAS / 86B/SIAS / 86BX/SIAS / 86A1/SIAS / 86A1X/SIAS have TRIPPED. [Step 6.c]
		• DETERMINE CIAS relays 86A/CIAS / 86B1/CIAS / 86B/CIAS / 86A1/CIAS relays TRIPPED. [Step 6.d]

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	24	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE Containment pressure ≤ 5 psig. [Step 7]
	ATCO	DETERMINE SIAS actuated and OPTIMIZE SI flow. [Step 8]
		• ENSURE HPSI / LPSI / Charging Pumps RUNNING. [Step 8.a]
		• DETERMINE HPSI Pumps SI-2A & SI-2B RUNNING.
		• DETERMINE LPSI Pumps SI-1A and SI-1B RUNNING.
		• DETERMINE Charging Pumps CH-1A, CH-1B, & CH-1C RUNNING.
		• DETERMINE Emergency Boration in progress per RC-11, Emergency Boration Verification. [Step 8.b]
		• DETERMINE adequate SI flow per IC-13, Safety Injection Flow vs. Pressurizer Pressure. [Step 8.c]
	CRS	VERIFY RCP operating parameters: [Step 9]
	ATCO	• ENSURE at least one RCP stopped if T_{COLD} < 500°F. [Step 9.a]
	ATCO	• ENSURE one RCP stopped in each loop if RCS pressure ≤ 1350 psia. [Step 9.b]
	ATCO	• ENSURE all RCPs STOPPED if RCS pressure < NPSH requirements per PC-12, RCS Pressure-Temperature Limits. [Step 9.c]
	CRS	RECORD time of SIAS initiation. [Step 10]
		• Time: __________
	ATCO	VERIFY normal Component Cooling Water (CCW) and Raw Water (RW) System operation: [Step 11]
		• ENSURE at least 2 CCW Pumps RUNNING. [Step 11.a]
		• VERIFY CCW Pump discharge pressure ≥ 60 psig. [Step 11.b]
		• ENSURE at least 2 RW Pumps RUNNING. [Step 11.c]
		• ENSURE at least 3 CCW Heat Exchangers in service. [Step 11.d]
		• ENSURE all RCP Coolers CCW Valves OPEN. [Step 11.e]

Time	Position	Applicant's Actions or Behavior
		NOTE Do **NOT** isolate a PORV if the pressurizer is water solid.
	ATCO	VERIFY PORVs and PZR Code Safety Valves are CLOSED. [Step 12]
		• DETERMINE Quench Tank temperature, pressure, and level in ALARM. [Step 12.a]
		• DETERMINE PZR Safety Valve discharge temperature high in ALARM. [Step 12.b]
	ATCO	• NOTIFY CRS that a PZR Safety Valve is OPEN.
		• DETERMINE PORV Acoustic Flow Alarms are CLEAR. [Step 12.c]
		NOTE Rising Radiation Monitor RM-053 count rate, rising CCW surge tank level or rising CCW surge tank pressure may be indications of a RCS-to-CCW leak.
	ATCO	DETERMINE RCS to CCW leak is NOT in progress. [Step 13]
	CRS	DETERMINE LOCA is inside Containment. [Step 14]
	ATCO	PERFORM the following for a LOCA inside Containment: [Step 15]
		• PLACE HC-504A, CNTMT SUMP PUMP WD-3A CONTROL SWITCH, in PULL-TO-LOCK. [Step 15.a]
		• PLACE HC-504B, CNTMT SUMP PUMP WD-3B CONTROL SWITCH, in PULL-TO-LOCK. [Step 15.a]
		• CLOSE HCV-506A, Containment Sump Isolation Valve. [Step 15.b]
		• CLOSE HCV-506B, Containment Sump Isolation Valve. [Step 15.b]
	ATCO	VERIFY all the following conditions exist: [Step 16]
		• DETERMINE all HPSI Pumps are operating.
		• DETERMINE SI flowrate is acceptable per IC-13 SI Flow vs. PZR Pressure.
		• DETERMINE Representative CET temperature less than superheat.
		• DETERMINE Reactor Vessel Level Monitoring System > 43% and NOT lowering.

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	26	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

Time	Position	Applicant's Actions or Behavior
	ATCO	ENSURE SI-2C, HPSI Pump Control Switch in PULL-TO-LOCK.
	ATCO	DETERMINE NONE of the following conditions exist: [Step 17]
		• SI flowrate is less than IC-13 SI Flow vs. PZR Pressure.
		• Representative CET temperature greater than superheat.
		• Reactor Vessel Level Monitoring System < 43% and lowering.
	CRS	DETERMINE RCS leak is NOT isolated. [Step 18]
	BOPO	DETERMINE Steam Generator Isolation Signal (SGIS) NOT actuated. [Step 19]
	BOPO	DETERMINE SG levels between 35% and 85% NR using Main Feedwater. [Step 20]
		• MAINTAIN Feedwater flow per HR-15, Main Feed Pump Operation. [Step 20.a]
		• CONTROL Feedwater flow per HR-11, Manual Feet Control (DCS). [Step 20.b]

CAUTION

Failure to place the Containment Spray Pumps to Pull to Lock may allow actuation of Spray into Containment. This can lead to Containment Sump Blockage.

Time	Position	Applicant's Actions or Behavior
	ATCO	SECURE all Containment Spray flow: [Step 21]
		• PLACE SI-3A, Containment Spray Pump in PULL-TO-LOCK.
		• PLACE SI-3B, Containment Spray Pump in PULL-TO-LOCK.
		• PLACE SI-3C, Containment Spray Pump in PULL-TO-LOCK.

CAUTION

1) When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr. When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.
2) No more than three RCPs shall be in operation when RCS temperature is less than 500°F.

Time	Position	Applicant's Actions or Behavior
	CRS	COMMENCE a Steam Generator cooldown per HR-12, Secondary Heat Removal Operation. [Step 22]

Time	Position	Applicant's Actions or Behavior
		• Time: ___________
	CRS	MAINTAIN RCS pressure per PC-12, Pressure-Temperature Limits. [Step 23]
	BOPO	• CONTROL RCS heat removal per HR-12, Secondary Heat Removal Operation. [Step 23.a]
	ATCO	• CONTROL RCS pressure per PC-11, Pressure Control. [Step 23.b]
	ATCO	• If HPSI Stop and Throttle criteria are met, CONTROL Charging, Letdown, and HPSI flow per IC-11, Inventory Control. [Step 23.c]
NOTE Voiding of the RCS is indicated by the inability to depressurize to SDC entry pressure. Attachment IC-14, RCS Void Elimination, provides guidance to correct this condition.		
	CRS	COMMENCE depressurizing RCS to ≤ 300 psia using any of the following per PC-11, Pressure Control: [Step 24]
		• CONTROL Pressurizer Spray flow.
		• CONTROL Charging and Letdown flow.
		• THROTTLE HPSI Pumps.
		• Time: ___________
CRITICAL TASK STATEMENT		**Commence a Cooldown and Depressurization of the Reactor Coolant System before Reactor Vessel Level Monitoring System (RFLMS) is less than 100%, indicating a bubble has formed in the head, to Reestablish RCS Inventory Control while maintaining RCS Heat Removal.**
CRITICAL TASK	BOPO	IMPLEMENT HR-12, Secondary Heat Removal Operation, to lower RCS temperature.
Examiner Note: The following steps are from HR-12, Secondary Heat Removal Operation.		
	BOPO	ENSURE Turbine Control is in MANUAL. [Step 1]
	BOPO	• [CA] DETERMINE Turbine NOT online and GO TO Step 4.

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	28	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

NOTES

1. In MANUAL, single arrows are 1% / double arrows are 5% change in OUTPUT value.
2. While Steam Dump and Bypass Control is in Temperature/Pressure Mode, the controllers PC0910 and TC0909_PI will alternate between controls, depending on the higher output signal. A red square outlining the controlling function signify which parameter is in control.
3. Steps 4 through 11 may be performed as needed, and in any order.

CAUTIONS

When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr.
When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.

Time	Position	Applicant's Actions or Behavior
	BOPO	If Steam Dump and Bypass (SD&B) is available, CONTROL RCS temperature with a single SD&B Valve. [Step 4]
	BOPO	• [CA] DETERMINE Steam Dump and Bypass is NOT available and GO TO Step 9.

Examiner Note: **HCV-1040, Atmospheric Dump Valve, may already be in service following the Loss of Condenser Vacuum that occurred on Reactor Trip.**

Time	Position	Applicant's Actions or Behavior
	BOPO	If HCV-1040, is available, CONTROL RCS temperature as follows: [Step 9]
		• DEPRESS the valve toggle to SELECT HCV-1040. [Step 9.a]
		• PUSH UP and DOWN arrows as required to ADJUST HCV-1040 output as needed. [Step 9.b]
CRITICAL TASK	ATCO	IMPLEMENT PC-11, Pressure Control, to lower RCS pressure.

Examiner Note: **The following steps are from PC-11, Pressure Control. PC-12, RCS Pressure-Temperature Limits (graph), is maintained on a Control Room hardcopy.**

CAUTION

A charging header flow path must be maintained at all times.

Time	Position	Applicant's Actions or Behavior
	ATCO	MAINTAIN RCS pressure per the PC-12, RCS Pressure-Temperature Limits graph: [Step 1]
		• DETERMINE Steps 1.a through 1.d N/A. [Step 1.e]

Operating Test :	NRC	Scenario #	1	Event #	6, 7, 8, & 9	Page	29	of	29
Event Description:	Inadvertent Main Turbine Trip / Pressurizer Safety Valve Failure / Loss of Condenser Vacuum / Pressurizer Pressure Low Signal Actuation Failure / Low Pressure Safety Injection Pumps Start Failure								

Time	Position	Applicant's Actions or Behavior
		• CONTROL Auxiliary Spray flow as necessary by operating the following: [Step 1.e]
		• HCV-240, PZR Auxiliary Spray Isolation Valve • HCV-249, PZR Auxiliary Spray Isolation Valve • HCV-238, Loop 1 Charging Isolation Valve • HCV-239, Loop 2 Charging Isolation Valve
		• If HPSI Stop and Throttle criteria is met, CONTROL Pressurizer level using Charging, Letdown, and/or HPSI flow per IC-11, Inventory Control. [Step 1.f]
		• CONTROL RCS heat removal per HR-12, Secondary Heat Removal Operation. [Step 1.g]
When RCS Cooldown and Depressurization is in progress, TERMINATE the scenario.		

Facility:	Fort Calhoun Station	Scenario No.:	2	Op Test No.:	Dec 2015 NRC
Examiners:				Operators:	

Initial Conditions: 100% power MOL - RCS Boron is 482 ppm (by sample).

Turnover: Maintain steady-state power conditions. Perform Containment Spray Pump SI-3A Operability Test per OI-CS-1, Containment Spray Pump Normal Operation, Attachment 1A.

Critical Tasks:

- Commence an Emergency Boration of the RCS Due to 2 or more Stuck CEAs when Diesel Generator DG-1 Breaker is Closed and Bus 1A3 is Reenergized to Restore Reactivity Control. (Event 7)
- Restore Feedwater Flow to At Least One Steam Generator to Reestablish any SG as a Heat Sink. (Event 6)
- Restore Power to any 4160 V Safeguards Bus using a Diesel Generator to Reestablish Maintenance of Vital Auxiliaries and Allow Branching to Meet other Safety Functions During a Station Blackout. (Event 8)

Event No.	Malf. No.	Event Type*	Event Description
1 +15 min		N (ATCO)	Perform OI-CS-1, Containment Spray Normal Operation, Attachment 1A, SI-3A Containment Spray Pump Operability Test.
2 +20 min			Severe Thunderstorm Watch from the National Weather Service. AOP-01, Acts of Nature, Section II, Severe Weather Entry Required.
3 +30 min		I (BOPO, CRS) TS (CRS)	Steam Generator RC-2A Level Channel LT-903Y Fails High. Feedwater Control System Automatically Shifts to Manual.
4 +45 min		I (ATCO, CRS) TS (CRS)	VCT Level Transmitter LT-219 Fails Low due to CVCS leak.
5 +55 min		C (BOPO, CRS)	Loss of 161 KV Line. Place Condensate Pump FW-2A in service.
6 +55 min		M (ATCO, BOPO, CRS)	Loss of Offsite Power. Reactor Trip.
7 +55 min		C (ATCO)	Four (4) Stuck CEAs on Reactor Trip. Emergency Boration Required Upon Power Restoration.
8 +60 min		M (ATCO, BOPO, CRS)	Diesel Generator DG-01 Breaker Failure with Diesel Generator DG-02 Overspeed Trip. Station Blackout.
9 +70 min		C (BOPO)	Diesel Driven Auxiliary Feedwater Pump FW-54 Start Failure. EOP-20, Functional Recovery Entry Required.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
2	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
2	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC 2

The crew will assume the shift at 100% power per OP-4, Load Change and Normal Power Operation. The scheduled activity is to perform OI-CS-1, Containment Spray Normal Operation, Attachment 1A, SI-3A Containment Spray Pump Operability Test.

The next event is a Severe Thunderstorm Watch from the National Weather Service requiring entry into AOP-01, Acts of Nature, Section II, Severe Weather. Once plant announcements have been made, a high failure of Steam Generator RC-2A Level Channel LT-903Y will occur. Initial operator actions are per ARP-DCS-FW, Feedwater DCS Annunciator Response Procedure, and include verifying Feedwater Control is in MANUAL and bypassing the failed input. Once completed, Feedwater Control is restored to AUTO per OI-FW-3, Steam Generator Level Control, Attachment 4, Level Controller Operation. The SRO will refer to Technical Specification LCO 2.15.3 – Steam Generator Narrow Range Level Instrument at AI-179.

The next event is a sensing line leak resulting in a low failure of Volume Control Tank (VCT) Level Transmitter LT-219. Actions are per ARP-CB-1/2/3/A2, VOLUME CONTROL TANK LEVEL HI-LO, until it is determined that a leak exists. Once identified, AOP-33, CVCS Leak, is entered. The SRO will refer to Technical Specification LCO 2.15.3 – Volume Control Tank Level Instrument at AI-185.

A lightning strike in the Fort Calhoun Switchyard will open 161 KV Breakers 110 and 111 and result in a loss of the 161 KV lines. A successful Fast Bus Transfer initially maintains power to all 4160 V Buses. The crew enters AOP-31, 161 KV Grid Malfunctions, Section II, All 4160 V Buses Fed from 22 KV. AOP-31 requires placing FW-2A, Condensate Pump in service to balance electric system loads per OI-FW-1, Condensate System Normal Operation. When AOP-31, Step 6, Matching Breaker Flags is performed, a Plant Trip will occur.

A Loss of Offsite Power occurs on the Plant Trip and initiates a failure of both Emergency Diesel Generators. When the Reactor Trip is verified, four (4) CEAs will be identified as stuck and an Emergency Boration is required but cannot be initiated due to loss of power. EOP-00, Standard Post Trip Actions, is entered and feedwater flow must be aligned to the Steam Generators using AFW-10, Steam Driven Auxiliary Feedwater (AFW) Pump via the AFW Nozzles or the Feed Ring. When Diagnostic Actions are performed in EOP-00 the crew should recognize a loss of both Reactivity Control (4 Stuck CEAs with no Emergency Boration flow) and Maintenance of Vital Auxiliaries (no energized 4160 V Safeguards Bus) and enter EOP-20, Functional Recovery. EOP-20, Resource Assessment Trees RC-1, CEA Insertion and MVA-AC, Restoration of AC are the significant Safety Functions to be addressed.

The event is complicated by a start failure of FW-54, Diesel Driven AFW Pump (normal post-trip AFW source), and requires starting and aligning of FW-10, Turbine Driven AFW Pump. The scenario is terminated in EOP-20 when power is restored to Safeguards Bus 1A3 via a replaced DG-01 Output Breaker and Emergency Boration flow is initiated to the Reactor Coolant System.

<u>Risk Significance</u>:

• Failure of risk important system prior to trip:	Loss of 161 KV Line
• Risk significant core damage sequence:	Loss of Reactivity Control Station Blackout/Loss of Feedwater
• Risk significant operator actions:	Establish Feedwater Flow Emergency Borate for 4 Stuck CEAs Restore Power to Safeguards Bus

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

RESET to IC-#2 (or any 100% MOL IC) and LOAD & EXECUTE NRC 2.sce for NRC Scenario 2.

Provide Lead Examiner key for HCV-2958 for performance of Normal evolution, Event 1.

Preset item – Event 7 – 4 Rods Stuck out on Reactor Trip

Type	Item	Value	Condition
Malfunction	ROD_PWR_229_2 ROD_PWR_B15_2 ROD_PWR_B16_2 ROD_PWR_228_2	Stuck Stuck Stuck Stuck	Scenario Event: “4 stuck rods”

Preset Item – Event 8 – Diesel Generator #1 Breaker Failure

Type	Item	Value	Condition
Malfunction	BUS_1A3_20_BKR_Trip (1AD-1 Breaker failure to Trip position)	True	Scenario Event: “DG1 Breaker Failure”

Preset Item – Event 9 – FW-54 Fails to Start Remotely

Type	Item	Value	Condition
Remote	REM:AFW_FWC04 REM:AFW_FWC02	Local Stop	Scenario Event: “FW-54 Start Failure”

Event 2 – Notification of Severe Thunderstorm Watch from National Weather Service

Type	Direction
Booth Operator	Call on the NAWAS phone by dialing 98*, wait 5-10 seconds and REPORT: “This is the National Advance Warning Alert System with an update. The National weather service in Valley, Nebraska has issued a Severe Thunderstorm Watch for Washington county Nebraska until (60 minute from current time). Current radar indicates conditions are met to produce severe thunderstorms with potentially heavy rain, high winds, and damaging hail. Individuals in the path of the storm are recommended to be attentive to weather conditions and consider moving to shelter in a sturdy structure.”

Event 3 – Steam Generator Level Transmitter LT-903X Fails High

Type	Item	Value	Condition
Transmitter	LT-903Y LT-903Y-1	100, ramp = 30 sec 100, ramp = 30 sec	When directed by examiner, trigger/activate this event. Scenario Event: “LT903X fail high”

Event 4 – VCT Level Transmitter Fails Low, VCT Leak

Type	Item	Value	Condition
Transmitter	CVC_LT219	0, ramp = 30 sec	When directed by examiner, trigger/activate this event. Scenario Event: “VCT LT-219 Fail Low” When directed as Aux Building operator to isolate leak, delete malfunction CVX07B.
Malfunction	CVX07B (VCT Leak)	2%	

Event 5 – Loss of 161 KV Line

Type	Item	Value	Condition
Malfunction	87L/161 (Relay 87L/161 trip)	True	When directed by examiner, trigger/activate this event. Scenario Event: "Loss of 161KV line"

Event 6 – Loss of Offsite Power

Type	Item	Value	Condition
Malfunction	SWD01	True	When directed by examiner, trigger/activate this event. Scenario Event: "Loss of Offsite Power"

Event 8 – Diesel Generator #2 Overspeed Trip

Type	Item	Value	Condition
Expert	H_PD2_301TL_1 REM:FDP_RCW1_1 REM:FDP_RCW1_2 REM:FDP_RCW1_5 H_PD2_311_1 H_PD2_311_1 DGAQRL112x2TVSP DGAQRL112X1TVSP	1 2 2 2 1200, ramp = 10 sec 0, Delay=11, ramp = 3 0 0	Event is triggered automatically 10 minutes after reactor trip. Scenario Event: "DG2 overspeed trip"

Scenario Event Description NRC Scenario 2

<u>Booth Operator</u>: INITIALIZE to IC-1 and LOAD NRC 2.sce. **ENSURE all Simulator Annunciator Alarms are ACTIVE.** **ENSURE all Control Board Tags are removed.** **ENSURE Condensate Pumps FW-2B and FW-2C in service.** **ENSURE Synchroscope Switch in a location other than DG-01 Breaker 1AD1.** **ENSURE Reactivity Briefing Sheet printout provided with Turnover.** **ENSURE Middle-of-Life Thumb Rule Sheet provided with Turnover.** **ENSURE procedures in progress provided to crew in Briefing Room:** **- COPY of OI-CS-1, Containment Spray Normal Operation, Attachment 1A, SI-3A Containment Spray Pump Operability Test, INITIALED through Prerequisites. Provide Key for HCV-2958.**
<u>Control Room Annunciators in Alarm</u>: **NONE**

<u>Procedure List</u>
Event 1: OI-CS-1, Containment Spray Normal Operation, Attachment 1A, SI-3A Containment Spray Pump Operability Test
Event 2: AOP-01, Acts of Nature, Section II, Severe Weather
Event 3: ARP-DCS-FW, Feedwater DCS Annunciator Response Procedure
Event 3: OI-FW-3, Steam Generator Level Control, Attachment 4, Level Controller Operation
Event 4: ARP-CB-1/2/3/A2, Window B-2U, VOLUME CONTROL TANK LEVEL HI-LO
Event 4: AOP-33, CVCS Leak
Event 5: AOP-31, 161 KV Grid Malfunctions, Section II, All 4160 V Buses Fed from 22 KV
Event 5: OI-FW-1, Condensate System Normal Operation, Attachment 4, Rotating Condensate Pumps
Event 6: EOP-00, Standard Post Trip Actions
Event 9: EOP-20, Functional Recovery
Event 9: EOP-20, Resource Assessment Trees RC-1, CEA Insertion
Event 9: EOP-20, Resource Assessment Trees MVA-AC, Restoration of AC

Operating Test :	NRC	Scenario #	2	Event #	1	Page	6 of 33
Event Description:	Containment Spray Pump Operability Test						

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, RESPOND to requests from Control Room.**		
Indications Available: **NONE**		
Examiner Note: **The following steps are from OI-CS-1, Containment Spray Normal Operation, Attachment 1A, SI-3A Containment Spray Pump Operability Test.**		
Booth Operator: **If requested, report Containment Spray Pump SI-3A is ready to start and all conditions locally are normal.**		
	ATCO	PLACE the following switches to TEST. [Step 1]
		• CNTMT Spray Valve HCV-344 Test Switch HC-344/Test.
		• CNTMT Spray Valve HCV-345 Test Switch HC-345/Test.
	ATCO	VERIFY the following annunciators are in ALARM: [Step 2]
		• HCV-344/345 SET SPRAY PUMPS TEST PERMIT at AI-30A, A33-1, Window H-5.
		• HCV-344/345 SET SPRAY PUMPS TEST PERMIT at AI-30B, A34-1, Window H-3.
	ATCO	RUN SI-3A by completing the following: [Step 3]
	CRS	• REVIEW Technical Specification LCO 2.4 requirements and LOG into the appropriate T.S. LCOs 2.4(2)c for HCV-345). [Step 3.a]
	ATCO	• OVERRIDE and CLOSE HCV-345: [Step 3.b]
		• PLACE HC-345, Containment Spray Valve Control Switch, to OVERRIDE
		• VERIFY annunciator SPRAY VALVE HCV-345 HEADER ISOLATED at AI-30B, A34-1, Window H-2 is in ALARM. [Step 3.c]
		Examiner Note: Provide Key for HCV-2958 controls to candidate. • CLOSE HCV-2958, Containment Spray Pump SI-3A Discharge at AI123A. [Step 3.d]
	ATCO	VERIFY annunciator SI PUMPS VALVES OFF NORMAL at AI-30A, A33-1, Window J-1 is in ALARM. [Step 4]

Operating Test : NRC	Scenario # 2	Event # 1	Page 7 of 33
Event Description: Containment Spray Pump Operability Test			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When contacted, REPORT SI-138 is OPEN and BACKSEATED.		
	ATCO	Direct operator to ENSURE SI-138, Containment Spray Pump SI-3A Minimum Recirc Isolation Valve is OPEN and BACKSEATED. [Step 5]
	ATCO	DETERMINE the following valves are OPEN: [Step 6]
		• HCV-385, SIRWT Tank Recirculation Valve
		• HCV-386, SIRWT Tank Recirculation Valve
	ATCO	START SI-3A, CNTMT Spray Pump. [Step 7]
Examiner Input: When timing is started, REPORT "time compress" and that 5 minutes has elapsed.		
	ATCO	When five minutes has elapsed, STOP SI-3A, CNTMT Spray Pump. [Step 8]
	ATCO	OPEN HCV-2958, Containment Spray Pump SI-3A Discharge ay AI-128A. [Step 9]
	ATCO	VERIFY annunciator SI-3A, SI PUMPS VALVES OFF NORMAL at AI-30A, A33-1, Window J-1) is CLEAR. [Step 10]
	ATCO	PLACE HC-345, Containment Spray Valve HCV-345 Control Switch, to AUTO. [Step 11]
	ATCO	VERIFY annunciator SPRAY VALVE HCV-345 HEADER ISOLATED at AI-30B, A34-1, Window H-2 is CLEAR. [Step 12]
	ATCO	PLACE the following switches to OFF: [Step 13]
		• CNTMT Spray Vlv HCV-344 Test Switch HC-344/Test
		• CNTMT Spray Vlv HCV-345 Test Switch HC-345/Test

Operating Test :	NRC	Scenario #	2	Event #	1	Page	8	of	33
Event Description:	Containment Spray Pump Operability Test								

Time	Position	Applicant's Actions or Behavior
	ATCO	VERIFY the following annunciators are CLEAR: [Step 14]
		• HCV-344/345 SET SPRAY PUMPS TEST PERMIT at AI-30A, A33-1, Window H-5
		• HCV-344/345 SET SPRAY PUMPS TEST PERMIT at AI-30B, A34-1, Window H-3
	CRS	LOG out of Technical Specification 2.4 LCOs. [Step 15]
Booth Operator: When contacted, REPORT signs are removed.		
	ATCO	CONTACT Auxiliary Operator to remove Protective Equipment Signs. [Step 16]
When CRS has logged out of Technical Specifications, PROCEED to Event 2.		

Operating Test : NRC	Scenario # 2	Event # 2	Page 9 of 33
Event Description: Severe Thunderstorm Watch from the National Weather Service			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, REPORT Event 2. - Severe Thunderstorm Watch from the National Weather Service		
Indications Available: **NONE**		
Booth Operator: CONTACT Control Room on NAWAS phone READ prepared message. Call on the NAWAS phone by dialing 98*, wait 5-10 seconds and **REPORT**: "This is the National Advance Warning Alert System with an update. The National weather service in Valley, Nebraska has issued a Severe Thunderstorm Watch for Washington county Nebraska until (60 minute from current time). Current radar indicates conditions are met to produce severe thunderstorms with potentially heavy rain, high winds, and damaging hail. Individuals in the path of the storm are recommended to be attentive to weather conditions and consider moving to shelter in a sturdy structure."		
	CRS	REFER to AOP-01, Acts of Nature, Section II, Severe Weather.
Examiner Note: The following steps are from AOP-01, Acts of Nature, Section II, Severe Weather.		
NOTE The Shift Manager and Station Duty Manager should discuss the potential for wind-generated missiles and the necessity to restore any Engineered Safeguards-Equipment that may be out of service.		
	CRS	NOTIFY Manager-Shift Operations and Station Duty Manager of weather conditions. [Step 4.1]
	CRS	If weather conditions allow, PERFORM a visual inspection of the Protected Area and Switchyard per SO-G-119, Site Wind Generated Missile Protection Standards. [Step 4.2]
NOTES 1. Guidance for announcements for the Administration Building and Training Center are located in EPIP-OSC-15, Communicator Actions. 2. Steps 3 through 6 can be performed in order and as needed as required by weather conditions.		
	CRS	If a severe thunderstorm watch exists, PERFORM the following: [Step 4.3]

Operating Test : NRC	Scenario # 2	Event # 2	Page 10 of 33
Event Description: Severe Thunderstorm Watch from the National Weather Service			

Time	Position	Applicant's Actions or Behavior
		• MONITOR NAWAS to determine changes in weather conditions. [Step 4.3.a]
		• ANNOUNCE and REPEAT the following over the plant communications system: [Step 4.3.b]
		• "Attention all personnel. Attention all personnel. A severe thunderstorm watch has been issued for area surrounding the plant until 10 PM tonight."
When Plant announcement has been made, PROCEED to Event 3.		

Operating Test : NRC	Scenario # 2	Event # 3	Page 11 of 33
Event Description: Steam Generator Level Channel Failure			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 3. - Steam Generator RC-2A Level Channel LT-903Y fails high.		
Indications Available: Feedwater Digital Control System Alarm		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS Steam Generator RC-2A Level Transmitter LT-903Y failed high.
	CRS	DIRECT actions of ARP-DCS-FW, LT-903Y.
Examiner Note: The following steps are from ARP-DCS-FW, Feedwater Digital Control System.		
	BOPO	DETERMINE failure is NOT from a Feedwater Flow, Steam Flow, or Steam Pressure Instrument. [Step 1]
	BOPO	PERFORM the following for Level Instrument LT-903Y failure: [Step 2]
		• VERIFY FWCS IN MANUAL is displayed on SECONDARY Feedwater Regulating System display. [Step 2.1]
		• TOUCH display with the BAD process. A 'B' will be displayed beside the level indication. [Step 2.2]
		• DETERMINE BAD input NOT automatically bypassed. [Step 2.3]
		• TOUCH Bypass on verification faceplate to BYPASS BAD input. [Step 2.3.1]
		• VERIFY point displays GOOD status and 'B' is no longer displayed. [Step 2.3.2]
		• When Steam Generator level is in band, RETURN Level Controller, LC0903_1E back to AUTO per OI-FW-3. [Step 2.4]
Examiner Note: The following steps are from OI-FW-3, Steam Generator Level Control, Attachment 4, Level Controller Operation, Step 5.		
	BOPO	PERFORM the following to return Level Controller LC0903_E1 (DCS) to AUTOMATIC control: [Step 5]

Operating Test :	NRC	Scenario #	2	Event #	3	Page	12	of	33
Event Description:	Steam Generator Level Channel Failure								

Time	Position	Applicant's Actions or Behavior
		• SELECT Level Controller LC0903_1E (DCS). [Step 5.a]
		• PERFORM one of the following to DISPLAY controller: [Step 5.a]
		• TOUCH FWTR CTRL-2A pushbutton from Feedwater Regulating System display, **OR**
		• TOUCH Feedwater Level Control Button from the LVLS display.
		• PERFORM the following for Steam Generator RC-2A: [Step 5.b]
		• TOUCH AUTO on LC0903_1E (DCS) Level Controller and VERIFY the 'T' is displayed. [Step 5.b.1)]
		• DETERMINE FC1101, S/G RC-2A FW REG VLV (DCS) is in AUTO. [Step 5.b.2)]
		• DETERMINE HC1105 is in AUTO. [Step 5.b.3)]
		• VERIFY Feed Regulating System return to 3 ELEMENT AUTO. [Step 5.b.4)]
Examiner Note: **The following steps continue from ARP-DCS-FW.**		
	CRS	DETERMINE other Steam Generator level instruments NOT affected. [Step 2]
	CRS	DETERMINE BAD input bypassed MANUALLY. [Step 3]
	BOPO	MONITOR Steam Generator levels. [Step 4]
	CRS	VERIFY XC-105, Secondary Calorimetric, is valid. [Step 5] **Examiner Note: This step is normally performed by the STA – The CRS may not address XC-105 at this time.**
Booth Operator: **When contacted, REPORT Level Transmitter LT-903Y-1 is failed high at AI-179.**		
	CRS	DETERMINE LT-903 is cause of alarm and CONTACT Auxiliary Operator to VERIFY level indication at AI-179. [Step 6]
	CRS	NOTIFY Work Week Manager of LT-903Y malfunction. [Step 7]

Operating Test : NRC	Scenario # 2	Event # 3	Page 13 of 33
Event Description: Steam Generator Level Channel Failure			

Time	Position	Applicant's Actions or Behavior
	CRS	EVALUATE Technical Specification LCO 2.15.3, Alternate Shutdown and Auxiliary Feedwater Panel
		• LCO 2.15.3.(1) – Steam Generator Narrow Range Level Instrument at AI-179 (Table 2.6 / Item #3.d)
		• CONDITION 2.15.3.(1) – One Steam Generator Narrow Range Level Transmitter inoperable • ACTION 2.15.3.(1) – RESTORE the required channel to OPERABLE status within seven (7) days.
When Technical Specifications have been addressed, PROCEED to Event 4.		

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 4. - VCT Level Transmitter leak on LT-219 line.		
Indications Available: CB-1,2,3/A2 – VOLUME CONTROL TANK LEVEL HI-LO VCT level indication LIC-219 slowly lowering		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
	ATCO	INFORM CRS of VCT Level Transmitter LT-219 slowly lowering.
	CRS	REFER to ARP-CB-1,2,3/A2, Window B-2U – VOLUME CONTROL TANK LEVEL HI-LO.
Examiner Note: The following steps are from ARP-CB-1.2.3/A2, Window B-2U – VOLUME CONTROL TANK LEVEL HI-LO.		
	ATCO	DETERMINE VCT Level Indication on LIC-219 NOT between 51.7% and 91.2%. [Step 1]
Booth Operator: When contacted, WAIT 2 minutes then REPORT indications of leakage from the VCT Level Transmitter.		
	ATCO	If level is low, PERFORM the following: [Step 2]
		• ALIGN LCV-218-1, VCT Inlet Valve is aligned to VCT. [Step 2.1]
		• DETERMINE Pressurizer level is at program. [Step 2.2]
		• DETERMINE makeup to VCT NOT required. [Step 2.3]
		• CONTACT Auxiliary Operator to check CVCS System for leakage. [Step 2.4]
	ATCO	DETERMINE VCT level is NOT high. [Step 3]
Booth Operator: When contacted, WAIT one minute and REPORT VCT level indication at AI-185 indicates 0%.		

Operating Test : NRC	Scenario # 2	Event # 4	Page 15 of 33
Event Description: VCT Level Transmitter Leak			

Time	Position	Applicant's Actions or Behavior
	ATCO	DETERMINE VCT level indication is due to a leak. [Step 4]
		• CONTACT Auxiliary Operator to verify level indication at AI-185, Alternate Shutdown Panel. [Step 4.1]
		• If low level is due to a system leak, IMPLEMENT AOP-33, CVCS Leak. [Step 4.1.1]
	CRS	EVALUATE Technical Specification LCO 2.15.3, Alternate Shutdown and Auxiliary Feedwater Panel
		• LCO 2.15.3.(1) – Volume Control Tank Level Instrument at AI-185 (Table 2.6 / Item #4.b)
		• CONDITION 2.15.3.(1) – One Volume Control Tank Level Instrument inoperable • ACTION 2.15.3.(1) – RESTORE the required channel to OPERABLE status within seven (7) days.
Examiner Note: **The following steps are from AOP-33, CVCS Leak.**		
	CRS	PERFORM the following to isolate CVCS: [Step 4.1]
	ATCO	• CLOSE both Letdown Isolation Valves. [Step 4.1.a]
		• CLOSE TCV-202.
		• CLOSE HCV 204.
Examiner Note: **With all 3 Charging Pumps in PULL-TO-LOCK, Technical Specification LCO 2.2.4 would be temporarily entered until a Charging Pump is restarted later in the AOP. This is an identified Procedural Enhancement Opportunity.**		
	ATCO	• PLACE Charging Pump Control Switches in PULL-TO-LOCK. [Step 4.1.b]
		• PLACE CH-1A in PULL-TO-LOCK
		• PLACE CH-1B in PULL-TO-LOCK
		• PLACE CH-1C in PULL-TO-LOCK
	ATCO	• ENSURE the following valves are CLOSED: [Step 4.1.c]
		• CLOSE HCV-238, Loop 1 Charging Isolation.
		• CLOSE HCV-239, Loop 2 Charging Isolation.
		• VERIFY HCV-240, PZR Auxiliary Spray Isolation Valve CLOSED.

Operating Test :	NRC	Scenario #	2	Event #	4	Page	16	of	33
Event Description:	VCT Level Transmitter Leak								

Time	Position	Applicant's Actions or Behavior
		• VERIFY HCV-249, PZR Auxiliary Spray Isolation Valve CLOSED.
	CRS	IMPLEMENT the Emergency Plan. [Step 4.2]
Booth Operator:		**REPORT as Auxiliary Operator that indications of leakage are from the VCT level transmitter line. REPORT as Chemistry that no sampling is in progress.**
Booth Operator:		**If directed to isolate the level transmitter, EXECUTE remote function to LOCALLY CLOSE CH-227 and CH-206 which isolates LT-218 and LT-219.**
	CRS	PERFORM the following to locate the leak: [Step 4.3]
	AO	• VISUALLY inspect CVCS system piping. [Step 4.3.a]
		• CHECK all the following levels: [Step 4.3.b]
		• DETERMINE Spent Regen Tank level normal.
		• DETERMINE Aux Building Sump Tank RISING.
		• DETERMINE Containment Sump level normal.
	CRS	• DIRECT Chemistry to isolate all CVCS sample lineups. [Step 4.3.c]
		NOTE If leak is contained by the actions in Step 1, PZR level will lower at a rate of approximately 1% every 12 minutes due to Reactor Coolant Pump Bleedoff flow.
	CRS	DETERMINE Pressurizer level NOT lowering abnormally. [Step 4.4]
		NOTE VCT level will tend to rise approximately 1% every 6 minutes due to Reactor Coolant Pump Bleedoff flow.
	ATCO	DETERMINE VCT level is lowering and CLOSE LCV-218-2, VCT Outlet Valve. [Step 4.5]
	ATCO	DETERMINE VCT level continues to lower and ISOLATE the VCT: [Step 4.6]
		• PLACE LCV-218-1, VCT Inlet Valve, in RWTS. [Step 4.6.a]

Operating Test : NRC	Scenario # 2	Event # 4	Page 17 of 33
Event Description: VCT Level Transmitter Leak			

Time	Position	Applicant's Actions or Behavior
		• ENSURE HCV-208, RCP Bleedoff to RCDT Isolation Valve, is open. [Step 4.6.b]
		• CLOSE all of the following valves: [Step 4.6.c]
		• HCV-241, RCP Bleedoff to VCT Isolation Valve
		• HCV-206, RCP Bleedoff to VCT Isolation Valve
Booth Operator: REPORT as Auxiliary Operator that SL-130 and SL-135 are CLOSED.		
		• SL-130, SAMPLE RETURN TO VOLUME CONTROL TANK CH-14 ISOLATION VALVE in Room 60.
		• SL-135, VCT CH-14 RCS SAMPLE RETURN ISOLATION VALVE in Room 60.
Booth Operator: CONTACT as Shift Manager and PERFORM RCS Makeup from the SIRWT.		
	CRS	PERFORM Step a or b to MAINTAIN PZR level 45-60%: [Step 4.7]
		• COMMENCE RCS makeup at existing boron concentration per Attachment A, Blended Makeup to the Charging Pump Suction Header. [Step 4.7.a]
		• PERFORM the following and COMMENCE RCS makeup from SIRWT: [Step 4.7.b]
		• OPEN LCV-218-3, Charging Pump Suction SIRWT Isolation Valve. [Step 4.7.b.1)]
		• OPEN both Charging Isolation Valves: [Step 4.7.b.2)]
		• OPEN HCV-238.
		• OPEN HCV-239.
		• START at least one Charging Pump. [Step 4.7.b.3)]
	CRS	EVALUATE need to implement AOP-09, High Radioactivity. [Step 4.8]
When Boron addition via SIRWT is commenced, PROCEED to Event 5.		

Operating Test : NRC	Scenario # 2	Event # 5	Page 18 of 33
Event Description: Loss of 161 KV Line			

Time	Position	Applicant's Actions or Behavior
Booth Operator:		**When directed, EXECUTE Event 5.** **- Loss of 161 KV Line.**
Indications Available:		**CB-20/A15 – BREAKER 111 TRIPPED** **CB-20/A15 – 161 KV SUPPLY BKR LOCKOUT RELAY OPERATED** **CB-20/A15 – BREAKER 110 TRIPPED** **CB-20/A15 – PLANT 161 KV LINE LOW FREQUENCY** **CB-20/A17 – TRANS T1A-3 SECONDARY LOW VOLTAGE** **CB-20/A17 – BKR 1A33 AUTO TRIP** **CB-20/A17 – TRANS TIA-3 LOCKOUT RELAY OPERATED 86/TIA-3** **CB-20/A18 – TRANS T1A-4 SECONDARY LOW VOLTAGE** **CB-20/A18 – BKR 1A44 AUTO TRIP** **CB-20/A18 – TRANS TIA-4 LOCKOUT RELAY OPERATED 86/TIA-4** **Supply Breakers 110 AND 111 white trip lights lit**
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS of loss of 161 KV line.
	CRS	REFER to AOP-31, 161 KV Grid Malfunctions, Section II, All 4160 V Buses Fed from 22 KV.
Booth Operator:		**If contacted, REPORT as T&D System Operation that cause of 161 KV line loss appears to be lightning strike in Fort Calhoun Station Switchyard. If requested, report repair teams are being dispatched.**
Examiner Note:		**The following steps are from AOP-31, 161 KV Grid Malfunctions, Section II, All 4160 V Buses Fed from 22 KV.**
		CAUTION To protect Bus 1A1 in the event of a fault, FW-2A and FW-4A should not both be left running when the Feedwater System is realigned.
	CRS	DETERMINE Reactor power is ≥ 50% and ENSURE all the following conditions are satisfied: [Step 4.1]
	BOPO	• DETERMINE two Condensate Pumps are RUNNING. [Step 4.1.a]
		• DETERMINE two Feedwater Pumps are RUNNING. [Step 4.1.b]
		• DETERMINE two Heater Drain Pumps are RUNNING. [Step 4.1.c]

Operating Test :	NRC	Scenario #	2	Event #	5	Page	19	of	33
Event Description:	Loss of 161 KV Line								

Time	Position	Applicant's Actions or Behavior
	BOPO	ADJUST Main Generator terminal voltage to less than 22,000 Volts. [Step 4.2]
		• NOTIFY Energy Marketing of the need to adjust voltage. [Step 4.2.a]
		• ADJUST Voltage Regulator per OI-ST-1, Turbine Generator Normal Operation. [Step 4.2.b]
		• VERIFY terminal voltage is less than 22,000 volts. [Step 4.2.c]
	CRS	ESTABLISH balanced 4160 V Bus loading on T1A1 and T1A2 by ensuring ALL of the following pumps on Bus 1A1 are operating: [Step 4.3]
	BOPO	• DETERMINE FW-2A, CONDENSATE Pump is NOT running and REFER to OI-FW-1 Condensate System Normal Operation. [Step 4.3.a]
		• DETERMINE FW-4A, Main Feedwater Pump is RUNNING. [Step 4.3.b]
		• DETERMINE FW-5A, Heater Drain Pump is RUNNING. [Step 4.3.c]

<u>Examiner Note</u>: The following steps are from OI-ST-1, Turbine Generator Normal Operation, Attachment 6, Generator VAR Adjustments (Automatic Mode).

<u>NOTE</u>

Lowering voltage of the Main Generator while synchronized to the grid may cause low voltage indications on T1A1 and T1A2.

Time	Position	Applicant's Actions or Behavior
	BOPO	PERFORM the following to lower Generator Reactive Load (VARS): [Step 1]
		• ROTATE Generator G1 AC Regulator Voltage Adjuster (90P) in the COUNTER-CLOCKWISE direction until desired load is attained, as indicated on VAR/G1. [Step 1.a]
		• PLACE CS-70E/G1F, Generator G1 DC Regulator Voltage Adjuster (70P), in the LOWER position until the V/G1R, Generator ST-2 Voltage Regulator Transfer Voltage, reads 0 Volts DC. [Step 1.b]

<u>Examiner Note</u>: The following steps are from OI-FW-1, Condensate System Normal Operation, Attachment 4, Rotating Condensate Pumps.

<u>NOTE</u>

FW-2B or FW-2C are the preferred pumps during one pump operation because of their ability to continue to run after a SIAS or CSAS actuation.

Operating Test :	NRC	Scenario #	2	Event #	5	Page	20	of 33
Event Description:	Loss of 161 KV Line							

Time	Position	Applicant's Actions or Behavior
	BOPO	SELECT Condensate Pump to be started: [Step 1]
		• FW-2A
	STA	SUSPEND GARDEL data feed per OI-ERFCS-2, Attachment 6. [Step 2]

CAUTIONS

1. The 43/FW Switch affects the Auto-Start operation of the Main Condensate Pumps, Main Feedwater Pumps and Heater Drain Pumps.
2. The Standby (Auto-Start) feature for these pumps will be inhibited when the 43/FW Switch is placed in off.
3. The 43/FW Switch must be in off to start a Condensate Pump.

Time	Position	Applicant's Actions or Behavior
	BOPO	PERFORM the following at CB-10/11: [Step 3]
		• PLACE Cond. & FW Pumps Transfer Switch 43/FW to OFF. [Step 3.a]
		• VERIFY annunciator CB-10/11/A10, Window B-6L – 43/FW TRANSFER SWITCH OFF-AUTO in alarm. [Step 3.b]

NOTE

During rotation of the Condensate Pumps, at the time the designated standby pump is started, declare XC105 INVALID and log in the Control Room Log. Once pump rotation is complete, and the 12-minute validity period has passed (Ref. OI-ERFCS-3), the STA should review all XC105 input parameters and determine they are at steady-state, then XC105 can be declared valid and available for monitoring reactor core output.

Time	Position	Applicant's Actions or Behavior
	BOPO	START Condensate Pump FW-2A. [Step 4]
	BOPO	VERIFY FW-2A ammeter returns to < 250 amps in < 15 seconds and STABILIZES on CB-10/11. [Step 5]

<u>Booth Operator</u>: When contacted, REPORT FW-2A discharge pressure of ~520 psig.

Time	Position	Applicant's Actions or Behavior
	BOPO	ENSURE Condensate Pump minimum flow is being maintained: [Step 6]
		• VERIFY discharge pressure of 490-600 psig at FW-2A. [Step 6.a]

<u>Booth Operator</u>: When contacted, REPORT all FW-2A parameters are normal.

Operating Test : NRC	Scenario # 2	Event # 5	Page 21 of 33
Event Description: Loss of 161 KV Line			

Time	Position	Applicant's Actions or Behavior
	BOPO	MONITOR Condensate Pump FW-2A parameters: [Step 7]
		• CHECK for unusual noise or vibration. [Step 7.a]
		• VERIFY lube oil levels in middle of sightglass. [Step 7.b]
		• CHECK PI-1214, Seal Water inlet pressure between 70 and 90 psig. [Step 7.c]
		• CHECK PI-1232A/B/C, FW-2A/B/C Discharge Pressure at 490-600 psig at pump. [Step 7.d]
	BOPO	• CHECK PI-1181A/B/C, FW-2A/B/C Discharge Pressure at 490-600 psig on CB-10/11. [Step 7.e]
	BOPO	• CHECK flow and temperatures on ERF Computer: [Step 7.f]
		• F1172, PRINT XC092 FOR COND. FLOW • T1179A, COND PMP A DISCH HDR TEMP • T1179B, COND PMP B DISCH HDR TEMP • T1184A/B/C, COND PMP A/B/C MTR OUT BRG TEMP • T1185A/B/C, COND PMP A/B/C MTR IN BRG TEMP

NOTE

Condensate Pump Control Switch should be positioned in AFTER-STOP or PULL-STOP per the Shift Manager or CRS, to prevent possible water hammer at power levels below 50%.

	BOPO	STOP Condensate Pump FW-2C. [Step 8]
	BOPO	PLACE FW-2C Condensate Pump Control Switch in AFTER-STOP. [Step 9]
	BOPO	VERIFY FW-2C Condensate Pump ammeter drops to 0. [Step 10]

Booth Operator: When contacted, REPORT no reverse rotation on FW-2C.

	BOPO	VERIFY FW-2C NOT rotating in reverse direction. [Step 11]
	BOPO	DETERMINE 43-SIAS/FW2, Post-SIAS/CSAS Running Condensate Pump Switch in FW-2B position. [Step 12]
	BOPO	PERFORM the following at CB-10/11: [Step 13]
		• PLACE 43/FW Switch in AUTO. [Step 13.a]

Operating Test : NRC	Scenario # 2	Event # 5	Page 22 of 33
Event Description: Loss of 161 KV Line			

Time	Position	Applicant's Actions or Behavior
		• VERIFY annunciator CB-10/11/A10, Window B-6L – 43/FW TRANSFER SWITCH OFF-AUTO is clear. [Step 13.b]
	STA	RESTORE GARDEL data feed per OI-ERFCS-2, Attachment 7. [Step 14]
Examiner Note: The following steps continue from AOP-31, Section II.		
	BOPO	DETERMINE all 480 V Buses greater than 430 volts: [Step 4.4]
		• OBSERVE Bus 1B3A voltage at ~470 V. • OBSERVE Bus 1B3B voltage at ~470 V. • OBSERVE Bus 1B3C voltage at ~470 V. • OBSERVE Bus 1B4A voltage at ~470 V. • OBSERVE Bus 1B4B voltage at ~460 V. • OBSERVE Bus 1B4C voltage at ~470 V.
	CRS	NOTIFY NRC Operation Center within 4 hours of loss of 161 KV Line. [Step 4.5]
Examiner Note: The Loss of Offsite Power event is triggered 30 seconds after the flag is matched on Breaker 1A44.		
	BOPO	MATCH flags on all the following breakers: [Step 4.6]
		• Breaker 110 flag MATCHED. • Breaker 111 flag MATCHED. • Breaker 1A31 flag already matched. • Breaker 1A33 flag MATCHED. • Breaker 1A42 flag already matched. • Breaker 1A44 flag MATCHED.
When Breaker 1A44 flag is MATCHED, PROCEED to Events 6, 7, 8, and 9.		

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	23	of 33

Event Description: Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>:		**When the control switch is matched for Breaker 1A44, Events 6, 7, 8, and 9 will automatically execute. - Loss of Offsite Power. - Four Stuck CEAs on Reactor Trip. - Diesel Generator DG-01 Output Breaker failure. - Diesel Generator DG-02 overspeed trip. - Diesel Driven Auxiliary Feedwater Pump FW-54 start failure.**
<u>Indications Available</u>: **Numerous Reactor Trip and Loss of Offsite Power Alarms.**		
	CREW	RECOGNIZE Reactor Trip due to Loss of Offsite Power.
	CRS	DIRECT performance of EOP-00, Standard Post Trip Actions.
<u>Examiner Note</u>:		**The following steps are from EOP-00, Standard Post Trip Actions.**
	ATCO	VERIFY Reactivity Control: [Step 1]
		• VERIFY ALL of the following: [Step 1.a]
		• DETERMINE more than one Regulating or Shutdown CEA NOT inserted.
		• VERIFY Reactor Power is LOWERING.
		• VERIFY Startup Rate is NEGATIVE.
		• DETERMINE uncontrolled RCS Cooldown NOT in progress. [Step 1.b]
<u>Examiner Note</u>:		**Applicant will be unable to Emergency Borate until power is restored in EOP-20, Functional Recovery, due to the Station Blackout (SBO).**
		• [CA] If more than one CEA is NOT fully inserted, PERFORM the following to initiate Emergency Boration: [Step 1.2]
		• [CA] ENSURE both following valves CLOSED: [Step 1.2.a]
		• [CA] FCV-269X, Demin Water Makeup Valve.
		• [CA] FCV-269Y, Boric Acid Makeup Valve.
		• [CA] OPEN all the following valves: [Step 1.2.b]

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	24	of	33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
		• [CA] HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve.
		• [CA] HCV-265, CH-11A Gravity Feed Valve.
		• [CA] HCV-258, CH-11B Gravity Feed Valve.
		• [CA] START all the following pumps: [Step 1.2.c]
		• [CA] Boric Acid Pump CH-4A.
		• [CA] Boric Acid Pump CH-4B.
		• [CA] Charging Pump CH-1A.
		• [CA] Charging Pump CH-1B.
		• [CA] Charging Pump CH-1C.
		• [CA] CLOSE LCV-218-2, VCT Outlet Valve. [Step 1.2.d]
		• [CA] ENSURE the following valves CLOSED: [Step 1.2.e]
		• [CA] LCV-218-3, Charging Pump Suction SIRWT Isolation Valve
		• [CA] HCV-257, CH-4B Recirc Valve.
		• [CA] HCV-264, CH-4A Recirc Valve.
		• [CA] BORATE until adequate Shutdown Margin is established. [Step 1.2.f]
	CRS	DETERMINE Reactivity Control criteria NOT SATISFIED.
	BOPO	VERIFY Turbine Trip: [Step 2]
		• VERIFY HP & LP Stop and Intercept Valves CLOSED.
	BOPO	ENSURE all Generator Breakers are tripped: [Step 3]
		• DETERMINE Generator Output Breaker 3451-4 tripped.
		• DETERMINE Generator Output Breaker 3451-5 tripped.
		• DETERMINE Generator Field Breaker 41E/G1F tripped.
	BOPO	DETERMINE <u>both</u> 4160 V Safeguards Buses 1A3 & 1A4 are DEENERGIZED. [Step 4]

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	25	of	33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
		• [CA] PERFORM the following with either Bus 1A3 or Bus 1A4 DEENERGIZED. [Step 4.1]
Booth Operator:		**When contacted, Wait 5 minutes, then REPORT minimizing DC loads in progress.**
Booth Operator:		**When contacted about condition of Diesel Generators, WAIT 1 minute and REPORT DG-01 Output Breaker overcurrent relays are TRIPPED. Electrical Maintenance on station investigating breaker replacement.**
Booth Operator:		**When contacted about condition of Diesel Generators, WAIT 1 minute and REPORT DG-02 tripped with oil vapor in the room.**
Booth Operator:		**When contacted, EXECUTE local actions to align emergency boration by manually opening HCV-268 and manually closing LCV-218-3, and local alignment of potable water cooling to an air compressor.**
		• [CA] Minimize DC Loads within 15 minutes of loss of bus per MVA-24, Minimizing DC Loads. [Step 4.1.a]
		• [CA] DEPRESS Diesel Generator EMERGENCY START pushbuttons. [Step 4.1.b]
	CRS	DETERMINE Maintenance of Vital Auxiliaries criteria NOT SATISFIED.
	BOPO	DETERMINE Safety Injection Actuation Signal has NOT occurred and DG-01 is RUNNING and DG-02 is STOPPED. [Step 5]
	BOPO	DETERMINE 4160 V Non-Safeguards Buses 1A1 and 1A2 are DEENERGIZED. [Step 6]
	BOPO	DETERMINE 125 VDC Buses 1 and 2 are ENERGIZED. [Step 7]
	BOPO	VERIFY Instrument Air is AVAILABLE: [Step 8]
		• DETERMINE Instrument Air pressure < 90 psig.
		• DETERMINE Instrument Air Compressors NOT RUNNING due to loss of power.
		• [CA] If Instrument Air pressure is < 90 psig, PERFORM the following to restore Instrument Air: [Step 8.1]

Operating Test : NRC	Scenario # 2	Event # 6, 7, 8, & 9	Page 26 of 33
Event Description: Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure			

Time	Position	Applicant's Actions or Behavior
	BOPO	• [CA] START a Bearing Water Pump.
	BOPO	• [CA] START an Air Compressor.
	ATCO	VERIFY Component Cooling Water System operation NORMAL: [Step 9]
		• DETERMINE NO CCW Pumps RUNNING. [Step 9.a]
		• DETERMINE CCW Pump discharge pressure < 60 psig. [Step 9.b]
		• [CA] DETERMINE all Reactor Coolant Pumps STOPPED. [Step 9.1]
		• DETERMINE HCV-438A/B/C/D, CCW to RCP Coolers OPEN. [Step 9.c]
		• DETERMINE NO Raw Water Pumps RUNNING. [Step 9.d]
	CRS	VERIFY RCS Inventory Control criteria satisfied: [Step 10]
	ATCO	• DETERMINE PZR level between 30% and 70% and TRENDING to between 45% and 60%.
		• DETERMINE RCS subcooling > 20°F.
	CRS	DETERMINE RCS Inventory Control criteria SATISFIED.
	CRS	VERIFY RCS Pressure Control criteria satisfied: [Step 11]
	ATCO	• DETERMINE RCS pressure between 1800 psia and 2300 psia.
		• DETERMINE RCS pressure TRENDING between 2050 psia and 2150 psia.
		• DETERMINE PORVs are CLOSED.
	CRS	DETERMINE RCS Pressure Control criteria SATISFIED.
	CRS	VERIFY Core Heat Removal criteria satisfied: [Step 12]
	ATCO	• DETERMINE RCP NPSH requirements met.
		• DETERMINE all RCPs STOPPED.
	BOPO	• [CA] PLACE TCV-909, Temperature Controller in MANUAL on DCS. [Step 12.2.a]

Time	Position	Applicant's Actions or Behavior
	BOPO	• [CA] ENSURE TCV-909, Temperature Controller OUTPUT is zero (0). [Step 12.2.b]
	CRS	• [CA] VERIFY Natural Circulation in at least one Loop. [Step 12.2.c]
		• [CA] DETERMINE Core $\Delta T \le 50°F$.
		• [CA] DETERMINE difference between CETs and RCS T_{HOT} is ≤ 10°F on ERF "CHR" display.
		• [CA] DETERMINE RCS subcooling is ≥ 20°F.
		• [CA] DETERMINE T_{HOT} and T_{COLD} are stable or lowering.
	CRS	DETERMINE Core Heat Removal criteria NOT SATISFIED.

NOTE

If Instrument Air to valves HCV-1105, HCV-1106, HCV-1107A/B and HCV-1108A/B is not available, throttling of these valves is not possible. Open or close operation of these valves is possible for a minimum of three cycles.

	CRS	VERIFY RCS Heat Removal criteria satisfied:

<u>**Examiner Note**</u>: **Preferred method to feed SGs during an SBO is via FW-54 (aligned through the SG Feed Ring). Applicant must recognize that FW-54 fails to start and place FW-10, Steam Driven AFW Pump, in service. There are no automatic initiated actions to restore Feedwater flow in this Scenario.**

<u>**Booth Operator**</u>: **If contacted about FW-54, WAIT 5 minutes and REPORT pump appears damaged.**

	BOPO	DETERMINE Main Feedwater is NOT restoring SG levels. [Step 13]
		• [CA] If Main Feedwater is NOT restoring S/G level and SGLS has NOT actuated, ESTABLISH Feedwater by performing step a, b, c, d, or e: [Step 13.1]
		• [CA] DETERMINE Main Feedwater NOT available. [Step 13.1.a]
		• [CA] DETERMINE AFW Pump FW-54 did NOT start. [Step 13.1.b]
		• [CA] DETERMINE AFW Pump FW-06 NOT available. [Step 13.1.c]

Operating Test : NRC	Scenario # 2	Event # 6, 7, 8, & 9	Page 28 of 33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure		

Time	Position	Applicant's Actions or Behavior

CRITICAL TASK STATEMENT		Restore Feedwater Flow to At Least One Steam Generator to Reestablish any SG as a Heat Sink prior to AFAS actuating.
CRITICAL TASK	BOPO	• [CA] INITIATE AFW using FW-10, AFW Pumps to AFW Nozzles: [Step 13.1.c]
		• [CA] START AFW Pump FW-10 at AI-66. [Step 13.1.c.1)]
		• [CA] RESTORE level in at least one SG to 35% to 85% NR or 73% to 94% WR via AFW Nozzles. [Step 13.1.c.2)]
		• OPEN HCV-1107A at AFW Panel AI-66.
		• OPEN HCV-1108A at AFW Panel AI-66.
		• OPEN HCV-1107B at AFW Panel AI-66.
		• OPEN HCV-1108B at AFW Panel AI-66.
		• DETERMINE FCV-1101 & FCV-1102 Feed Regulating Valves CLOSED. [Step 13.a]
		• DETERMINE HCV-1105 & HCV-1106 Feed Regulating Bypass Valves ramped to between 40% & 45% OPEN. [Step 13.b]
	BOPO	• PLACE 43/FW switch in OFF. [Step 13.c]
		• DETERMINE NO Main Feedwater Pumps RUNNING. [Step 13.d]
		• DETERMINE NO Condensate Pumps RUNNING. [Step 13.e]
		• DETERMINE NO Heater Drain Pumps RUNNING. [Step 13.f]
	BOPO	• ENSURE SG Blowdown Isolation Valves CLOSED. [Step 13.g]
		• HCV-1387A & HCV-1387B
		• HCV-1388A & HCV-1388B
	BOPO	VERIFY Steam Dump and Bypass Valves controlling <u>both</u> of the following: [Step 14]
		• DETERMINE RCS T_{COLD} between 525°F and 535°F.
		• DETERMINE Steam Generator pressures ~1000 psia.
		• [CA] If T_{COLD} greater than 525°F, PERFORM the following: [Step 14.1]
		• [CA] DETERMINE Steam Dump and Bypass Valves NOT available. [Step 14.1.a]

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	29	of	33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
	BOPO	• [CA] CONTROL HCV-1040, Atmospheric Dump Valve as required. [Step 14.1.b]
		• [CA] IF REQUIRED, OPERATE Air Assisted Main Steam Safety Valves MS-291 and MS-292. [Step 14.1.c]
	CRS	DETERMINE RCS Heat Removal criteria SATISFIED.
	CRS	VERIFY Containment Isolation criteria satisfied:
	ATCO	VERIFY Normal Containment conditions exist: [Step 15]
		• DETERMINE no unexpected rise in Containment Sump level. [Step 15.a]
		• DETERMINE Containment Area Radiation Monitors NOT in alarm. [Step 15.b]
		• DETERMINE Containment Ventilation Radiation Monitors NOT in alarm. [Step 15.c]
		• DETERMINE SG Blowdown and Condenser Off Gas Radiation Monitors NOT alarming. [Step 15.d]
	ATCO	• DETERMINE SG Blowdown and Condenser Off Gas Radiation Monitors NOT TRENDING to alarm. [Step 15.e]
	CRS	DETERMINE Containment Integrity criteria SATISFIED.
	CRS	DIAGNOSE event in progress: [Step 16]
		• DETERMINE Reactivity Control requirements NOT met.
		• If not, GO TO EOP-20, Functional Recovery.
Booth Operator: **When EOP-20 is entered, If previously contacted for repairs, REPORT as Electrical Maintenance that the DG-01 Output Breaker has been replaced and the area cordoned off for closure. Request Breaker controller to be placed in Pull-to-Lock so that the breaker can be racked up. After 2 minutes, remove malfunction and report that the breaker is ready for closure.**		
Examiner Note: **The following steps are from EOP-20, Functional Recovery.**		

Time	Position	Applicant's Actions or Behavior
	CRS	CONFIRM Standard Post Trip Actions have been performed. [Step 1]
	CRS	IMPLEMENT the Emergency Plan. [Step 2]
		• Time: __________
	CREW	MONITOR the Floating Steps. [Step 3]
	CRS	DETERMINE Feedwater flow has NOT been lost. [Step 4]
	ATCO	DETERMINE all RCPs are STOPPED. [Step 5]
	CRS	DETERMINE that CIAS has NOT occurred and DIRECT Shift Chemist to sample both Steam Generators. [Step 6]
	CRS	IDENTIFY EOP-20 Success Path to satisfy each Safety Function using Safety Function Status Checks or Resource Assessment Trees. [Step 7]
		• VERIFY Reactivity Control NOT SATISFIED and CONSIDER Reactivity Control - Resource Tree A, RC-2: Boration using CVCS, Condition 2.
		• VERIFY Maintenance of Vital Auxiliaries NOT SATISFIED and CONSIDER Maintenance of Vital Auxiliaries - Resource Tree B, MVA-AC: Restoration of AC.
		• DETERMINE RCS Inventory Control SATISFIED.
		• DETERMINE RCS Pressure Control SATISFIED.
		• DETERMINE RCS and Core Heat Removal SATISFIED.
		• DETERMINE Containment Integrity SATISFIED.
Examiner Note:		**The following steps are from EOP-20, Functional Recovery, Section 10, Maintenance of Vital Auxiliaries - AC.**
	CRS	DETERMINE NO 4160 V Safeguards Bus is energized and Reactivity Control Safety Function is in jeopardy. [Step 10.1]
	CRS	DETERMINE both 4160 V Safeguards Buses are DEENERGIZED. [Step 10.2]

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	31	of 33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure							

Time	Position	Applicant's Actions or Behavior
		• [CA] PERFORM step A or B to RESTORE deenergized bus. [Step 10.2.1]
		• [CA] If 1A3 is deenergized, GO TO Step 3. [Step 10.2.1.a]
	BOPO	VERIFY NONE of the following Lockout Relays are tripped: [Step 10.3]
		• 86/1A13 • 86/1A33 • 86/1A3-TFB
	BOPO	If Bus 1A3 is deenergized and DG-1 is running, PERFORM the following to ENERGIZE Bus 1A3: [Step 10.4]
		• OPEN all the following breakers: [Step 10.4.a]
		• 1A33 • 1A13 • FW-6, Electric AFW Pump • RC-3C, RCP • AC-10A, RW Pump • AC-10C, RW Pump • SI-1A, LPSI Pump
CRITICAL TASK STATEMENT		**Restore Power to any 4160 V Safeguards Bus using a Diesel Generator to Reestablish Maintenance of Vital Auxiliaries and Allow Branching to Meet other Safety Functions During a Station Blackout.**
CRITICAL TASK	BOPO	• If DG-1 frequency is > 60 Hz and voltage is > 4160 V, CLOSE breaker 1AD1. [Step 10.4.b]
		• PLACE Breaker 1AD1 in CLOSE. **Examiner Note: Breaker will close when taken out of Pull-to-Lock if repairs have been made.**
		• Time: _______
Examiner Note: The following steps are from EOP-00, Standard Post Trip Actions.		

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	32	of	33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
CRITICAL TASK STATEMENT		**Commence an Emergency Boration of the RCS Due to 2 or more Stuck CEAs when Diesel Generator DG-1 Breaker is Closed and Bus 1A3 is Reenergized to Restore Reactivity Control.**
CRITICAL TASK	ATCO	• [CA] If more than one CEA is NOT fully inserted, PERFORM the following to initiate Emergency Boration: [Step 1.2]
		• [CA] ENSURE both following valves CLOSED: [Step 1.2.a]
		• [CA] FCV-269X, Demin Water Makeup Valve.
		• [CA] FCV-269Y, Boric Acid Makeup Valve.
		• [CA] ENSURE all the following valves OPEN: [Step 1.2.b]
		• [CA] HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
		• [CA] HCV-265, CH-11A Gravity Feed Valve.
		• [CA] HCV-258, CH-11B Gravity Feed Valve.
		• [CA] START all the following pumps: [Step 1.2.c]
		• [CA] Boric Acid Pump CH-4A.
		• [CA] Charging Pump CH-1A.
		• [CA] CLOSE LCV-218-2, VCT Outlet Valve. [Step 1.2.d]
		• [CA] ENSURE the following valves CLOSED: [Step 1.2.e]
<u>**Booth Operator**</u>: **When contacted, EXECUTE remote function to LOCALLY CLOSE LCV-218-3, Charging Pump Suction SIRWT Isolation Valve.**		
		• [CA] LOCALLY OPEN LCV-218-3, Charging Pump Suction SIRWT Isolation Valve.
		• [CA] HCV-257, CH-4B Recirc Valve.
		• [CA] HCV-264, CH-4A Recirc Valve.
		• [CA] BORATE until adequate Shutdown Margin is established. [Step 1.2.f]

Operating Test :	NRC	Scenario #	2	Event #	6, 7, 8, & 9	Page	33	of	33
Event Description:	Loss of Offsite Power / Four Stuck CEAs / Train A Diesel Generator Breaker Failure / Train B Diesel Generator Overspeed Trip / Diesel Driven Auxiliary Feedwater Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
<u>Examiner Note</u>: The following steps continue from EOP-20, Functional Recovery.		
	CRS	VERIFY Safety Functions are being satisfied at 15 minute intervals. [Step 8]
	CRS	If Safety Function Status Check Acceptance Criteria are satisfied, PERFORM instructions for all Success Paths in use. [Step 9]
	CRS	IMPLEMENT Section 18, Long Term Actions, when both of the following are SATISFIED: [Step 10]
		• INSTRUCTIONS for all Success Paths have been performed.
		• Safety Function Status Check Acceptance Criteria for Success Paths in use are being SATISFIED.
When Emergency Boration is initiated, TERMINATE the scenario.		

Facility:	Fort Calhoun Station	Scenario No.:	3	Op Test No.:	Dec 2015 NRC
Examiners:	____________	Operators:	____________		
	____________		____________		
	____________		____________		

Initial Conditions: 100% power MOL - RCS Boron is 482 ppm (by sample).

Turnover: Maintain steady-state power conditions. Rotate Heater Drain Pumps FW-5B and FW-5C per OI-VD-1, Feedwater Heater Vents and Drains Normal Operation. Charging Pump CH-1C out of service for packing repair.

Critical Tasks:

- Stop One Reactor Coolant Pump in Each Loop when Reactor Coolant System Pressure is < 1350 psia, Prior to losing Reactor Coolant Pump Net Positive Suction Head. (Event 7)
- Isolate the Affected Steam Generator with a Tube Rupture to Minimize Spread of Contamination. (Event 7)
- Reduce and Maintain RCS T_{HOT} ≤ 510°F to Maintain SG Pressure Below Safety Valve Setpoint of 1000 psia. (Event 7)

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min		N (BOPO)	Rotate Heater Drain Pumps per OI-VD-1, Feedwater Heater Vents and Drains Normal Operation, Attachment 2.
2 +20 min		I (ATCO, CRS)	Pressurizer Level Channel Transmitter LT-101X Fails Low. Transfer Pressurizer Level Control to LT-101Y.
3 +30 min		I (BOPO, CRS)	Steam Generator RC-2A Steam Flow Transmitter FT-907 Fails High. Bypass Affected Transmitter.
4 +40 min		C (ATCO, CRS) TS (CRS)	Charging Pump CH-1A Trip. Restore Letdown and Charging Flow.
5 +50 min		C (ATCO,BOPO, CRS) TS (CRS)	Steam Generator RC-2B Tube Leak Greater Than 150 GPD. Isolate Blowdown Flow.
6 +60 min		R (ATCO) N (BOPO, CRS)	Commence Plant Shutdown per AOP-05, Emergency Shutdown.
7 +70 min		M (ATCO, BOPO, CRS)	Steam Generator RC-2B Tube Rupture at 500 GPM on 10 Minute Ramp Upon 3% to 5% Load Reduction.
8 +70 min		I (BOPO)	Diesel Generator DG-01 Start Failure on SIAS. Manual Start Required.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
1	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC 3

The crew will assume the shift at 100% power per OP-4, Load Change and Normal Power Operation. The scheduled activity is to rotate Heater Drain Pumps by starting FW-5C and securing FW-5B per OI-VD-1, Feedwater Heater Vents and Drains Normal Operation, Attachment 2, Rotating Operating Heater Drain Pumps.

The next event is a low failure of Pressurizer Level Control Channel, LT-101X. Operator actions are per ARP-CB-1/2/3/A4, Window C-8 – PRESSURIZER LEVEL LO-LO CHANNEL X. The crew will transfer to the standby channel LT-101Y and restore Letdown per OI-RC-8, Reactor Coolant System Level Control Normal Operation, Attachment 8, Transferring Pressurizer Level Control Channel in CASCADE and Attachment 4, Transferring Letdown Controller from AUTOMATIC to MANUAL.

When plant conditions are stable, a high failure of Steam Generator RC-2A Steam Flow Transmitter FT-907 will occur. Initial operator actions are per ARP-DCS-FW, Feedwater DCS Annunciator Response Procedure and include verifying Feedwater Control is in Single Element Control, bypassing the failed input, and determining 3 Element Control is restored.

The next event is a trip of the running Charging Pump. Operator actions are per ARP-CB-1/2/3/A2, Window A-6L – CHARGING FLOW LO and include isolating of Letdown and verifying no system leaks exist. Charging Pump CH-1B is placed in service per OI-CH-1, Chemical and Volume Control System Normal Operation, Attachment 1, Startup of Charging and Letdown. The SRO will refer to Technical Specification LCO 2.2.4 – Charging Pumps - Operating.

When Charging flow is restored, a Steam Generator Tube Leak of greater than 150 gallons per day will occur on Steam Generator RC-2B. The crew will enter AOP-22, Reactor Coolant Leak, and implement Attachment B, Primary to Secondary Leak Rate Actions. RM-064, Main Steam Line Radiation Monitor, is placed in service to assist in determining leak size and location. Various Secondary Side valves are closed to minimize system contamination and HR-21, Blowdown Operation is performed to isolate blowdown flow from SG RC-2B. The SRO will refer to Technical Specification LCO 2.1.4 – Reactor Coolant System Leakage Limits.

Once blowdown is isolated, entry into AOP-05, Emergency Shutdown, is performed to bring the plant into MODE 4. When power has been reduced 3% to 5%, a Steam Generator Tube Rupture of 500 gpm will commence on a 10 minute ramp.

The crew enters EOP-00, Standard Post Trip Actions, and then transitions to EOP-04, Steam Generator Tube Rupture. Diesel Generator DG-01 fails to start upon SIAS and must be manually started. While in EOP-04, the Reactor Coolant System is cooled per HR-12, Secondary Heat Removal Operation, and the RCS is depressurized to less than 1000 psia per PC-11, Pressure Control, to allow isolating the affected Steam Generator. When SG RC-2B is isolated, the scenario is terminated.

<u>Risk Significance</u>:

• Failure of risk important system prior to trip:	Loss of Charging Pump Steam Generator Tube Leak
• Risk significant core damage sequence:	Steam Generator Tube Rupture
• Risk significant operator actions:	Stop RCPs Upon Loss of Subcooling Isolate Affected Steam Generator Cooldown and Depressurize RCS

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

RESET to IC-#103 (or any 100% MOL IC) and LOAD & EXECUTE NRC 3.sce for NRC Scenario 3.

Preset Item – CH-1C Removed from Service

Type	Item	Value	Condition
Malfunction	BUS_1B3B_4B_5_BKR_Trip	True	Scenario Event: "CH-1C OOS"

Preset Item – Event 9 – Diesel Generator #1 Auto Start Failure

Type	Item	Value	Condition
Expert	H_PD1_033_3 H_PD1_031_3	Reset Reset	Scenario Event: "DG-1 Auto Start Failure"

Event 2 – Pressurizer Level Transmitter LT-101X Fails Low

Type	Item	Value	Condition
Transmitter	RCS_LT101X	0, ramp = 5 seconds	When directed by examiner, trigger/activate this event. Scenario Event: "Pzr Level LT-101X Fail Low"

Event 3 – Steam Generator Flow Transmitter LT-907 Fails High

Type	Item	Value	Condition
Transmitter	FT-907 FT-907 DCS FT-907-1 DCS	4000000, ramp = 5 sec Fail High Fail High	When directed by examiner, trigger/activate this event. Scenario Event: "SG Flow FT-907 Fail High"

Event 4 – Charging Pump CH-1A trips

Type	Item	Value	Condition
Malfunction	BUS_1B3A_4_BKR_TRIP	True	When directed by examiner, trigger/activate this event. Scenario Event: "CH-1A Trip"

Event 5 – Primary-to-Secondary SG Tube Leak Develops in Steam Generator RC-2B

Type	Item	Value	Condition
Malfunction	RCS04B	0.001	When directed by examiner, trigger/activate this event. Scenario Event: "RC-2B S/G Tube Leak"

Event 7 – Steam Generator Tube Leak in RC-2B Grows to Tube Rupture

Type	Item	Value	Condition
Malfunction	RCS04B	1.4, ramp = 600 sec	When directed by examiner, trigger/activate this event. Scenario Event: "RC-2B S/G Tube Rupture"

Scenario Event Description NRC Scenario 3

Booth Operator:	**INITIALIZE to IC-1 and LOAD NRC 3.sce.** **ENSURE all Simulator Annunciator Alarms are ACTIVE.** **ENSURE all Control Board Tags are removed.** **ENSURE Charging Pump CH-1A in service.** **ENSURE Charging Pump CH-1C OOS for emulsified oil replacement with Information Tag attached.** **ENSURE Channel X Pressurizer Pressure and Level selected.** **ENSURE Reactivity Briefing Sheet printout provided with Turnover.** **ENSURE Middle-of-Life Thumb Rule Sheet provided with Turnover.** **ENSURE ERF Computer System Display set to FWD for BOPO.** **ENSURE procedures in progress provided to crew in Briefing Room:** **- COPY of OI-VD-1, Feedwater Heater Vents and Drains Normal Operation, Attachment 2, Rotating Operating Heater Drains Pumps, INITIALED through Prerequisites and Procedure Step 2.**
Control Room Annunciators in Alarm: **NONE**	

Procedure List
Event 1: OP-4, Load Change and Normal Power Operation.
Event 1: OI-VD-1, Feedwater Heater Vents and Drains Normal Operation
Event 2: ARP-CB-1/2/3/A4, Window C-8, PRESSURIZER LEVEL LO-LO CHANNEL X
Event 3: ARP-DCS-FW, Feedwater DCS Annunciator Response Procedure
Event 4: ARP-CB-1/2/3/A2, Window A-6L, CHARGING FLOW LO
Event 4: OI-CH-1, Chemical and Volume Control System Normal Operation, Attachment 1, Startup of Charging and Letdown
Event 5: AOP-22, Reactor Coolant Leak
Event 5: HR-21, Blowdown Operation
Event 6: AOP-05, Emergency Shutdown
Event 7: EOP-00, Standard Post Trip Actions
Event 7: EOP-04, Steam Generator Tube Rupture
Event 8: HR-12, Secondary Heat Removal Operation
Event 8: PC-11, Pressure Control

Operating Test : NRC	Scenario # 3	Event # 1	Page 5 of 34
Event Description: Rotate Heater Drain Pumps			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, RESPOND to requests from Control Room.		
Indications Available: **NONE**		
Examiner Note: The following steps are from OI-VD-1, Feedwater Heater Vents and Drains Normal Operation, Attachment 2.		
	BOPO	PERFORM the following at CB-10, 11: [Step 3]
		• PLACE 43/FW Switch in OFF. [Step 3.a]
		• VERIFY Annunciator CB-10,11/A10, Window B-6L – 43/FW TRANSFER SWITCH OFF AUTO in ALARM. [Step 3.b]
Examiner Note: XC105 is the Computer (DCS) generated value for Secondary Calorimetric.		
	CRS	DECLARE XC105 invalid. [Step 4]
	BOPO	Make plant announcement, then: PLACE FW-5C, Heater Drain Pump control switch to AFTER-START at CB-10, 11. [Step 5]
	BOPO	VERIFY FW-5C, Heater Drain Pump ammeter returns to less than 80 amps in less than 15 seconds and STABILIZES at ~ 66 amps. [Step 6]
Booth Operator: If contacted, REPORT FCV-1216C is closed.		
	BOPO	VERIFY FCV-1216C, Heater Drain Pump FW-5C Recirculation Control Valve CLOSES. [Step 7]
	BOPO	PLACE FW-5B, Heater Drain Pump control switch to AFTER-STOP at CB-10, 11. [Step 8]
NOTE Verification of Cooling Water Flow to the Seal cooler will be used to ensure Stuffing Box pressure is < 250 psig when Pressure Gauge PI-1192A, B, or C is out of service.		

Operating Test : NRC	Scenario # 3	Event # 1	Page 6 of 34
Event Description: Rotate Heater Drain Pumps			

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: If contacted, REPORT FW-5C discharge and stuffing box pressures normal.		
	BOPO	MONITOR the following parameters on Heater Drain Pump FW-5C: [Step 9]
		• Motor amperage at ~66 amps.
		• PI-1269C, Pump Discharge pressure at ~160 psig on ERF Computer.
		• Heater Drain Tank level ~54% on CB-10, 11.
		• Bearing temperatures on ERF Display FWD normal.
		• PI-1192C, Stuffing Box pressure < 250 psig read locally.
<u>Booth Operator</u>: If contacted, REPORT FW-5B is not rotating in reverse.		
	BOPO	CONTACT Auxiliary Operator to VERIFY FW-5B, Heater Drain Pump NOT ROTATING in reverse direction. [Step 10]
	BOPO	PERFORM the following at CB-10, 11: [Step 11]
		• PLACE 43/FW Switch in AUTO. [Step 11.a]
		• VERIFY Annunciator CB-10, 11/A10, Window B-6L – 43/FW TRANSFER SWITCH OFF AUTO is CLEAR. [Step 11.b]
<u>Booth Operator</u>: If contacted, REPORT Shift Technical Advisor will restore GARDEL.		
	CRS	CONTACT Shift Technical Advisor to RESTORE GARDEL data feed per OI-ERFCS-2. [Step 12]
	STA	When 12 minute validity period has passed <u>and</u> parameters are steady-state, DECLARE XC105 valid and ENTER in Control Room Log. [Step 13]
When restoration of XC105 is discussed, PROCEED to Event 2.		

Operating Test : NRC	Scenario # 3	Event # 2	Page 7 of 34
Event Description: Pressurizer Level Channel Transmitter Failure			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 2. - Pressurizer Level Channel Transmitter LT-101X fails low.		
Indications Available: **CB-1,2,3/A4 – PRESSURIZER LEVEL LO-LO CHANNEL X** **CB-1,2,3/A4 – PRESSURIZER LEVEL HI-LO CHANNEL X** **Charging Pump CH-1B starts** **Letdown flow to minimum (~26 gpm)**		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
	ATCO	INFORM CRS of Pressurizer Level Channel LT-101X failure.
Examiner Note: ATCO may "Operate to Mitigate" per OPD 4-09 and TRANSFER to Channel Y.		
	CRS	REFER to ARP-CB-1,2,3/A4, Window C-8 – PRESSURIZER LEVEL LO-LO CHANNEL X.
Examiner Note: During this event, pressurizer pressure may decrease to less than 2075 psia. If this occurs, the crew should address TS 2.10.4.5 for pressurizer low pressure.		
Examiner Note: The following steps are from ARP-CB-1,2,3/A4, Window C-8, PRESSURIZER LEVEL LO-LO CHANNEL X.		
	ATCO	VERIFY Pressurizer Level on LR-101X/LR-101Y. [Step 1]
		• If Pressurizer level is NOT low, PERFORM the following: [Step 1.1]
		• PLACE HC-101 to Channel Y per OI-RC-8. [Step 1.1.1]
		• If desired, PLACE HIC-101-1/101-2, Letdown Throttle Valves Controller to MANUAL per OI-RC-8. [Step 1.1.2]
		• PLACE HC-101-1, Pzr Heater Cutout Channel Select Switch, to Channel Y. [Step 1.1.3]
Examiner Note: The following steps are from OI-RC-8, Reactor Coolant System Level Control Normal Operation, Attachment 8, Transferring Pressurizer Level Control Channel (X to Y or Y to X) in CASCADE.		
	ATCO	ENSURE both Level Controllers are in (C) CASCADE: [Step 1]

Time	Position	Applicant's Actions or Behavior
		• LC-101X-1, Pressurizer Level Controller
		• LC-101Y-1, Pressurizer Level Controller
	ATCO	If desired, PLACE Letdown Controller HIC-101-1/101-2, Letdown Throttle Valves Controller in MANUAL per Attachment 4. [Step 2]
Examiner Note: **The following steps are from OI-RC-8, Attachment 4, Transferring Letdown Controller from AUTOMATIC to MANUAL.**		
	ATCO	ENSURE Letdown Controller HIC-101-1/101-2, Letdown Throttle Valves Controller in AUTO. [Step 1]
	ATCO	PLACE HIC-101-1/101-2 Manual/Auto Transfer Switch to BAL. [Step 2]
	ATCO	ADJUST Manual Control Knob on HIC-101-1/101-2 until TOP SCALE indicates 50% (zero deviation; red pointer aligned with the red dot). [Step 3]
	ATCO	PLACE HIC-101-1/101-2 Manual/Auto Transfer Switch to MAN. [Step 4]
	ATCO	If necessary, MAKE adjustments to HIC-101-1/101-2 Manual Control Knob to MAINTAIN desired Pressurizer Level. [Step 5]
Examiner Note: **The following steps continue from OI-RC-8, Attachment 8.**		
CAUTION Transfer from the Selected Controller to the Non-Selected Controller should not be performed until both controller outputs are approximately equal.		
	ATCO	VERIFY Controller LR-101Y has INDICATED Pressurizer Level and PROGRAMMED Pressurizer Level Setpoint MATCHED prior to transfer. [Step 3]
	ATCO	PLACE HC-101, Pressurizer Level Channel Selector Switch, to Channel Y. [Step 4]

Operating Test :	NRC	Scenario #	3	Event #	2	Page	9	of	34
Event Description:	Pressurizer Level Channel Transmitter Failure								

Time	Position	Applicant's Actions or Behavior
	ATCO	ENSURE Controller LC-101Y-1 is controlling INDICATED Pressurizer Level at PROGRAMMED Setpoint. [Step 5]
	ATCO	PUSH LC-101-1 & LC-101-2, Charging Pump Bistable Reset buttons on Reactor Regulating System Panel AI-4B and VERIFY all bistables are RESET. [Step 6]
	ATCO	If required, PLACE Letdown Controller HIC-101-1/101-2, Letdown Throttle Valves Controller, in AUTO per Attachment 3. [Step 7]
Examiner Note: **The following steps are from OI-RC-8, Attachment 3, Transferring Letdown Controller from MANUAL to AUTOMATIC.**		
	ATCO	ENSURE Letdown Controller HIC-101-1/101-2, Letdown Throttle Valves Controller is in (M) MANUAL. [Step 1]
	ATCO	Manually ADJUST HIC-101-1/101-2, Letdown Throttle Valves Controller and PIC-210, Letdown Press Controller until following parameters are met: [Step 2]
		• Indicated Pressurizer Level matches the Programmed Pressurizer Level Setpoint on LR-101X or LR-101Y, Pressurizer Level Recorder.
		• PIC-210 is maintaining 200 psi to 400 psi.
	ATCO	ADJUST bias knob on HIC-101-1/101-2 until the top scale indicates 50% (zero deviation; red pointer aligned with the red dot). [Step 3]
	ATCO	PLACE HIC-101-1/101-2 Manual/Auto Transfer Switch to BAL, then to AUTO. [Step 4]
	ATCO	If necessary, ADJUST the bias knob of HIC-101-1/101-2 to ENSURE Indicated Pressurizer Level is maintained at Programmed Pressurizer Level setpoint. [Step 5]
Examiner Note: **The following steps continue from ARP-CB-1,2,3/A4, Window C-8.**		
	ATCO	VERIFY RCS Pressure on PR-103X/PR-103Y > 1600 psia. [Step 2]

Operating Test :	NRC	Scenario #	3	Event #	2	Page	10	of	34
Event Description:	Pressurizer Level Channel Transmitter Failure								

Time	Position	Applicant's Actions or Behavior
	ATCO	ENSURE all Pressurizer Heaters DEENERGIZED. [Step 3]
	ATCO	DETERMINE RCS Cold Leg temperatures on A-D/TI-112C and A-D/TI-122C are NOT lowering. [Step 4]
		• CHECK VCT level on LI-219, for indication of lowering level. [Step 4.1]
		• DETERMINE VCT level is NOT lowering. [Step 4.2]
	ATCO	VERIFY the following CVCS parameters: [Step 5]
		• ENSURE Letdown at minimum flow of 26 gpm on FIC-212. [Step 5.1]
		• ENSURE Charging Pumps CH-1A & CH-1B are RUNNING. [Step 5.2]
	ATCO	NOTIFY Work Week Manager of Pressurizer level instrument failure. [Step 6]
When Pressurizer level is normal, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	3	Event #	3	Page	11	of	34
Event Description:	Steam Generator Steam Flow Transmitter Failure								

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 3. - Steam Generator RC-2A Steam Flow Transmitter FT-907 fails high.		
Indications Available: Feedwater Digital Control System Alarm		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS Steam Generator RC-2A Steam Flow Transmitter FT-907 failed high.
	CRS	DIRECT actions of ARP-DCS-FW, FT-907.
Examiner Note: The following steps are from ARP-DCS-FW, Feedwater Digital Control System.		
	BOPO	PERFORM the following for Steam Flow Instrument FT-907 failure: [Step 1]
		• VERIFY that FORCED TO 1 ELEM and 1 ELEM AUTO is displayed on Feedwater Regulating System display for RC-2A PT-907. [Step 1.1]
		• TOUCH display with the BAD process. [Step 1.2]
		• DETERMINE BAD input NOT automatically bypassed. [Step 1.3]
		◦ TOUCH Bypass on verification faceplate to BYPASS BAD input. [Step 1.3.1]
		◦ VERIFY point displays GOOD status. [Step 1.3.2]
		◦ ENSURE control SHIFT to 3 ELEMENT AUTO. [Step 1.3.3]
	CRS	DETERMINE Steam Generator level instruments NOT affected. [Step 2]
	CRS	DETERMINE BAD input bypassed MANUALLY. [Step 3]
	BOPO	MONITOR Steam Generator levels. [Step 4]
	CRS	VERIFY XC-105, Secondary Calorimetric, is valid. [Step 5]
	CRS	DETERMINE LT-903 or LT-906 NOT cause of alarm. [Step 6]

Operating Test :	NRC	Scenario #	3	Event #	3	Page	12	of	34

Event Description: Steam Generator Steam Flow Transmitter Failure

Time	Position	Applicant's Actions or Behavior
	BOPO	NOTIFY Work Week Manager of FT-907 malfunction. [Step 7]

When Steam Generator levels are normal, PROCEED to Event 4.

Operating Test : NRC	Scenario # 3	Event # 4	Page 13 of 34
Event Description: Charging Pump Trip			

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Event 4. - Charging Pump CH-1A trip.**		
Indications Available: **CB-1,2,3/A2 – CHARGING PUMPS TRIP** **CB-1,2,3/A2 – CHARGING FLOW LO**		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
Examiner Note: **ATCO may "Operate to Mitigate" per OPD 4-09 and START CH-1B to avoid losing Letdown flow. Charging Pump CH-1B does not AUTO START until a level deviation exists.**		
	ATCO	INFORM CRS of Charging Pump CH-1A trip.
	CRS	REFER to ARP-CB-1,2,3/A2, Window A-6L – CHARGING FLOW LO.
Examiner Note: **The following steps are from ARP-CB-1,2,3/A2, Window A-6L – CHARGING FLOW LO.**		
	ATCO	OBSERVE Charging Header flow LOW. [Step 1]
	ATCO	If Charging flow is lost, CLOSE TCV-202 and HCV-204 to ISOLATE Letdown. [Step 2]
		• DETERMINE TCV-202, Letdown to Regenerative Heat Exchanger Isolation Valve AUTO CLOSED or manually CLOSE.
		• Manually CLOSE HCV-204, Reactor Coolant to Letdown Heat Exchanger Isolation Valve.
NOTE Based on plant conditions, XC-105 and GARDEL may be invalid.		
Booth Operator: **When contacted about the status of CH-1A, REPORT a breaker overcurrent trip. Investigation of CH-1A: The pump looks normal locally. If Maintenance or Work Week Manager is contacted, estimated time to restore CH-1C is 4 hours.**		

Operating Test :	NRC	Scenario #	3	Event #	4	Page	14	of	34
Event Description:	Charging Pump Trip								

Time	Position	Applicant's Actions or Behavior
	ATCO	If required, ROTATE Charging Pumps per OI-CH-1, CVCS Normal Operation, Attachment 1, Startup of Charging and Letdown. [Step 5]
	CRS	EVALUATE Technical Specification LCO 2.2, Chemical and Volume Control System
		• LCO 2.2.4 - Charging Pumps - Operating
		• CONDITION LCO 2.2.4.(1) – Two Charging Pumps inoperable • ACTION LCO 2.2.4.(1) – RESTORE to at least two OPERABLE Charging Pumps within 72 hours.
When Charging and Letdown flows are restored, PROCEED to Event 5.		

Operating Test :	NRC	Scenario #	3	Event #	5	Page	15	of	34
Event Description:	Steam Generator Tube Leak								

Time	Position	Applicant's Actions or Behavior
Booth Operator:		**When directed, EXECUTE Event 5. - Steam Generator RC-2B Tube Leak greater than 150 gpd.**
Indications Available: **RM-057, Condenser Off Gas Radiation Monitor in alarm and trending up RM-054B, Steam Generator RC-2B Blowdown Radiation Monitor in alarm and trending up**		
+30 sec	ATCO	RESPOND to Radiation Monitor Alarms.
	ATCO	INFORM CRS of indications of the tube leak on Steam Generator RC-2B.
	CRS	REFER to AOP-22, Reactor Coolant Leak.
Examiner Note:		**The following steps are from AOP-22, Reactor Coolant Leak, Section I, Leak Rate Determination and Leak Isolation.**
	CRS	DETERMINE Shutdown Cooling is NOT in operation. [Step 4.1]
Booth Operator:		**When contacted as Shift Chemist, WAIT 2 minutes and REPORT Steam Generator RC-2B has increased activity and RC-2A has normal activity.**
	CRS	DETERMINE CIAS is NOT present and DIRECT Shift Chemist to PERFORM the following: [Step 4.2]
		• PERFORM rapid activity analysis of both Steam Generators. [Step 4.2.a]
		• SAMPLE both SGs per CH-SMP-SE-0015, Steam Generator Sampling - Room 60. [Step 4.2.b]
	CRS	IMPLEMENT the Emergency Plan. [Step 4.3]
	CREW	MONITOR the Floating Steps. [Step 4.4]
	ATCO	DETERMINE Pressurizer level is NOT below programmed level. [Step 4.5]

Operating Test : NRC	Scenario # 3	Event # 5	Page 16 of 34
Event Description: Steam Generator Tube Leak			

Time	Position	Applicant's Actions or Behavior
	ATCO and/or BOPO	DETERMINE RCS leakage rate per IC-17, RCS Manual Leak Rate Calculation. [Step 4.6]
	CRS	DETERMINE RCS leak rate is NOT greater than 40 gpm. [Step 4.7]
<u>Booth Operator</u>: When contacted as Shift Chemist, WAIT 10 minutes, then REPORT initial Steam Generator RC-2B leak rate is greater than 150 gallon per day.		
	CRS	DIRECT Shift Chemist to verify primary to secondary leak rate < 1 gpd per CH-AD-0007, Primary to Secondary Leak Rate Determination. [Step 4.8]
		• [CA] If primary to secondary leak rate is > 1 gpd, IMPLEMENT Attachment B, Primary to Secondary Leak Rate Actions.
<u>Examiner Note</u>: The following steps are from AOP-22, Reactor Coolant Leak, Attachment B, Primary to Secondary Leak Rate Actions.		
	CRS	IMPLEMENT SO-G-105, Steam Generator Tube Leakage. [Step 1]
<u>Booth Operator</u>: When contacted, REPORT Work Week Manager will implement SO-G-105.		
	ATCO	Continuously MONITOR count rate trends for radiation monitors RM-054A, RM-054B and RM-057 on ERF Computer System. [Step 2]
	ATCO	PERFORM the following to PLACE RM-064, Main Steam Line Radiation Monitor, in service at AI-33C: [Step 3]
		• PLACE Main Steam Line A/B Enable Switch for HCV-921 and HCV-922 in ON. [Step 3.a]
		• PLACE Main Steam Line A/B Mode Selector Switch in AUTO. [Step 3.b]
	CRS	PERFORM the following to IDENTIFY SG with tube leak: [Step 4]
	CRS	• DIRECT Shift Chemist to continue sampling. [Step 4.a]
	CRS	• REVIEW Steam Generator analysis for activity. [Step 4.b]
	ATCO	• MONITOR RM-057 & RM-064, Steam Line Radiation Monitors and DETERMINE both radiation levels RISING. [Step 4.c]

Operating Test : NRC	Scenario # 3	Event # 5	Page 17 of 34
Event Description: Steam Generator Tube Leak			

Time	Position	Applicant's Actions or Behavior
	ATCO	• MONITOR RM-054A & RM-054B, SG Blowdown Radiation Monitors and DETERMINE RM-054B is RISING [Step 4.d]
	BOPO	• MONITOR SG levels and DETERMINE no apparent change. [Step 4.e]
Booth Operator: When contacted, EXECUTE remote functions to position HC-2509 / HC-2508 / FW-268 / FW-266 as required.		
	CREW	Direct Equipment Operators to PERFORM the following to MINIMIZE spread of contamination: [Step 5]
		• ENSURE HC-2509, SAMPLE DRAIN TO DRAIN HEADER, is OPEN at AI-107 in Room 60. [Step 5.a]
		• ENSURE HC-2508, SAMPLE DRAIN TO CONDENSER C.W. TUNNEL, is CLOSED at AI-107 in Room 60. [Step 5.b]
		• ENSURE FW-268, CONDENSATE DUMP VALVE LCV-1193 OUTLET ISOLATION VALVE, is CLOSED at Turbine Building Mezzanine. [Step 5.c]
		• ENSURE FW-266, CONDENSATE DUMP VALVE LCV-1193 BYPASS VALVE, is CLOSED at Turbine Building Mezzanine. [Step 5.d]
	CRS	• DETERMINE SG RC-2A is least affected Steam Generator. [Step 5.e]
	BOPO	• DETERMINE SG RC-2B is most affected Steam Generator and PERFORM the following: [Step 5.f]
		• PLACE YCV-1045B, RC-2B to FW-10 Isolation Valve in OVERRIDE.
		• PLACE YCV-1045B, RC-2B to FW-10 Isolation Valve to CLOSE.
	CRS	• CONSIDER stopping Turbine Building Sump Pumps VD-1A & VD-1B. [Step 5.g]
	CRS	• CONSIDER isolating steam generator blowdown. [Step 5.h]
	BOPO	• PLACE RCV-978, 6th Stage Extraction Isolation Valve to STOP. [Step 5.i]
Booth Operator: When contacted, EXECUTE remote function to align Condenser Evacuation Discharge to Auxiliary Building Stack.		
	CRS	• CONTACT Auxiliary Operator to ALIGN Condenser Evacuation Discharge to Auxiliary Building stack per OI-CE-1, Condenser Evacuation System Normal Operation. [Step 5.j]
	CRS	• DIRECT Radiation Protection to develop a method for processing contaminated Condensate. [Step 5.k]

Operating Test :	NRC	Scenario #	3	Event #	5	Page	18	of	34
Event Description:	Steam Generator Tube Leak								

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE primary to secondary leakage greater than 5 gpd. [Step 6]
	CRS	DETERMINE primary to secondary leakage greater than 30 gpd. [Step 7]
	CRS	DETERMINE primary to secondary leakage greater than 30 gpd independent of Xe-133 concentration. [Step 8]
	CRS	DETERMINE primary to secondary leakage greater than 75 gpd independent of Xe-133 concentration. [Step 9]
	CRS	DETERMINE primary to secondary leakage greater than 75 gpd with a rate increase greater than 30 gpd in any 1 hour period. [Step 10]
	CRS	DETERMINE primary to secondary leakage greater than 75 gpd with a rate increase greater than 30 gpd in 1 hour. [Step 11]
	CRS	DETERMINE primary to secondary leak rate greater than 150 gpd (0.10 gpm) and PERFORM the following: [Step 12]
		• ISOLATE blowdown from SG RC-2B per HR-21, Blowdown Operation. [Step 12.a]
		• COMMENCE a Plant Shutdown to MODE 4 per AOP-05, Emergency Shutdown. [Step 12.b]
	CRS	EVALUATE Technical Specification LCO 2.1, Reactor Coolant System.
		• LCO 2.1.4 - Reactor Coolant System Leakage Limits
		• CONDITION LCO 2.2.4.(1).(d) – RCS operational LEAKAGE shall be limited to 150 gallons per day primary to secondary LEAKAGE through anyone Steam Generator. • ACTION LCO 2.1.4.(3) – Primary to secondary LEAKAGE is not within limits, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown within 36 hours.
When Technical Specifications have been addressed, PROCEED to Event 6.		

Operating Test : NRC	Scenario # 3	Event # 6	Page 19 of 34
Event Description: Commence Plant Shutdown			

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When directed, RESPOND to requests from Control Room.		
<u>Indications Available</u>: **NONE**		
<u>Examiner Note</u>: The following steps are from AOP-05, Emergency Shutdown.		
<u>NOTE</u> TDB-III-23a and the Power Ascension/Power Reduction Strategy (PAPRs) provide guidance for the shutdown.		
	CRS	CONTACT Reactor Engineer if additional guidance is required. [Step 4.1]
<u>NOTE</u> Operation of more than one Charging Pump will raise the rate of the power reduction.		
<u>Examiner Note</u>: Unless directed, boration will occur from the Safety Injection Refueling Water Tank (SIRWT) when in AOP-05 to avoid time constraints.		
	CRS	If borating from SIRWT, COMMENCE boration by performing the following: [Step 4.2]
	ATCO	• DETERMINE Charging Pump, CH-1B RUNNING. [Step 4.2.a]
	ATCO	• OPEN LCV-218-3, Charging Pump Suction SIRWT Isolation Valve. [Step 4.2.b]
	ATCO	• CLOSE LCV-218-2, VCT Outlet Valve. [Step 4.2.c]
	CRS	DETERMINE Boration alignment from CVCS NOT required. [Step 4.3]
	CRS	NOTIFY Energy Marketing of power reduction. [Step 4.4]
<u>NOTE</u> During the power reduction, maintain T_C <u>PER</u> TDB Figure III.1, <u>T_{ave} Program</u>.		
	BOPO	MAINTAIN RCS Temperature Control via Turbine Load per HR-12, Secondary Heat Removal Operation: [Step 4.5]

Operating Test : NRC	Scenario # 3	Event # 6	Page 20 of 34
Event Description: Commence Plant Shutdown			

Time	Position	Applicant's Actions or Behavior
		• MAINTAIN T_{COLD} 527°F to 547°F AND
		• MAINTAIN T_{COLD} +0°F to -1°F of program.
	ATCO	MAINTAIN Pressurizer Level via Charging and Letdown per IC-11, Inventory Control: [Step 4.6]
		• MAINTAIN Pressurizer Level 45% to 60% AND
		• MAINTAIN Pressurizer Level within 4% of program.
	ATCO	PERFORM the following to MAINTAIN VCT level between 55% and 85%: [Step 4.7]
		• As required, PLACE LCV-218-1, VCT Inlet Valve to RWTS. [Step 4.7.a]
		• When diversion is complete, PLACE LCV-218-1, VCT Inlet Valve to AUTO. [Step 4.7.b]
	ATCO	PERFORM the following to MAXIMIZE Pressurizer Heaters and Spray: [Step 4.8]
		• As required, PLACE Backup Heater Control Switches to ON. [Step 4.8.a]
		• ADJUST PC-103X or PC-103Y, Pressurizer Pressure Controller Setpoint Pushbutton to maintain pressure between 2080 psia and 2145 psia.[Step 4.8.b]

CAUTION

Do not insert CEAs below power dependent insertion limit.

Time	Position	Applicant's Actions or Behavior
	ATCO	As required, ADJUST Regulating Group 4 to CONTROL ASI per OI-RR-1, Attachment 4, Axial Shape Index (ASI) Control. [Step 4.9]

Examiner Note: The following steps are from HR-12, Secondary Heat Removal Operation.

Time	Position	Applicant's Actions or Behavior
	BOPO	ENSURE Turbine Control is in MANUAL. [Step 1]

NOTE

Output will be highlighted by a yellow box when selected.

	BOPO	PUSH the OUT button to select OUTPUT. [Step 2]

NOTES

1. Depressing the single arrow will adjust turbine load by 0.1%. Depressing the double arrow will adjust turbine load by 0.5%.
2. T_c should be maintained within (+)0°F, (-)1°F of program per TDB-III.1, $\underline{T_{ave}}$ <u>Program</u>.

	BOPO	PRESS single <u>or</u> double UP[▲] or DOWN[▼] arrow to maintain Turbine Load: [Step 3]
		• MAINTAIN T_{COLD} 527°F to 547°F.
		• MAINTAIN T_{COLD} +0°F to -1°F of program.

<u>Examiner Note</u>: Do not proceed to the next event during electrical plant realignment to 161KV.

When Reactor power is reduced 3% to 5%, PROCEED to Events 7 and 8.

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Events 7 and 8.** **- Steam Generator RC-2B Tube Rupture @ 500 gpm on 10 minute ramp.** **- Diesel Generator DG-01 start failure on SIAS.**		
Indications Available: **Pressurizer pressure and level lowering.**		
+2 min	ATCO	RECOGNIZE Pressurizer pressure and level lowering, upward trending Radiation Monitors and MANUALLY TRIP Reactor.
	CRS	DIRECT performance of EOP-00, Standard Post Trip Actions.
Examiner Note: **The following steps are from EOP-00, Standard Post Trip Actions.**		
	ATCO	VERIFY Reactivity Control: [Step 1]
		• VERIFY ALL of the following: [Step 1.a]
		• VERIFY no more than one Regulating or Shutdown CEA NOT inserted.
		• VERIFY Reactor Power is LOWERING.
		• VERIFY Startup Rate is NEGATIVE.
		• MONITOR plant for an uncontrolled RCS Cooldown. [Step 1.b]
	CRS	DETERMINE Reactivity Control criteria SATISFIED.
	BOPO	VERIFY Turbine Trip: [Step 2]
		• VERIFY HP & LP Stop and Intercept Valves CLOSED.
	BOPO	ENSURE all Generator Breakers are tripped: [Step 3]
		• DETERMINE Generator Output Breaker 3451-4 tripped.
		• DETERMINE Generator Output Breaker 3451-5 tripped.
		• DETERMINE Generator Field Breaker 41E/G1F tripped.
	BOPO	VERIFY 4160 V Safeguards Buses 1A3 and 1A4 are ENERGIZED. [Step 4]

Operating Test : NRC	Scenario # 3	Event # 7 & 8			Page 23	of	34
Event Description:	Steam Generator Tube Rupture / Diesel Generator Start Failure						

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE Maintenance of Vital Auxiliaries criteria SATISFIED.
<u>Examiner Note:</u> The following step (Verify Diesel Generators running) is not required until Reactor Coolant System Pressure is less than 1600 psia and PPLS has actuated.		
	BOPO	VERIFY both Diesel Generators RUNNING on Safety Injection Actuation Signal. [Step 5]
		• [CA] DEPRESS DG-01 Emergency Start pushbutton and VERIFY DG-01 running at 900 RPM.
	BOPO	VERIFY 4160 V Non-Safeguards Buses 1A1 and 1A2 are ENERGIZED. [Step 6]
	BOPO	VERIFY 125 VDC Buses 1 and 2 are ENERGIZED. [Step 7]
	BOPO	VERIFY Instrument Air is AVAILABLE: [Step 8]
		• DETERMINE Instrument Air pressure ≥ 90 psig.
		• DETERMINE Instrument Air Compressor CA-1C RUNNING.
	ATCO	VERIFY Component Cooling Water System operation NORMAL: [Step 9]
		• DETERMINE at least one CCW pump RUNNING. [Step 9.a]
		• DETERMINE CCW Pump discharge pressure ≥ 60 psig. [Step 9.b]
		• DETERMINE HCV-438A/B/C/D, CCW to RCP Coolers OPEN. [Step 9.c]
		• DETERMINE at least one Raw Water Pump RUNNING. [Step 9.d]
	CRS	VERIFY RCS Inventory Control criteria satisfied: [Step 10]
		• DETERMINE PZR level NOT between 30% and 70% and NOT TRENDING to between 45% and 60%.
		• [CA] RESTORE Inventory Control by manually controlling Charging and Letdown. [Step 10.1.a]
		• DETERMINE RCS subcooling > 20°F.
	CRS	DETERMINE RCS Inventory Control criteria NOT SATISFIED.

Time	Position	Applicant's Actions or Behavior
	CRS	VERIFY RCS Pressure Control criteria satisfied: [Step 11]
	ATCO	• DETERMINE RCS pressure less than 1600 psia.
		• [CA] VERIFY RCS pressure < 2300 psia and PORV NOT open. [Step 11.1]
		• [CA] When RCS pressure < 1350 psia, PERFORM the following: [Step 11.2]
	ATCO	• [CA] STOP one RCP in each Loop.
		• [CA] DETERMINE RCS pressure < 1600 psia and VERIFY Engineered Safeguards ACTUATED. [Step 11.3]
		• [CA] DETERMINE PPLS relays 86A/PPLS / 86B/PPLS / 86A1/PPLS / 86B1/PPLS have TRIPPED. [Step 11.3.a]
		• [CA] DETERMINE all PPLS relays have TRIPPED. [Step 11.3.b]
		• [CA] DETERMINE VIAS relays 86A/VIAS / 86B1/VIAS / 86B/VIAS / 86A1/VIAS have TRIPPED. [Step 11.3.c]
		• [CA] DETERMINE SIAS relays 86A/SIAS / 86AX/SIAS / 86B1/SIAS / 86B1X/SIAS / 86B/SIAS / 86BX/SIAS / 86A1/SIAS / 86A1X/SIAS have TRIPPED. [Step 11.3.d]
		• [CA] DETERMINE CIAS relays 86A/CIAS / 86B1/CIAS / 86B/CIAS / 86A1/CIAS have TRIPPED. [Step 11.e]
		• [CA] ENSURE required pumps RUNNING [Step 11.3.f]
		• DETERMINE HPSI Pumps SI-2A & SI-2B RUNNING.
		• DETERMINE LPSI Pumps SI-1A and SI-1B RUNNING.
		• DETERMINE Charging Pumps CH-1B RUNNING.
		• [CA] ENSURE acceptable SI flow per Attachment IC-13, SI Flow vs. Pressurizer Pressure. [Step 11.3.g]
	ATCO	• [CA] ENSURE Emergency Boration in progress. [Step 11.3.h]
	CRS	DETERMINE RCS Pressure Control criteria NOT SATISFIED.

Operating Test : NRC	Scenario # 3	Event # 7 & 8	Page 25 of 34
Event Description: Steam Generator Tube Rupture / Diesel Generator Start Failure			

Time	Position	Applicant's Actions or Behavior
<u>Examiner Note</u>: The following steps are from RC-11, Emergency Boration Verification.		
	ATCO	ENSURE the following valves are CLOSED: [Step 1]
		• FCV-269X, Demin Water Makeup Valve
		• FCV-269Y, Boric Acid Makeup Valve
		• HCV-264, CH-4A Recirc Valve
		• HCV-257, CH-4B Recirc Valve
	ATCO	VERIFY all the following valves OPEN: [Step 2]
		• HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
		• HCV-265, CH-11A Gravity Feed Valve
		• HCV-258, CH-11B Gravity Feed Valve
	ATCO	ENSURE all available Boric Acid Pumps RUNNING: [Step 3]
		• CH-4A, Boric Acid Pump
		• CH-4B, Boric Acid Pump
	ATCO	ENSURE all available Charging Pumps RUNNING: [Step 4]
		• CH-1A, Charging Pump is tripped.
		• CH-1B, Charging Pump is RUNNING.
		• CH-1C, Charging Pump is OOS.
	ATCO	ENSURE the following valves are CLOSED: [Step 5]
		• LCV-218-2, VCT Outlet Valve
		• LCV-218-3, Charging Pump Suction SIRWT Isolation Valve
		• HCV-257, CH-4B Recirculation Valve
		• HCV-264, CH-4A Recirculation Valve
	ATCO	DETERMINE Emergency Boration is in progress. [Step 6]

Operating Test : NRC	Scenario # 3	Event # 7 & 8	Page 26 of 34
Event Description: Steam Generator Tube Rupture / Diesel Generator Start Failure			

Time	Position	Applicant's Actions or Behavior
Examiner Note: The following steps continue from EOP-00, Standard Post Trip Actions.		
	CRS	VERIFY Core Heat Removal criteria satisfied: [Step 12]
	ATCO	• DETERMINE RCP NPSH requirements met.
		• DETERMINE at least one RCP operating.
		• DETERMINE Core $\Delta T \leq 10°F$.
CRITICAL TASK STATEMENT		**Stop One Reactor Coolant Pump in Each Loop when Reactor Coolant System Pressure is < 1350 psia, Prior to losing Reactor Coolant Pump Net Positive Suction Head.**
CRITICAL TASK	ATCO	DETERMINE Reactor Coolant System pressure < 1350 psia and PERFORM the following:
	ATCO	• STOP one RCP in each Loop.
	CRS	DETERMINE Core Heat Removal criteria SATISFIED.

NOTE

If Instrument Air to valves HCV-1105, HCV-1106, HCV-1107A/B and HCV-1108A/B is not available, throttling of these valves is not possible. Open or close operation of these valves is possible for a minimum of three cycles.

Time	Position	Applicant's Actions or Behavior
	CRS	VERIFY RCS Heat Removal criteria satisfied:
	BOPO	VERIFY Main Feedwater is restoring SG levels to 35% to 80% NR and 73% to 94% WR. [Step 13]
		• DETERMINE FCV-1101 & FCV-1102 Feed Regulating Valves CLOSED. [Step 13.a]
		• DETERMINE HCV-1105 & HCV-1106 Feed Regulating Bypass Valves ramped to between 40% & 45% OPEN. [Step 13.b]
	BOPO	• PLACE 43/FW switch in OFF. [Step 13.c]
		• ENSURE no more than one Main Feedwater Pump RUNNING. [Step 13.d]
		• ENSURE no more than one Condensate Pump RUNNING. [Step 13.e]

Operating Test : NRC	Scenario # 3	Event # 7 & 8	Page 27 of 34
Event Description: Steam Generator Tube Rupture / Diesel Generator Start Failure			

Time	Position	Applicant's Actions or Behavior
	BOPO	• STOP running Heater Drain Pumps FW-5A, FW-5B, and/or FW-5C. [Step 13.f]
		• ENSURE SG Blowdown Isolation Valves CLOSED. [Step 13.g]
		• HCV-1387A & HCV-1387B
		• HCV-1388A & HCV-1388B
	BOPO	VERIFY Steam Dump and Bypass Valves controlling <u>both</u> of the following: [Step 14]
		• DETERMINE RCS T_{COLD} between 525°F and 535°F.
		• DETERMINE Steam Generator pressure between 850 psia & 925 psia.
	CRS	DETERMINE RCS Heat Removal criteria SATISFIED.
	CRS	VERIFY Containment Isolation criteria satisfied:
	ATCO	VERIFY Normal Containment conditions exist: [Step 15]
		• DETERMINE no unexpected rise in Containment Sump level. [Step 15.a]
		• DETERMINE Containment Area Radiation Monitors NOT in alarm. [Step 15.b]
		• DETERMINE Containment Ventilation Radiation Monitors NOT in alarm. [Step 15.c]
	ATCO	• DETERMINE RM-054B, SG Blowdown Radiation Monitor ALARMING. [Step 15.d]
		• [CA] MINIMIZE spread of contamination: [Step 15.d.1]
	BOPO	• [CA] VERIFY RCV-978, 6th Stage Extraction Isolation Valve CLOSED. [Step 15.d.1.1)]
		• [CA] VERIFY all Blowdown Isolation Valves CLOSED. [Step 15.d.1.2)]
		• [CA] HCV-1387A & HCV-1387B
		• [CA] HCV-1388A & HCV-1388B
	ATCO	• DETERMINE RM-054B, SG Blowdown Radiation Monitor <u>and</u> RM-057, Condenser Off Gas Radiation Monitor TRENDING upward. [Step 15.e]
	CRS	• [CA] DETERMINE if Steam Generator Tube Rupture is in progress: [Step 15.e.1]

Operating Test :	NRC	Scenario #	3	Event #	7 & 8	Page	28	of 34
Event Description:	Steam Generator Tube Rupture / Diesel Generator Start Failure							

Time	Position	Applicant's Actions or Behavior
		• [CA] DIRECT Shift Chemist to perform rapid activity analysis of both SGs. [Step 15.e.1.1)]
	BOPO	• [CA] DETERMINE SG RC-2B has an abnormal rise in level. [Step 15.e.1.2)]
	ATCO	• VERIFY Containment conditions: [Step 15.f]
		• DETERMINE Containment pressure < 3 psig.
		• DETERMINE Containment temperature < 120°F.
	CRS	DETERMINE Containment Integrity criteria NOT SATISFIED.
	CRS	DIAGNOSE event in progress: [Step 16]
		• DETERMINE Reactivity Control requirements met.
		• DETERMINE both DC buses energized.
		• DETERMINE at least one Vital 4160 V Bus energized.
		• DETERMINE at least one Non-Vital 4160 V Bus energized.
		• DETERMINE at least one RCP running.
		• DETERMINE adequate Feedwater flow to at least one SG.
		• VERIFY Pressurizer pressure > 1800 psia with high subcooled margin, normal SG pressure, and no indications of primary to secondary leakage.
		• If not, CONSIDER EOP-04, Steam Generator Tube Rupture.
NOTE Certain events (i.e., LOCA, SGTR, UHE and Loss of All Feedwater) do not require offsite power in order to adequately, mitigate the effects of the accident. For this reason, the LOCA, SGTR, UHE or Loss of All Feedwater procedure may be implemented even if a Loss of Offsite Power has also occurred.		
		• DETERMINE all Safety Function Acceptance Criteria NOT SATISFIED.
		• DETERMINE single event in progress and TRANSITION to EOP-04, Steam Generator Tube Rupture.
Examiner Note: The following steps are from EOP-04, Steam Generator Tube Rupture.		
	CRS	CONFIRM Standard Post Trip Actions have been performed. [Step 1]

Operating Test :	NRC	Scenario #	3	Event #	7 & 8	Page	29	of 34
Event Description:	Steam Generator Tube Rupture / Diesel Generator Start Failure							

Time	Position	Applicant's Actions or Behavior
	CRS	CONFIRM Steam Generator Tube Rupture Diagnosis: [Step 2]
		• VERIFY Safety Function Status Check Acceptance Criteria being satisfied. [Step 2.a]
		• VERIFY CIAS is present and SAMPLE both SGs. [Step 2.c]
	CRS	IMPLEMENT the Emergency Plan. [Step 3]
		• Time: __________
	CREW	MONITOR the Floating Steps. [Step 4]
	CRS	DETERMINE RCS pressure ≤ 1600 psia and VERIFY Engineered Safeguards are ACTUATED: [Step 5]
		• DETERMINE PPLS relays 86A/PPLS / 86B/PPLS / 86A1/PPLS / 86B1/PPLS have TRIPPED. [Step 5.a]
		• DETERMINE SIAS relays 86A/SIAS / 86AX/SIAS / 86B1/SIAS / 86B1X/SIAS / 86B/SIAS / 86BX/SIAS / 86A1/SIAS / 86A1X/SIAS have TRIPPED. [Step 5.b]
		• DETERMINE CIAS relays 86A/CIAS / 86B1/CIAS / 86B/CIAS / 86A1/CIAS relays TRIPPED. [Step 5.c]
		• DETERMINE VIAS relays 86A/VIAS / 86B1/VIAS / 86B/VIAS / 86A1/VIAS relays TRIPPED. [Step 5.d]
	ATCO	OPTIMIZE Safety Injection and Charging flow and PERFORM the following: [Step 6]
		• ENSURE required Safety Injection Pumps RUNNING: [Step 6.a]
		• DETERMINE HPSI Pumps SI-2A & SI-2B RUNNING.
		• DETERMINE LPSI Pumps SI-1A and SI-1B RUNNING.
		• DETERMINE Charging Pumps CH-1B RUNNING.
	ATCO	• DETERMINE Emergency Boration already in progress per RC-11, Emergency Boration Verification. [Step 6.b]
		• ENSURE adequate SI flow per IC-13, Safety Injection Flow vs. Pressurizer Pressure. [Step 6.c]

Operating Test :	NRC	Scenario #	3	Event #	7 & 8	Page	30	of	34
Event Description:	Steam Generator Tube Rupture / Diesel Generator Start Failure								

Time	Position	Applicant's Actions or Behavior
NOTE Main PZR Spray flow will be reduced with less than four-pump operation. Pressure should be controlled using Main and Auxiliary PZR Spray whenever the Plant is placed in a two-pump configuration.		
	ATCO	VERIFY RCP operating parameters: [Step 7]
		• ENSURE at least one RCP stopped if T_{COLD} < 500°F. [Step 7.a]
		• DETERMINE one RCP stopped in each loop when RCS pressure ≤ 1350 psia following SIAS. [Step 7.b]
		• DETERMINE all RCPs STOPPED on low subcooling. [Step 7.c]
		• Time: _______
	CRS	DETERMINE Condenser vacuum greater than 10.92 inches Hg absolute or 19 inches Hg. [Step 8]
NOTE Reducing RCS T_H to less than or equal to 510°F will maintain adequate RCP NPSH and RCS subcooling when RCS pressure is reduced below SG safety valve setpoint of 1000 psia.		
CAUTION When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr. When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.		
	BOPO	COMMENCE a cooldown using both SGs to reduce RCS T_{HOT} to ≤ 510°F per Attachment HR-12, Secondary Heat Removal Operation. [Step 9]
	ATCO	COMMENCE a depressurization of RCS to less than 1000 psia per Attachment PC-11, Pressure Control. [Step 10]
<u>Examiner Note</u>: The following steps are from HR-12, Secondary Heat Removal Operation.		
	BOPO	ENSURE Turbine Control is in MANUAL. [Step 1]
	BOPO	• [CA] DETERMINE Turbine NOT online and GO TO Step 4.

Operating Test :	NRC	Scenario #	3	Event #	7 & 8	Page	31	of	34
Event Description:	Steam Generator Tube Rupture / Diesel Generator Start Failure								

Time	Position	Applicant's Actions or Behavior

NOTES

1. In MANUAL, single arrows are 1% / double arrows are 5% change in OUTPUT value.
2. While Steam Dump and Bypass Control is in Temperature/Pressure Mode, the controllers PC0910 and TC0909_PI will alternate between controls, depending on the higher output signal. A red square outlining the controlling function signify which parameter is in control.
3. Steps 4 through 11 may be performed as needed, and in any order.

CAUTIONS

When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr.
When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.

CRITICAL TASK STATEMENT		**Reduce and Maintain RCS T_{HOT} ≤ 510°F to Maintain SG Pressure Below Safety Valve Setpoint of 1000 psia.**
CRITICAL TASK	BOPO	DETERMINE Steam Dump and Bypass (SD&B) available and CONTROL RCS temperature with a single SD&B Valve. [Step 4]
		• DEPRESS Valve Toggle to SELECT valve to be operated: [Step 4.a]
		• PCV-910 / TCV-909-1 / TCV-909-2 / TCV-909-3 / TCV-909-4
		• PLACE Controller for selected valve in MANUAL. [Step 4.b]
		• PUSH UP and DOWN arrows to ADJUST Controller Output. [Step 4.c]
		• When no longer required, PLACE Controller for selected valve in AUTO. [Step 4.d]

<u>**Examiner Note**</u>: **The following steps are from PC-11, Pressure Control. PC-12, RCS Pressure-Temperature Limits (graph), is maintained on a Control Room hardcopy.**

CAUTION

A charging header flow path must be maintained at all times.

	ATCO	MAINTAIN RCS pressure per the PC-12, RCS Pressure-Temperature Limits graph: [Step 1]
		• DETERMINE Steps N/A due to RCS pressure. [Step 1.a to 1.d]

CRITICAL TASK	ATCO	• OPERATE the following to CONTROL Auxiliary Spray flow and REDUCE RCS pressure to < 1000 psia: [Step 1.e]
		• HCV-240, PZR Auxiliary Spray Isolation Valve • HCV-249, PZR Auxiliary Spray Isolation Valve • HCV-238, Loop 1 Charging Isolation Valve • HCV-239, Loop 2 Charging Isolation Valve
		• If HPSI Stop and Throttle criteria is met, CONTROL Pressurizer level using Charging, Letdown, and/or HPSI flow per IC-11, Inventory Control. [Step 1.f]
		• CONTROL RCS heat removal per HR-12, Secondary Heat Removal Operation. [Step 1.g]
	ATCO	MAINTAIN RCS Pressure per PC-12, RCS Pressure-Temperature Limits by performing ANY of the following: [Step 11]
		• CONTROL RCS Heat Removal per HR-12, Secondary Heat Removal Operation. [Step 11.a]
		• CONTROL Pressurizer Heaters and Spray per PC-11 Pressure Control. [Step 11.b]
		• If HPSI Stop and Throttle criteria are met, CONTROL Charging, Letdown, and/or HPSI flow per IC-11, Inventory Control. [Step 11.c]
	BOPO	If feeding through Feed Ring, MAINTAIN SG levels 44% to 85% NR (77% to 94% WR) using Main Feedwater or FW-54. [Step 12]
		• FEED SGs using HR-15, Main Feed Pump Operation **or** HR-16, FW-54 Operation. [Step 12.a]
		• CONTROL feed flow per HR-11, Manual Feed Control. [Step 12.b]
	ATCO	PERFORM the following to PLACE RM-064, Main Steam Line Radiation Monitor in service at AI-33C. [Step 13]
		• PLACE Main Steam Line A/B Enable Switch for HCV-921 and HCV-922 in ON. [Step 13.a]
		• PLACE Main Steam Line A/B Mode Selector Switch in AUTO. [Step 13.b]
	CRS	DETERMINE Steam Generator RC-2B has the tube rupture. [Step 14]
	BOPO	PERFORM the following to MINIMIZE spread of contamination: [Step 15]

Operating Test : NRC	Scenario # 3	Event # 7 & 8	Page 33 of 34
Event Description: Steam Generator Tube Rupture / Diesel Generator Start Failure			

Time	Position	Applicant's Actions or Behavior
		• CONTACT Auxiliary Operator to POSITION following valves: [Step 15.a]
		• ENSURE HC-2509, SAMPLE DRAIN TO DRAIN HEADER is OPEN at AI-107, Room 60. [Step 15.a]
		• ENSURE HC-2508, SAMPLE DRAIN TO CONDENSER C.W. TUNNEL is CLOSED at AI-107, Room 60. [Step 15.b]
		• ENSURE FW-268, CONDENSATE DUMP VALVE LCV-1193 OUTLET ISOLATION VALVE, is CLOSED at Turbine Building Mezzanine. [Step 15.c]
		• ENSURE FW-266, CONDENSATE DUMP VALVE LCV-1193 BYPASS VALVE, is CLOSED at Turbine Building Mezzanine. [Step 15.d]
	BOPO	When RCS T_{HOT} is ≤ 510°F, ISOLATE SG RC-2B. [Step 16]
		• ISOLATE SG RC-2B per HR-20, Isolate/Restore Steam Generator B. [Step 16.a]

Examiner Note: The following steps are from HR-20, Isolate/Restore Steam Generator B.

NOTE

RCS Heat Removal takes precedence over isolation of a S/G with a tube rupture.

CRITICAL TASK STATEMENT		**Isolate the Affected Steam Generator with a Tube Rupture to Minimize Spread of Contamination.**
CRITICAL TASK	BOPO	PERFORM the following to isolate Steam Generator RC-2B: [Step 1]
		• ENSURE all the following valves are CLOSED: [Step 1.a]
	BOPO	• **CLOSE HCV-1042A, RC-2B MSIV**. • VERIFY HCV-1042C, RC-2B MSIV Bypass Valve CLOSED. • VERIFY MS-292, Air Assisted Main Steam Safety Valve CLOSED. • **CLOSE FCV-1102, RC-2B Feed Regulating Valve.** • **CLOSE HCV-1106, Feed Regulating Bypass Valve.** • **CLOSE HCV-1385, RC-2B Feed Header Isolation Valve**. • **CLOSE HCV-1104, Feed Regulating Block Valve**. • VERIFY HCV-1387A, Blowdown Isolation Valve CLOSED. • VERIFY HCV-1387B, Blowdown Isolation Valve CLOSED. • **CLOSE HCV-1108A, AFW Isolation Valve**. • **CLOSE HCV-1108B, AFW Isolation Valve**.

Operating Test : NRC	Scenario # 3	Event # 7 & 8	Page 34 of 34
Event Description: Steam Generator Tube Rupture / Diesel Generator Start Failure			

Time	Position	Applicant's Actions or Behavior
		• CONTACT Auxiliary Operator to CLOSE MS-298, Steam Valves HCV-1041A & 1042A Packing Leakoff Line Isolation Valve in Room 81. [Step 1.b]
		• If sampling is NOT in progress, CLOSE both Sample Valves: [Step 1.c]
		• HCV-2506A, RC-2A Blowdown Sample Isolation Valve
		• HCV-2506B, RC-2B Blowdown Sample Isolation Valve
	BOPO	• PERFORM the following to CLOSE YCV-1045B: [Step 1.d]
		• DETERMINE Isolation Valve YCV-1045B OVERRIDE SW in OVERRIDE. [Step 1.d.1)]
		• DETERMINE SG RC-2B STM TO FW-10 HDR A ISOLATION VALVE YCV-1045B in CLOSE. [Step 1.d.2)]

NOTE

Air accumulators will maintain the valve in a closed position for 30 minutes after a loss of Instrument Air.

		• CONTACT Auxiliary Operator HANDJACK YCV-1045B, MAIN STEAM LINE "B" TO AUX FEEDWATER PUMP FW-10 SUPPLY VALVE to CLOSE in Room 81. [Step 1.d.3)]
	CRS	• RECORD time Steam Generator RC-2B ISOLATED. [Step 1.e]
		• Time: ________
	CRS	VERIFY RC-2B is most affected SG per Attachment HR-18, Most Affected Steam Generator Determination. [Step 2]

When Steam Generator RC-2B is isolated, TERMINATE the scenario.

Facility: Fort Calhoun Station	Scenario No.: 4	Op Test No.: Dec 2015 NRC	
Examiners:		Operators:	

Initial Conditions: MODE 2 at ~1% power - RCS Boron is 959 ppm (by sample).

Turnover: Continue in OP-2A, Plant Startup and OI-RR-1, Reactor Regulating System Operation to raise Reactor power to 7% power. When MODE 1 is entered, place Steam Dump and Bypass Valves in AUTO per OI-MS-1A, Main Steam System Operation.

Critical Tasks:
- Manually Trip Reactor to meet Core Design Criteria of Lowering Reactor Power and Negative Startup Rate to Verify Reactivity Control Established During ATWS Event. (Event 5)
- Isolate the Affected Steam Generator to Prevent Excess Plant Cooldown and Reactivity Additions Prior to Steam Generator Level = 0% Wide Range Level. (Event 7)

Event No.	Malf. No.	Event Type*	Event Description
1 +20 min		R (ATCO) N (BOPO, CRS)	Raise Power Using Control Rods to 7% per OP-2A, Plant Startup. Place Steam Dump and Bypass Valves in AUTO per OI-MS-1A.
2 +30 min		C (BOPO, CRS) TS (CRS)	Raw Water Pump Discharge Line Leak Upstream of HCV-2879A in the Auxiliary Building.
3 +45 min		I (BOPO, CRS) TS (CRS)	Inadvertent Channel B Auxiliary Feedwater Actuation Signal On Steam Generator RC-2A.
4 +60 min		C (ATCO, CRS) TS (CRS)	Loss of Instrument Bus AI-40A. Loss of Letdown and Pressurizer Level Control.
5 +60 min		M (ATCO, BOPO, CRS)	Reactor Coolant Pump RC-3A Trip. Automatic Reactor Trip Failure, Manual Reactor Trip Required.
6 +65 min		C (BOPO)	Instrument Air Compressor CA-1B and CA-1C Trip. Bearing Cooling Water Pump AC-9B Trip.
7 +70 min		M (ATCO, BOPO, CRS)	Steam Line Break inside Containment on RC-2A @ 1% Severity on 5 Minute Ramp.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
1	Malfunctions after EOP entry (1-2)
3	Abnormal events (2-4)
2	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO SUMMARY NRC 4

The crew will assume the shift at 1% power and raise Power to ~7% using CEAs per OP-2A, Plant Startup, Attachment 4, Hot Standby, MODE 2 to Minimum Load, MODE 1 and OI-RR-1, Reactor Regulating System Normal Operation, Attachment 7, Manual Sequential Mode Checklist. When MODE 1 is entered, temperature control is placed in AUTO per OI-MS-1A, Main Steam System Operation, Attachment 5, Steam Dump and Bypass Manual Control Function.

The next event is a Raw Water Pump AC-10C discharge line leak in the Auxiliary Building upstream of HCV-2879A. The crew enters AOP-18, Loss of Raw Water, and must observe Raw Water System indications in order to determine the location of the leak. Once identified, the leak is isolated per AOP-18, Attachment C, Equipment Isolation, and Raw Water flow is restored. The SRO will refer to Technical Specification LCO 2.4(1) – Raw Water Header.

The next event is an inadvertent Channel B Auxiliary Feedwater Actuation Signal (AFAS) on Steam Generator RC-2A. The crew responds per ARP-AI-66B/A66B, Window 41 and verifies Auxiliary Feedwater Pumps FW-6 and FW-10 are running. Once it is determined the AFAS was inadvertent, AOP-23, Reset of Engineered Safeguards, Section IX, Reset of Inadvertent AFAS, is performed. The SRO will refer to Technical Specification LCO 2.15.1(1) – Automatic Initiation Steam Generator Water Level Logic Subsystem B.

When plant conditions are stable, a loss of Instrument Bus AI-40A occurs. The crew enters AOP-16, Loss of Instrument Bus Power, Section I, Loss of Instrument Bus Power, then Section II, Loss of Instrument Bus AI-40A. Actions include isolating Letdown, transferring Pressurizer Level Control, and operating Charging Pumps as required. Electrical Maintenance is notified and the Plant remains in this configuration through the end of the Scenario. The SRO will refer to Technical Specification LCO 2.15.2 – Reactor Protective System Logic and Trip Initiation and LCO 2.7(1) – 120 VAC Instrument Bus A.

The next event is a trip of Reactor Coolant Pump RC-3A. The crew should recognize failure of the Reactor Protection System Low Flow trips and manually trip the Reactor and enter EOP-00, Standard Post Trip Actions. When the Reactor is tripped, a 1% severity Steam Line Break inside Containment initiates on a 5 minute ramp. Due to the small size of this break, RCS pressure remains above the SIAS initiation setpoint of 1600 psia. The crew will transition to EOP-05, Uncontrolled Heat Extraction, and identify and isolate the affected Steam Generator RC-2A.

The event is complicated by a trip of the running and standby Instrument Air Compressors CA-1B and CA-1C and a trip of Bearing Water Cooling Pump AC-9B. The crew must restore a Bearing Cooling Water Pump and Instrument Air Compressor while in EOP-00. The scenario is terminated when Steam Generator RC-2A is isolated per HR-19, Isolate/Restore Steam Generator A while in EOP-05.

Risk Significance:

• Failure of risk important system prior to trip:	Loss of Raw Water System Header Loss of Instrument Bus
• Risk significant core damage sequence:	Automatic Reactor Trip Failure Steam Line Break Inside Containment
• Risk significant operator actions:	Isolate Raw Water East Header Manually Trip Reactor Restore Instrument Air Isolate Affected Steam Generator

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

RESET to IC-122 and LOAD & EXECUTE NRC 4.sce for NRC Scenario 4.

Preset item – Event 5 – Reactor Fails to Trip Automatically, CB-4 Trip Button Works

Type	Item	Value	Condition
Expert	RPS02 RPS01 RPS03 RPS04 P6A_026_1 P6B_028_1 ANN-P6A_0026R1C_Fail ANN-P6A_0027R1C_Fail ANN-P6B_0026R5C_Fail ANN-P6B_0027R5C_Fail ANN-P6B_0025R5C_Fail ANN-P6A_0025R1C_Fail H_P6A_022A_1 H_P6B_024A_1	Energized Energized Energized Energized True True Alarm Off Alarm Off Alarm Off Alarm Off Alarm Off Alarm Off True True	Scenario Event: “Rx Fail to Trip, CB-4 works”

Event 2 – Raw Water leak in the Auxiliary Building

Type	Item	Value	Condition
Malfunction	RWS02B	25	When directed by examiner, trigger/activate this event. Scenario Event: “Raw Water Leak in Aux Building”

Event 3 – Inadvertent AFAS on RC-2A

Type	Item	Value	Condition
Expert	B_RC_2A_AFWS	True	When directed by examiner, trigger/activate this event. Scenario Event: “Inadvertent AFAS”

Event 4 – Loss of Instrument Bus AI-40A

Type	Item	Value	Condition
Malfunction	EDA08	10	When directed by examiner, trigger/activate this event. Scenario Event: “Loss of AI-40A”

Event 5 – ‘A’ Reactor Coolant Pump Trips

Type	Item	Value	Condition
Malfunction	BUS_1A1_5_BKR_TRIP	True	When directed by examiner, trigger/activate this event. Scenario Event: “A RCP Trip”

Event 6 – Following RX Trip, Loss of Instrument Air and Bearing Cooling Water

Type	Item	Value	Condition
Remote	BCW_AC9B_BRKR	Trip	Event is triggered automatically after reactor trip. Scenario Event: "Loss of Inst Air and Bearing Water"
Malfunction	BUS_1B3A_4A_2_BKR_Trip BUS_1B4B_4_BKR_TRIP	True True	

Event 7 – Main Steam Break Inside Containment

Type	Item	Value	Condition
Malfunction	SGN01A	1%, ramp = 300 sec	Event is triggered automatically after reactor trip. Scenario Event: "Steam Line Break in Containment"

Booth Operator:	**INITIALIZE to IC-122 and LOAD NRC 4.sce. ENSURE all Simulator Annunciator Alarms are ACTIVE. ENSURE all Control Board Tags are removed. ENSURE Bearing Water Pump AC-9B running. ENSURE Air Compressors CA-1B & CA-1C alignment: 1 in Standby, 1 running. PLACE Steam Dump & Bypass Controllers in Manual. ENSURE Lead Examiner has AFAS Keys 55 & 57 for Event 3. ENSURE Lead Examiner has RPS Trip Unit Keys 1-12 for Event 4. ENSURE Operator Aid Tags reflect current boron conditions. ENSURE Reactivity Briefing Sheet printout provided with Turnover. ENSURE Middle-of-Life Thumb Rule Sheet provided with Turnover. ENSURE Steam Dump and Turbine Bypass System in MANUAL control. ENSURE Control Room hard copy for OI-RR-1 is CLEAN. ENSURE CEA Regulating Group 4 @ 72". ENSURE procedures in progress provided to crew in Briefing Room: - COPY of ReMA Data for Reactor Power Ascension. - COPY of OP-2A, Plant Startup, Attachment 4, Hot Standby, MODE 2 to Minimum Load, MODE 1, INITIALED through Step 6.b. - COPY of OI-RR-1, Reactor Regulating System Normal Operation, Attachment 7, Manual Sequential Mode Checklist. - Copy of OI-MS-1A, Main Steam System Operation, Attachment 5, Steam Dump and Bypass Manual Control Function, INITIALED through Prerequisites and Steps 1.a & 2.a.**

Control Room Annunciators in Alarm:

A9-B-1(U) – TURBINE DIFFERENTIAL EXPANSION
A10-A-1(U) – MOTOR SUCT PUMP RUNNING OR NOT IN AUTO
A10-B-6(L) – 43/FW TRANSFER SWITCH OFF-AUTO
A11-A-4(U) – HEATER 5A HEATER HI-LO
A11-A-4(L) – HEATER 5B HEATER HI-LO
A11-B-3(U) – HEATER DRAIN TANK LEVEL HI-LO
A20-D-5 – LOSS OF LOAD CHANNEL TRIP BYPASSED
A20-E-4 – HIGH POWER RATE OF CHANGE TRIP ENABLED
A21-B-1(U) – HC-909 INHIBIT
A21-C-6(U) – HEATING STEAM PRESS LO
AI-66B/A66B-Window 3 – FW-10 TURBINE DRIVEN FEEDWATER PUMP RUNNING

Procedure List
Event 1: OP-2A, Plant Startup, Attachment 4, Hot Standby, MODE 2 to Minimum Load, MODE 1
Event 1: OI-RR-1, Reactor Regulating System Normal Operation, Attachment 7, Manual Sequential Mode Checklist
Event 1: OI-MS-1A, Main Steam System Operation, Attachment 5, Steam Dump and Bypass Manual Control Function
Event 2: AOP-18, Loss of Raw Water
Event 3: ARP-AI-66B/A66B, Window 41, AFWS STEAM GEN RC-2A CHANNEL B ACTUATED
Event 3: AOP-23, Reset of Engineered Safeguards, Section IX, Reset of Inadvertent AFAS
Event 3: OI-AFW-2, Auxiliary Feedwater System Actuation and Bypass, Attachment 1, Bypass of the Auxiliary Feedwater Actuation Signal (AFAS) (Modes 1 or 2)
Event 3: OI-AFW-2, Auxiliary Feedwater System Bypass, Table 2, AFAS Logic Subsystem Channel Bypass Switch Alignment
Event 4: AOP-16, Loss of Instrument Bus Power, Section I - Loss of Instrument Bus Power
Event 4: AOP-16, Loss of Instrument Bus Power, Section II, Loss of Instrument Bus AI-40A
Event 5: EOP-00, Standard Post Trip Actions
Event 7: EOP-05, Uncontrolled Heat Extraction

Operating Test :	NRC	Scenario #	4	Event #	1	Page	7	of	36
Event Description:	Raise Reactor Power								

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, RESPOND to requests from Control Room.		
Examiner Note: This Scenario Section contains guidance for the following Operator actions: **1. Raising power per OP-2A.** **2. Withdrawing Control Rods per OI-RR-1.** **3. Control of Steam Dumps and Bypass per OI-MS-1A.**		
Examiner Note: The following steps are from OP-2A, Plant Startup, Attachment 4, Hot Standby, MODE 2 to Minimum Load, MODE 1, Step 6.		
		RAISE Reactor power to ~ 10% while performing the following: [Step 6]
		• DETERMINE Main Feedwater Pump FW-4B is RUNNING. [Step 6.a]
		• DETERMINE Auxiliary Feedwater SECURED. [Step 6.b]
		• MAINTAIN RCS temperature 527°F to 535°F using Steam Dump and Bypass Valves. [Step 6.c]
		• Prior to exceeding 15% power, VERIFY Secondary Chemistry parameters. [Step 6.d]
		• Prior to exceeding 15% power, VERIFY Condensate Pump Discharge Suspended Solids within specification. [Step 6.e]
		• PERFORM daily grab samples for Secondary activity or DECLARE RM-057 Radiation Monitor in service. [Step 6.f]
NOTE This step is performed to ensure that the DVM NI indication is greater than or equal to actual power.		
		• When power is stable at approximately 10% (as indicated by highest of NI and ΔT power), ADJUST RPS power per OI-NI-1. [Step 6.g]
		• OPEN MFW Isolation Valves HCV-1103 & HCV-1104. [Step 6.h]
Examiner Note: The following steps are from OI-RR-1, Reactor Regulating System Normal Operation, Attachment 7, Manual Sequential Mode Checklist, and is maintained as a Control Room hard copy.		
	ATCO	ENSURE an out-of-scan CEA is NOT selected as Target Rod on CB-4. [Step 1]
	ATCO	VERIFY alarm REGULATING GROUP WITHDRAWAL PROHIBIT is clear. [Step 2]

Operating Test : NRC	Scenario # 4	Event # 1	Page 8	of 36			
Event Description:	Raise Reactor Power						

Time	Position	Applicant's Actions or Behavior
	ATCO	PLACE Rod Control Mode Selector Switch in Manual Sequential (MS). [Step 3]

NOTE

Continuous CEA motion shall be avoided whenever possible. CEA motion should be stopped at least every 33 inches (43 seconds of continuous CEA motion) to check position of CEAs in Group and Reactor response.

Time	Position	Applicant's Actions or Behavior
	ATCO	MOVE Manual Rod Control Switch to **RAISE** or **LOWER** as required. [Step 4]
	ATCO	DETERMINE appropriate Group Overlap during WITHDRAWAL is N/A. [Step 5]
	ATCO	When CEAs are at desired position, RELEASE Manual Rod Control Switch. [Step 6]
	ATCO	VERIFY all CEA motion has stopped. [Step 7]
	ATCO	If additional movement is required, GO TO Step 3. [Step 10]
	ATCO	When completed, PLACE Rod Control Mode Selector Switch in OFF. [Step 11]

<u>Examiner Note</u>: The following steps are from OI-MS-1A, Main Steam System Operation, Attachment 5, Steam Dump and Bypass Manual Control Function.

Time	Position	Applicant's Actions or Behavior
	BOPO	If operating all Steam Dump and Bypass Valves via the Pressure Controller (PC0910), PERFORM the following (SEC/MS/SD&B Control): [Step 1]
		• DETERMINE PC0910, STM DMP & BYP PRESS CONTROL, in MANUAL control. [Step 1.a]
		• As required, ADJUST output to Steam Dump and Bypass Valves. [Step 1.b]

Operating Test :	NRC	Scenario #	4	Event #	1	Page	9	of	36
Event Description:	Raise Reactor Power								

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When power has been raised approximately 3%, and prior to transitioning to the next event, CONTACT the Control room as the Shift Manager and direct placing Steam Dump and Turbine Bypass System (pressure and temperature control) in AUTO.**		
	BOPO	• If desired to transfer back to AUTO at Output that has been selected, COMPLETE the following on Digital Control System: [Step 1.c]
		• PLACE PC0910 in LOCAL. [Step 1.c.1)]
		• ADJUST PC0910 SPT to approximately match PC0910 MEAS value. [Step 1.c.2)]
		• PLACE PC0910 back in AUTO. [Step 1.c.3)]
	BOPO	If operating all Steam Dump and Bypass Valves via the Temperature Controller (TC0909_PI), PERFORM the following (SEC/MS/SD&B Control): [Step 2]
		• DETERMINE TC0909_PI, STM DMP & BYP TEMP CONTROL, in MANUAL control. [Step 2.a]
		• As required, ADJUST output to Steam Dump and Bypass Valves. [Step 2.b]
+20 min	BOPO	• If desired to transfer back to AUTO at Output that has been selected, COMPLETE the following on Digital Control System: [Step 2.c]
		• PLACE TC0909_PI in LOCAL. [Step 2.c.1)]
		• ADJUST TC0909_PI SPT to approximately match TC0909_PI MEAS value. [Step 2.c.2)]
		• PLACE TC0909_PI back in AUTO. [Step 2.c.3)]
When Reactor power is raised 3% to 5%, PROCEED to Event 2.		

Operating Test : NRC	Scenario # 4	Event # 2	Page 10 of 36
Event Description: Raw Water Pump Discharge Line Leak			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 2. - Raw Water Pump discharge line leak upstream of HCV-2879A.		
Indications Available: **CB-1,2,3/A1 – RAW WATER SUPPLY HEADER FLOW LO** **CB-1,2,3/A1 – RAW WATER SUPPLY HEADER PRESS LO** **All Raw Water System 10 psig and 25 psig pressure indicating lights OUT**		
+30 sec	ATCO	RESPOND to Annunciator Response Procedures.
	ATCO	INFORM CRS of Raw Water System low pressure and low flow.
Examiner Note: ATCO may "Operate to Mitigate" per OPD 4-09 and START another Raw Water Pump.		
	CRS	REFER to AOP-18, Loss of Raw Water.
Examiner Note: The following steps are from AOP-18, Loss of Raw Water.		
	ATCO	DETERMINE Raw Water Pump AC-10C is RUNNING. [Step 4.1]
Booth Operator: If not already contacted, 1 minute after Control Room Receipt of alarms, REPORT as Auxiliary Building Operator that he observed water flowing out of room 18, and he is going in to investigate. WAIT 30 seconds and REPORT Raw Water System leak in room 18, upstream of HCV-2879A/B on the header side of the system.		
	ATCO	If Raw Water System rupture is indicated, DIRECT Operators to identify location of leak: [Step 4.2]
		• OBSERVE East RW Header Flow FIC-2890 OSCILLATING. • OBSERVE West RW Header Flow FIC-2891 OSCILLATING. • OBSERVE RW Pump(s) Current OSCILLATING. • OBSERVE RW System Pressure PIC-2892 OSCILLATING. • OBSERVE RW Pump Room Water Level LIC-2889/LC-2825 Level NORMAL.

Operating Test : NRC	Scenario # 4	Event # 2	Page 11 of 36
Event Description: Raw Water Pump Discharge Line Leak			

Time	Position	Applicant's Actions or Behavior
	ATCO	DETERMINE Raw Water vault flooding is NOT occurring. [Step 4.3]
	ATCO	DETERMINE Raw Water leak in Auxiliary Building and PERFORM the following: [Step 4.4]
		• ENSURE only one Raw Water Pump RUNNING. [Step 4.4.a]
		• IMPLEMENT Attachment C, Equipment Isolation. [Step 4.4.b]
	ATCO	DETERMINE CCW temperature ≤ 110°F. [Step 4.5]
	CRS	IMPLEMENT the Emergency Plan. [Step 4.6]
Examiner Note: The following steps are from AOP-18, Attachment C, Equipment Isolation.		
	CRS	If leak is on Raw Water System, GO TO Step 8. [Step 1]
NOTE The leak isolation Steps 8 through 15 may be performed in any logical order.		
	ATCO	DETERMINE leak is NOT on any of the following: [Step 8]
		• WEST RW header • AC-12A, Raw Water Strainer • AC-1C, RW Heat Exchanger
	ATCO	DETERMINE leak is on East Raw Water Header and PERFORM the following to ISOLATE Header: [Step 9]
		• PLACE AC-10D, Raw Water Pump, in PULL-TO-LOCK. [Step 9.a]
	ATCO	• CLOSE all Raw Water Header Isolation Valves: [Step 9.b]
		• CLOSE HCV-2876A. • CLOSE HCV-2876B. • CLOSE HCV-2894. • CLOSE HCV-2879A. • CLOSE HCV-2879B. • CLOSE HCV-2883A. • CLOSE HCV-2883B.

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When contacted, REPORT RW-145 is CLOSED. When contacted, EXECUTE local actions and report handjacks applied to Raw Water System Valves as directed.		
	ATCO	• Locally CLOSE RW-145, RAW WATER STRAINER AC-12B BACKWASH VALVE HCV-2805B OUTLET ISOLATION VALVE in RW Vault. [Step 9.c]
	CRS	• DETERMINE leak is isolated and one Raw Water Pump RUNNING. [Step 9.d]
<u>Examiner Note</u>: The following steps continue from AOP-18.		
	CRS	DETERMINE Raw Water System restored to service. [Step 4.8]
	CRS	EVALUATE Technical Specification LCO 2.4, Containment Cooling
		• LCO 2.4.(1).a.iv. – Raw Water Header
		• CONDITION 2.4.(2).d – Raw Water Header inoperable. • ACTION 2.4.(2).d – RESTORE Raw Water Header within 24 hours <u>OR</u> PLACE Reactor in HOT SHUTDOWN condition within 12 hours.
When Raw Water System is realigned and Technical Specifications have been addressed, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	4	Event #	3	Page	13	of	36
Event Description:	Inadvertent Auxiliary Feedwater Actuation Signal								

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 3. - Inadvertent Auxiliary Feedwater Actuation Signal.		
Indications Available: **AI-66B/A66B – AFWS STEAM GEN RC-2A CHANNEL B ACTUATED** **AI-66B/A66B – FW-10 TURBINE DRIVEN FEEDWATER PUMP RUNNING (~30 seconds later)** **AI-66B/A66B – FW-10 TURBINE OIL PUMP RUNNING (~30 seconds later)**		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	BOPO	INFORM CRS of Auxiliary Feedwater Actuation Signal initiation.
Examiner Note: BOPO may "Operate to Mitigate" per OPD 4-09 and CLOSE HCV-1107A and HCV-1107B to stop FW-10, Turbine Driven Auxiliary Feedwater Pump.		
	CRS	REFER to ARP-AI-66B/A66B, Window 41, AFWS STEAM GEN RC-2A CHANNEL B ACTUATED.
Examiner Note: The following steps are from ARP-AI-66B/A66B, Window 41 – AFWS STEAM GEN RC-2A CHANNEL B ACTUATED.		
	BOPO	CHECK A/B/LI-911, Steam Generator RC-2A Level at AI-66A and AI-66B. [Step 1]
		• DETERMINE SG level LI-911A at Panel AI-66A DEENERGIZED.
		• DETERMINE SG level LI-911B at Panel AI-66B NORMAL.
Booth Operator: When contacted, REPORT LI-911D, RC-2A level at AI-179 is ~ 64% and LI-911C, RC-2A pressure is ~ 884 psia (or as indicated).		
	BOPO	DISPATCH Operator to check C/D/LI-911, RC-2A Level at AI-179. [Step 2]
	BOPO	DETERMINE Steam Generator Wide Range level is > 32%. [Step 3]
	CRS	DETERMINE AFAS initiation is inadvertent and IMPLEMENTS AOP-23, Reset of Engineered Safeguards, Section IX, Reset of Inadvertent AFAS. [Step 4]

Operating Test :	NRC	Scenario #	4	Event #	3	Page	14	of	36
Event Description:	Inadvertent Auxiliary Feedwater Actuation Signal								

Time	Position	Applicant's Actions or Behavior
	CRS	REFER to Technical Specification LCOs 2.14 and 2.15. [Step 5]
	CRS	EVALUATE Technical Specification LCO 2.15.1, Instrumentation and Control Systems
		• LCO 2.15.1.(3) – Automatic Initiation Steam Generator Water Level Logic Subsystem B (Table 2.3 / Function 5).
		• CONDITION 2.15.1.(3) – Logic Subsystem B inoperable • ACTION 2.15.1.(3) – RESTORE inoperable channel within 48 hours <u>OR</u> PLACE Reactor in HOT SHUTDOWN condition within 12 hours.
<u>Examiner Note</u>:	**The following steps are from AOP-23, Reset of Engineered Safeguards, Section IX, Reset of Inadvertent AFAS.**	
	CRS	DETERMINE the AFAS is inadvertent. [Step 4.1]
	CRS	REFER to the following Technical Specifications: [Step 4.2]
		• LCO 2.5, Steam and Feedwater Systems
		• LCO 2.15, Instrumentation and Control Systems
<u>Examiner Note</u>:	**Entry into Technical Specification LCO 2.5.(1).d is required until FW-10, TDAFW Pump is reset and returned to AUTO at the end of this event.**	
	CRS	EVALUATE Technical Specification LCO 2.5, Steam and Feedwater Systems
		• LCO 2.5.(1) – Two AFW Trains OPERABLE
		• CONDITION 2.5.(1).d – Both AFW Trains inoperable • ACTION 2.5.(1).d – RESTORE one train to OPERABLE status immediately.
	BOPO	ENSURE both of the following valves in AUTO: [Step 4.3]
		• DETERMINE FCV-1368, FW-6 Recirc Valve in AUTO. • DETERMINE FCV-1369, FW-10 Recirc Valve in AUTO.

Operating Test :	NRC	Scenario #	4	Event #	3	Page	15	of	36
Event Description:	Inadvertent Auxiliary Feedwater Actuation Signal								

Time	Position	Applicant's Actions or Behavior
	BOPO	PLACE control switches for the following AFW Isolation Valves in CLOSE: [Step 4.4]
		• PLACE HCV-1107A in CLOSE. • PLACE HCV-1107B in CLOSE. • PLACE HCV-1108A in CLOSE. • PLACE HCV-1108B in CLOSE.
	CRS	BYPASS affected logic subsystem per OI-AFW-2, Auxiliary Feedwater System Actuation and Bypass. [Step 4.5]
<u>Examiner Note</u>:		**The following steps are from OI-AFW-2, Auxiliary Feedwater System Actuation and Bypass, Attachment 1, Bypass of the Auxiliary Feedwater Actuation Signal (AFAS) (Modes 1 or 2).**
	BOPO	DETERMINE AFAS is aligned for automatic initiation. [Step 2]
	BOPO	DETERMINE plant is in Mode 1. [Step 3]
	CRS	DETERMINE if an Instrument Channel or a Logic Subsystem Channel is to be bypassed. [Step 1]
		• DETERMINE an Instrument Channel will NOT be bypassed. [Step 1.a]
		• DETERMINE a Logic Subsystem Channel will be bypassed and GO TO Step 3. [Step 1.b]
	CRS	If a Logic Subsystem Channel of AFAS is to be bypassed, COMPLETE the following: [Step 3]
	SM/CRS	• LOG entry into Technical Specification 2.15.1(3), 48 hour LCO.

Operating Test : NRC	Scenario # 4	Event # 3	Page 16 of 36
Event Description: Inadvertent Auxiliary Feedwater Actuation Signal			

Time	Position	Applicant's Actions or Behavior

NOTE

The following alarms are expected depending on the Logic Subsystem Channel that is bypassed:

- **AFWS RC-2A CH A MATRIX TS-A/RC-2A/AFWS TEST SWITCH OFF NORM** (AI-66A, Window 24)
- **AFWS RC-2B CH A MATRIX TS-A/RC-2B/AFWS TEST SWITCH OFF NORM** (AI-66A, Window 25)
- **AFWS OVERRIDE SWITCH A/OR-RC-2A/AFWS OFF NORMAL** (AI-66A, Window 29)
- **AFWS OVERRIDE SWITCH A/OR-RC-2B/AFWS OFF NORMAL** (AI-66A, Window 30)
- **HCV-1107A & B AFWS OVERRIDE SWITCH CH A OR B OFF NORM** (AI-66A, Window 35)
- **AFWS RC-2A CH B MATRIX TS-B/RC-2A AFWS TEST SWITCH OFF NORM** (AI-66B, Window 21)
- **AFWS RC-2B CH B MATRIX TS-B/RC-2B/AFWS TEST SWITCH OFF NORM** (AI-66B, Window 22)
- **AFWS OVERRIDE SWITCH B/OR-RC-2A/AFWS OFF NORMAL** (AI-66B, Window 26)
- **AFWS OVERRIDE SWITCH B/OR-RC-2B/AFWS OFF NORMAL** (AI-66B, Window 27)
- **HCV-1108A & B AFWS OVERRIDE SWITCH CHA OR B OFF NORMAL** (AI-66A, Window 32)

Time	Position	Applicant's Actions or Behavior
	BOPO	BYPASS selected Logic Subsystem using Table 2, AFAS Logic Subsystem Bypass Switch Alignment, and RECORD as left information in appropriate slots. [Step 3.b]

Examiner Note: The following steps are from OI-AFW-2, Table 2, AFAS Logic Subsystem Channel Bypass Switch Alignment.

Table 2 - AFAS Logic Subsystem Channel Bypass Switch Alignment

Bypassing Channel	Panel No.	Switch	Position	As-Left Switch Position
RC-2A Channel B (Amber lamps S/G RC-2A Chan B/B1)	AI-66B	S/G RC-2A Chan. B Auto Sig Override Relay Test Sw	Bypass	
		S/G RC-2A Chan. B Auto Sig Override Sw AFW Pumps FW-6/FW-10	Override	
		Chan. B AFW Auto Sig Override S/G Feed Valves	B/OR -1107 AFWS	

Examiner Note: Acting as Shift Manager, PROVIDE Keys #55 and #57 when requested.

Time	Position	Applicant's Actions or Behavior
	BOPO	PERFORM the following at Panel AI-66B for RC-2A Channel B:
		• INSERT key #57 and PLACE S/G RC-2A Channel B Auto Signal Override Relay Test Switch in BYPASS.

Time	Position	Applicant's Actions or Behavior
		• INSERT key #55 and PLACE S/G RC-2A Channel B Auto Signal Override Switch AFW Pumps FW-6/FW-10 in OVERRIDE.
		• PLACE Channel B AFW Auto Signal Override S/G Feed Valves to B/OR -1107 AFWS position.
Examiner Note:		**The following steps continue from AOP-23, Section IX, Reset of Inadvertent AFAS.**
	BOPO	PERFORM the following to STOP all AFW Pumps: [Step 4.6]
		• CLOSE YCV-1045, FW-10 Steam Inlet Valve. [Step 4.6.a]
		• PLACE both Override Switches in OVERRIDE: [Step 4.6.b]
		• ISOLATION VALVE YCV-1045A OVERRIDE SW.
		• ISOLATION VALVE YCV-1045B OVERRIDE SW.
		• CLOSE both FW-10 Steam Supply Valves: [Step 4.6.c]
		• YCV-1045A, RC-2A to FW-10 Isolation Valve.
		• YCV-1045B, RC-2B to FW-10 Isolation Valve.
		• ENSURE FIC-1369, AUX FW PUMP FW-10 SUCTION FLOW drops to zero. [Step 4.6.d]
		• STOP FW-6, Electric AFW Pump, and PLACE HC-1367, FW-6 Control Switch, in PULL-TO-LOCK. [Step 4.6.e]
		• ENSURE FIC-1368, AUX FW PUMP FW-6 SUCTION FLOW drops to zero. [Step 4.6.f]
	BOPO	PERFORM the following to return the AFW System to automatic operation: [Step 4.7]
		• PLACE Control Switches for AFW Isolation Valves in RESET: [Step 4.7.a]
		• PLACE HCV-1107A in RESET. • PLACE HCV-1107B in RESET. • PLACE HCV-1108A in RESET. • PLACE HCV-1108B in RESET.
		• PLACE Control Switches for AFW Isolation Valves in AUTO: [Step 4.7.b]

Operating Test :	NRC	Scenario #	4	Event #	3	Page	18	of	36
Event Description:	Inadvertent Auxiliary Feedwater Actuation Signal								

Time	Position	Applicant's Actions or Behavior
		• PLACE HCV-1107A in AUTO. • PLACE HCV-1107B in AUTO. • PLACE HCV-1108A in AUTO. • PLACE HCV-1108B in AUTO.
		• PLACE Control Switch for YCV-1045, FW-10 Steam Inlet Valve, in RESET. [Step 4.7.c]
		• PLACE Control Switch for YCV-1045, FW-10 Steam Inlet Valve, in AUTO. [Step 4.7.d]
		• PLACE both Override Switches in NORMAL. [Step 4.7.e]
		• ISOLATION VALVE YCV-1045A OVERRIDE SW
		• ISOLATION VALVE YCV-1045B OVERRIDE SW
		• PLACE HC-1367, FW-6 Control Switch, in AFTER-STOP. [Step 4.7.f]
Booth Operator: **When contacted, EXECUTE remote functions to RESET FW-10 and Trip Latch Clamp is finger tight.**		
	BOPO	CONTACT Auxiliary Operator ENSURE FW-64-RL, AUX FEED PUMP FW-10 MANUAL TRIP LATCH RESET LEVER is latched: [Step 4.8]
		• VERIFY Reset Lever is seated.
		• ENSURE FW-64-C, AUX FEED PUMP FW-10 MANUAL TRIP LATCH CLAMP is installed finger tight.
	CRS	EXIT Technical Specification LCO 2.5, Steam and Feedwater. [Step 4.9]
When AFAS has been RESET, PROCEED to Event 4.		

Operating Test : NRC	Scenario # 4	Event # 4	Page 19 of 36
Event Description: Loss of Instrument Bus			

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 4. - Loss of Instrument Bus AI-40A.		
Indications Available: CB-20/A15 – INVERTER A TROUBLE; CB-20/A15 – INSTRUMENT BUS A LOW VOLTAGE/GROUND (~10 seconds later); Multiple Loss of Instrument Bus alarms		
+30 sec	BOPO	RESPOND to Annunciator Response Procedures.
	CREW	INFORM CRS of Loss of Instrument Bus AI-40A.
Booth Operator: When contacted, REPORT Inverter A Output Breaker is TRIPPED.		
	CRS	REFER to AOP-16, Loss of Instrument Bus Power, Section I, Loss of Instrument Bus Power.
Examiner Note: The following steps are from AOP-16, Loss of Instrument Bus Power, Section I, Loss of Instrument Power.		
	CRS	DETERMINE a Reactor Trip has NOT occurred: [Step 4.1]
	CRS	DETERMINE appropriate AOP-16 Section: [Step 4.2]
		• OBSERVE an INVERTER A TROUBLE alarm.
		• OBSERVE an INSTRUMENT BUS A LOW VOLTAGE/GROUND alarm.
	CRS	GO TO AOP-16, Section II, Loss of Instrument Bus AI-40A.
Examiner Note: The following steps are from AOP-16, Loss of Instrument Bus Power, Section II, Loss of Instrument Bus AI-40A.		
	CRS	VERIFY Loss of Instrument Bus AI-40A by the following: [Step 4.1]
		• INVERTER A TROUBLE alarm.
		• INSTRUMENT BUS A LOW VOLTAGE/GROUND alarm.

Operating Test :	NRC	Scenario #	4	Event #	4	Page	20	of	36
Event Description:	Loss of Instrument Bus								

Time	Position	Applicant's Actions or Behavior

NOTE

1. Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Reactivity Control Safety Function** is affected as follows:
   - All RPS Channel A is in trip
   - Channel A "VARIABLE OVER POWER TRIP POWER MARGIN A/JI-007" meter is inoperable
   - Channel A Wide Range Log Power Meter and Rate Meter are inoperable
   - The Diverse Scram System is in half-trip
   - Three RPS Logic Matrix channels are inoperable
2. Loss of more than one RPS Logic Matrix channel requires entry into T.S. 2.15.2.
3. If the associated clutch power supply is selected to Instrument Bus A then two RPS Trip Initiation Logic channels (AB, AC, AD) are inoperable.

Time	Position	Applicant's Actions or Behavior
	ATCO	DETERMINE clutch power supply selected to AI-40A and VERIFY clutch power supply is DEENERGIZED: [Step 4.2]
		• OBSERVE AI-3-PS1 output current is 0. • OBSERVE AI-3-PS3 output current is 0. • OBSERVE AI-3-PS1 Indicating lights are out. • OBSERVE AI-3-PS3 Indicating lights are out. • OBSERVE clutch power supply breaker in "half trip" position.
<u>Examiner Note</u>: Acting as Shift Manager, PROVIDE Trip Unit Keys #1 to #12 when requested.		
	ATCO	INSERT keys and BYPASS all RPS Channel A Bistable Trip Units. [Step 4.3]
	CRS	COMPLY with Technical Specification 2.15.2(5). [Step 4.4]
	CRS	EVALUATE Technical Specification LCO 2.15, Instrumentation and Control Systems
		• LCO 2.15.2 – Reactor Protective System Logic and Trip Initiation
		• CONDITION 2.15.2.(2) – One RPS Trip Initiation Logic channel inoperable. • ACTION 2.15.2.(2) – Deenergize the affected clutch power supply within one hour (in ½ trip). • ACTION 2.15.2.(5) – With the required actions of (2) not met, be in HOT SHUTDOWN and verify no more than one CEA is capable of being withdrawn within 6 hours.

Operating Test : NRC	Scenario # 4	Event # 4	Page 21 of 36
Event Description: Loss of Instrument Bus			

Time	Position	Applicant's Actions or Behavior

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Vital Auxiliaries Safety Function** are inoperable:

- "WEST RW SUPPLY HEADER FLOW FIC-2891" indicator
- "CC HT EXCH AC-1A RW OUTLET TEMP TIC-2885"
- "CNTMT CLG COIL VA-1A OUTLT ISOL VLV CNTRLR HCV-400C"
- "CNTMT CLG COIL VA-1B OUTLT ISOL VLV CNTRLR HCV-401C"
- "CNTMT CLG COIL VA-8A OUTLT ISOL VLV CNTRLR HCV-402C"
- "CNTMT CLG COIL VA-8B OUTLT ISOL VLV CNTRLR HCV-403C"

Time	Position	Applicant's Actions or Behavior
	ATCO	ENSURE CCW System operation satisfactory: [Step 4.5]
		• DETERMINE one CCW Pump RUNNING.
		• DETERMINE CCW pressure ≥ 60 psig.
	ATCO	DETERMINE one Raw Water Pump RUNNING. [Step 4.6]
	BOPO	DETERMINE Instrument Air pressure ≥ 90 psig. [Step 4.7]

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **RCS Inventory Control Safety Function** is affected as follows:

- Letdown is isolated
- Pressurizer Level Channel X is inoperable
- CH-1A, Charging Pump, is inoperable
- Charging Pump Backup Auto starts are disabled

Time	Position	Applicant's Actions or Behavior
	ATCO	MAINTAIN Pressurizer level between 30% and 70% and TRENDING to between 45% percent by operating Charging Pumps CH-1B and/or CH-1C per IC-11, Inventory Control. [Step 4.8]
	ATCO	CLOSE TCV-202, Letdown Isolation Valve. [Step 4.9]
	ATCO	PLACE HC-101, Pressurizer Level Channel Selector Switch, in CHAN Y position. [Step 4.10]

Operating Test :	NRC	Scenario #	4	Event #	4	Page	22	of	36
Event Description:	Loss of Instrument Bus								

Time	Position	Applicant's Actions or Behavior

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **RCS Pressure Control Safety Function** is affected as follows:

- RCS Pressure Control Channel X is inoperable
- "PRESSURIZER PRESSURE A/PIA-102X AND A/PIA-102Y" indicators are inoperable
- PZR Backup Heaters are on
- PZR Heater Cutout is inoperable

Time	Position	Applicant's Actions or Behavior
	ATCO	PLACE HC-103, Pressurizer Pressure Channel Selector Switch in CHAN Y position. [Step 4.11]
	ATCO	Manually CONTROL Pressurizer Heaters per PC-11, Pressure Control. [Step 4.12]
	ATCO	MAINTAIN RCS pressure per PC-12, RCS Pressure-Temperature Limits. [Step 4.13]

NOTE

1. Only one additional channel trip is needed to actuate the PORVs, even if the channel in trip is bypassed.
2. When RCS Heatup or Cooldown is in progress, the PORVs are the primary means of Low Temperature Overpressure Protection.
3. Closing the PORV block valves requires entry into Tech Spec 2.1.6.

Time	Position	Applicant's Actions or Behavior
	CRS	CONSIDER closing PORV Block Valves HCV-150 and HCV-151. [Step 4.14]

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Core Heat Removal Safety Function** are inoperable:

- "SUBCOOLED MARGIN MONITOR A-168"
- "RC LOOP TEMPERATURES LOOP 1A "T-COLD" A/TI-112C"
- "RC LOOP TEMPERATURES LOOP 1 "T-HOT" A/TI-112H"
- "RC LOOP TEMPERATURES LOOP 2A "T-COLD" A/TI-122C"
- "RC LOOP TEMPERATURES LOOP 2 "T-HOT" A/TI-122H"
- "SHTDN HT EXCH AC-4A OUTLET VALVE CNTRLR HCV-484"

Time	Position	Applicant's Actions or Behavior
	ATCO	DETERMINE all RCPs are RUNNING. [Step 4.15]

Operating Test :	NRC	Scenario #	4	Event #	4	Page	23	of 36
Event Description:	Loss of Instrument Bus							

Time	Position	Applicant's Actions or Behavior

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **RCS Heat Removal Safety Function** is inoperable:

- "EMGY FW STOR TNK LEVEL LIA-1183"
- "STEAM GENERATOR PRESSURE S/G RC-2A A/PIC-902"
- "STEAM GENERATOR PRESSURE S/G RC-2B A/PIC-905"
- "AUX FW PUMP FW-6 SUCTION FLOW FIC-1368"

	BOPO	DETERMIN Steam Generator NR levels steady at ~63%. [Step 4.16]

NOTE

- Upon loss of Instrument Bus A, RM-091A, which is associated with the **Containment Integrity Safety Function** is inoperable.

	ATCO	PERFORM the following to CONFIRM Containment Integrity: [Step 4.17]
		• DETERMINE no unexpected rise in Containment Sump level. [Step 4.17.a]
		• DETERMINE no Containment Area Radiation Monitor alarms. [Step 4.17.b]
		• DETERMINE Radiation Monitors RM-051 / RM-052 / RM-062 NOT in alarm. [Step 4.17.c]
		• DETERMINE SG Blowdown or Condenser off Gas Radiation Monitors RM-054A / RM-054B / RM-057 NOT in alarm. [Step 4.17.d]
		• DETERMINE Containment conditions NORMAL. [Step 4.17.e]
		• DETERMINE Containment pressure < 3 psig.
		• DETERMINE Containment temperature <120°F.
	ATCO	PLACE the following switches in TEST: [Step 4.18]
		• HC-344/TEST, CNTMT SPRAY VLV HCV-344 TEST SWITCH • HC-345/TEST, CNTMT SPRAY VLV HCV-345 TEST SWITCH

Operating Test :	NRC	Scenario #	4	Event #	4	Page	24	of	36
Event Description:	Loss of Instrument Bus								

Time	Position	Applicant's Actions or Behavior

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Engineered Safety Features Systems** is affected as follows:

- Safety Injection Tanks 6A and 6C level and pressure indicators are inoperable
- OPLS is in half-trip
- Sequencer S2-2 is inoperable
- PPLS is in a two-out-of-three logic mode
- SGLS is in a two-out-of-three logic mode

Time	Position	Applicant's Actions or Behavior
	CRS	REFER to all the following Technical Specifications: [Step 4.19]
		• 2.1.6, Pressurizer and Steam System Safety Valves • 2.2, Chemical and Volume Control System • 2.3, Emergency Core Cooling System • 2.5, Steam and Feedwater Systems • 2.7, Electrical Systems • 2.15, Instrumentation and Control Systems • 2.21, Post-Accident Monitoring Instrumentation
	CRS	EVALUATE Technical Specification LCO 2.7, Electrical Systems
		• LCO 2.7.(1).h – 120 VAC Instrument Bus A (Panel AI-40A).
		• CONDITION 2.7.(2).h – 120 VAC Instrument Bus A (Panel AI-40A) inoperable • ACTION 2.7.(2).h – May remain inoperable for 8 hours provided RPS and ESF instrument channels supplied by the remaining 3 buses are all OPERABLE.
	CREW	REFER to Electrical Load Distribution Listing Manual for a list of components powered from AI-40A. [Step 4.20]

Examiner Note: **Instrument Bus IA-40A will remain deenergized for duration of scenario.**

Operating Test : NRC	Scenario # 4	Event # 4	Page 25 of 36
Event Description: Loss of Instrument Bus			

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When contacted, REPORT Electrical Maintenance investigating issue with Inverter A.		
+15 min	CRS	When cause of power loss has been determined and corrected, RESTORE AI-40A to normal per Attachment 1 or 12 of OI-EE-4, 120 Volt AC System Normal Operation. [Step 4.21]
When Technical Specifications have been addressed, PROCEED to Events 5, 6, and 7.		

Operating Test : NRC	Scenario # 4	Event # 5, 6, & 7	Page 26 of 36
Event Description: RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment			

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Events 5, 6, and 7.** **- Reactor Coolant Pump RC-3A trip.** **- Instrument Air Compressors CA-1B and CA-1C trip.** **- Bearing Cooling Water Pump AC-9B trip.** **- Steam Line Break inside Containment on RC-2A @ 1% severity and 5 minute ramp.**		
Indications Available: **CB-1,2,3,4/A6 – REACTOR COOLANT PUMP RC-3A BREAKER O/L OR TRIP** **Low Flow Trip Unit lights lit on all RPS Channels B/C/D.** **ERF Computer System alarms for low RCS flow**		
+30 sec	ATCO	RECOGNIZE RPS Low Flow lights lit and MANUALLY trip Reactor.
	CRS	DIRECT performance of EOP-00, Standard Post Trip Actions.
Examiner Note: **The following steps are from EOP-00, Standard Post Trip Actions.**		
	ATCO	VERIFY Reactivity Control: [Step 1]
		• VERIFY ALL of the following: [Step 1.a]
		• DETERMINE more than one Regulating or Shutdown CEA NOT inserted.
		• [CA] If Reactor did NOT trip, ESTABLISH Reactivity Control by performing step a, b, c or d: [Step 1.1]
CRITICAL TASK STATEMENT		**Manually Trip Reactor to meet Core Design Criteria of Lowering Reactor Power and Negative Startup Rate to Verify Reactivity Control Established During ATWS Event.**
CRITICAL TASK	ATCO	• [CA] Manually TRIP Reactor at CB-4. [Step 1.1.a]
		• VERIFY Reactor Power is LOWERING.
		• VERIFY Startup Rate is NEGATIVE.

Operating Test : NRC	Scenario # 4	Event # 5, 6, & 7	Page 27 of 36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment		

Time	Position	Applicant's Actions or Behavior
<u>Examiner Note</u>: An Emergency Boration is performed once the cooldown is recognized.		
		• DETERMINE an uncontrolled RCS Cooldown in progress. [Step 1.b]
		• [CA] PERFORM Emergency Boration with uncontrolled cooldown in progress. [Step 1.2]
		• [CA] ENSURE both following valves CLOSED: [Step 1.2.a]
		• [CA] FCV-269X, Demin Water Makeup Valve
		• [CA] FCV-269Y, Boric Acid Makeup Valve
		• [CA] OPEN all the following valves: [Step 1.2.b]
		• [CA] HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
		• [CA] HCV-265, CH-11A Gravity Feed Valve
		• [CA] HCV-258, CH-11B Gravity Feed Valve
		• [CA] START all the following pumps: [Step 1.2.c]
		• [CA] Boric Acid Pump CH-4A
		• [CA] Boric Acid Pump CH-4B
		• [CA] Charging Pump CH-1A (inoperable)
		• [CA] Charging Pump CH-1B (running)
		• [CA] Charging Pump CH-1C (unavailable)
		• [CA] CLOSE LCV-218-2, VCT Outlet Valve. [Step 1.2.d]
		• [CA] ENSURE the following valves CLOSED: [Step 1.2.e]
		• [CA] LCV-218-3, Charging Pump Suction SIRWT Isolation Valve
		• [CA] HCV-257, CH-4B Recirc Valve
		• [CA] HCV-264, CH-4A Recirc Valve
		• [CA] BORATE until adequate Shutdown Margin is established. [Step 1.2.f]
	CRS	DETERMINE Reactivity Control criteria SATISFIED.
	BOPO	VERIFY Turbine Trip: [Step 2]
		• VERIFY HP & LP Stop and Intercept Valves CLOSED.

Operating Test :	NRC	Scenario #	4	Event #	5, 6, & 7	Page	28	of	36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment								

Time	Position	Applicant's Actions or Behavior
<u>Examiner Note</u>: The Generator Output Breakers are CLOSED due to back feeding.		
	BOPO	ENSURE all Generator Breakers are tripped: [Step 3]
		• DETERMINE Generator Output Breaker 3451-4 CLOSED.
		• DETERMINE Generator Output Breaker 3451-5 CLOSED.
		• DETERMINE Generator Field Breaker 41E/G1F tripped.
	BOPO	VERIFY 4160 V Safeguards Buses 1A3 and 1A4 are ENERGIZED. [Step 4]
	CRS	DETERMINE Maintenance of Vital Auxiliaries criteria SATISFIED.
	BOPO	DETERMINE Safety Injection Actuation Signal has NOT occurred and both Diesel Generators are STOPPED. [Step 5]
	BOPO	VERIFY 4160 V Non-Safeguards Buses 1A1 and 1A2 are ENERGIZED. [Step 6]
	BOPO	VERIFY 125 VDC Buses 1 and 2 are ENERGIZED. [Step 7]
	BOPO	VERIFY Instrument Air is AVAILABLE: [Step 8]
		• DETERMINE Instrument Air pressure < 90 psig.
		• DETERMINE Instrument Air Compressors NOT RUNNING.
		• [CA] If Instrument Air pressure is < 90 psig, PERFORM the following to restore Instrument Air: [Step 8.1]
	BOPO	• [CA] START Bearing Water Pump AC-9A.
	BOPO	• [CA] START Air Compressor CA-1A.
	ATCO	VERIFY Component Cooling Water System operation NORMAL: [Step 9]
		• DETERMINE at least one CCW pump RUNNING. [Step 9.a]
		• DETERMINE CCW Pump discharge pressure ≥ 60 psig. [Step 9.b]

Operating Test :	NRC	Scenario #	4	Event #	5, 6, & 7	Page	29	of	36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment								

Time	Position	Applicant's Actions or Behavior
		• DETERMINE HCV-438A/B/C/D, CCW to RCP Coolers OPEN. [Step 9.c]
		• DETERMINE at least one Raw Water Pump RUNNING. [Step 9.d]
	CRS	VERIFY RCS Inventory Control criteria satisfied: [Step 10]
	ATCO	• DETERMINE PZR level between 30% and 70% and NOT TRENDING to between 45% and 60%.
		• [CA] RESTORE Inventory Control by manually controlling Charging and Letdown. [Step 10.1.a]
		• DETERMINE RCS subcooling > 20°F.
	CRS	DETERMINE RCS Inventory Control criteria NOT SATISFIED.
	CRS	VERIFY RCS Pressure Control criteria satisfied: [Step 11]
	ATCO	• DETERMINE RCS pressure between 1800 psia and 2300 psia.
		• DETERMINE RCS pressure NOT TRENDING between 2050 psia and 2150 psia.
		• [CA] MANUALLY CONTROL PZR Heaters and Spray to restore RCS pressure.
		• DETERMINE PORVs are CLOSED.
	CRS	DETERMINE RCS Pressure Control criteria NOT SATISFIED.
	CRS	VERIFY Core Heat Removal criteria satisfied: [Step 12]
	ATCO	• DETERMINE RCP NPSH requirements met.
		• DETERMINE at least one RCP operating.
		• DETERMINE Core $\Delta T \leq 10°F$.
	CRS	DETERMINE Core Heat Removal criteria SATISFIED.
	CRS	VERIFY RCS Heat Removal criteria satisfied:

Operating Test : NRC	Scenario # 4	Event # 5, 6, & 7	Page 30 of 36

Event Description: RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment

Time	Position	Applicant's Actions or Behavior
	BOPO	VERIFY Main Feedwater is restoring SG levels to 35% to 80% NR and 73% to 94% WR. [Step 13]
		• DETERMINE FCV-1101 & FCV-1102 Feed Regulating Valves CLOSED. [Step 13.a]
		• DETERMINE HCV-1105 & HCV-1106 Feed Regulating Bypass Valves ramped to between 40% & 45% OPEN. [Step 13.b]
	BOPO	• PLACE 43/FW switch in OFF. [Step 13.c]
		• ENSURE no more than one Main Feedwater Pump RUNNING. [Step 13.d]
		• ENSURE no more than one Condensate Pump RUNNING. [Step 13.e]
		• STOP running Heater Drain Pumps FW-5A, FW-5B, and/or FW-5C. [Step 13.f]
	BOPO	• ENSURE SG Blowdown Isolation Valves CLOSED. [Step 13.g]
		• HCV-1387A & HCV-1387B
		• HCV-1388A & HCV-1388B
<u>Booth Operator</u>: When contacted, REPORT Air Assisted Main Steam Safety Valves CLOSED.		
	BOPO	VERIFY Steam Dump and Bypass Valves controlling <u>both</u> of the following: [Step 14]
		• DETERMINE RCS T_{COLD} NOT between 525°F and 535°F.
		• DETERMINE Steam Generator RC-2A pressure < 700 psia.
		• [CA] If T_{COLD} less than 525°F, PERFORM the following: [Step 14.1]
	BOPO	• [CA] CLOSE Steam Dump and Bypass Valves. [Step 14.1.a]
		• [CA] VERIFY HCV-1040, Atmospheric Dump Valve CLOSED. [Step 14.1.b]
		• [CA] CHECK Air Assisted Main Steam Safety Valves CLOSED. [Step 14.1.c]
		• [CA] If Steam Generator pressure < 700 psia, ISOLATE Main Steam Header. [Step 14.1.d]
		• [CA] CLOSE HCV-1041A, MSIV. [Step 14.1.d.1)]
		• [CA] CLOSE HCV-1042A, MSIV. [Step 14.1.d.1)]
		• [CA] VERIFY HCV-1041A, MSIV Bypass CLOSED. [Step 14.1.d.2)]

Operating Test :	NRC	Scenario #	4	Event #	5, 6, & 7	Page	31	of	36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment								

Time	Position	Applicant's Actions or Behavior
		• [CA] VERIFY CLOSE HCV-1042A, MSIV Bypass CLOSED. [Step 14.1.d.2)]
		• [CA] DETERMINE Steam Generator pressure > 500 psia. [Step 14.1.e]
	CRS	DETERMINE RCS Heat Removal criteria NOT SATISFIED.
	CRS	VERIFY Containment Isolation criteria satisfied:
	ATCO	VERIFY Normal Containment conditions exist: [Step 15]
		• DETERMINE rise in Containment Sump level in progress. [Step 15.a]
		• DETERMINE Containment Area Radiation Monitors NOT in alarm. [Step 15.b]
		• DETERMINE Containment Ventilation Radiation Monitors NOT in alarm. [Step 15.c]
	ATCO	• DETERMINE SG Blowdown and Condenser Off Gas Radiation Monitors NOT alarming. [Step 15.d]
	ATCO	• DETERMINE SG Blowdown and Condenser Off Gas Radiation Monitors NOT TRENDING to alarm. [Step 15.e]
	ATCO	• VERIFY Containment conditions: [Step 15.f]
		• DETERMINE Containment pressure > 3 psig.
		• DETERMINE Containment temperature > 120°F.
		• [CA] INITIATE Containment Cooling. [Step 15.f.1]
	ATCO	• [CA] ENSURE CCW flow to Containment Vent Fan coils.
		• [CA] PLACE HCV-402B/D to OPEN.
		• [CA] PLACE HCV-403B/D to OPEN.
		• [CA] PLACE HCV-402A/C to OPEN.
		• [CA] PLACE HCV-403A/C to OPEN.
	ATCO	• [CA] START all Containment Vent Fans.
		• [CA] VERIFY Containment Vent Fans VA-3A & VA-3B RUNNING.
		• [CA] START Containment Vent Fans VA-7C & VA-7D.
		• [CA] DETERMINE Containment pressure < 5 psig. [Step 15.f.2]

Operating Test :	NRC	Scenario #	4	Event #	5, 6, & 7	Page	32	of 36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment							

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE Containment Integrity criteria NOT SATISFIED.
	CRS	DIAGNOSE event in progress: [Step 16]
		• DETERMINE Reactivity Control requirements met.
		• DETERMINE both DC buses energized.
		• DETERMINE at least one Vital 4160 V Bus energized.
		• DETERMINE at least one Non-Vital 4160 V Bus energized.
		• DETERMINE at least one RCP running.
		• DETERMINE adequate Feedwater flow to at least one SG.
		• VERIFY Pressurizer pressure > 1800 psia with high subcooled margin, normal SG pressure, and no indications of primary to secondary leakage.
		• If not, CONSIDER EOP-05, Uncontrolled Heat Extraction.

NOTE

Certain events (i.e., LOCA, SGTR, UHE and Loss of All Feedwater) do not require offsite power in order to adequately, mitigate the effects of the accident. For this reason, the LOCA, SGTR, UHE or Loss of All Feedwater procedure may be implemented even if a Loss of Offsite Power has also occurred.

Time	Position	Applicant's Actions or Behavior
		• DETERMINE all Safety Function Acceptance Criteria NOT SATISFIED.
		• DETERMINE single event in progress and TRANSITION to EOP-05, Uncontrolled Heat Extraction.

Event Description: RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment

Time	Position	Applicant's Actions or Behavior
Examiner Note: The following steps are from EOP-05, Uncontrolled Heat Extraction.		
	CRS	CONFIRM Standard Post Trip Actions have been performed. [Step 1]
	CRS	CONFIRM Uncontrolled Heat Extraction Diagnosis: [Step 2]
		• VERIFY Safety Function Status Check Acceptance Criteria being satisfied. [Step 2.a]
		• DETERMINE CIAS is NOT present and DIRECT Shift Chemist to SAMPLE both SGs for activity. [Step 2.c]
	CRS	IMPLEMENT the Emergency Plan. [Step 3]
		• Time: __________
	CREW	MONITOR the Floating Steps. [Step 4]
	CRS	DETERMINE RCS pressure > 1600 psia, Containment pressure < 5 psig, with Steam Generator ≤ 500 psia. [Step 5]
	BOPO	• ENSURE SGIS closes all the following valves: [Step 5.d]
		• DETERMINE HCV-1041A, RC-2A MSIV CLOSED.
		• DETERMINE HCV-1041C, RC-2A MSIV Bypass Valve CLOSED.
		• DETERMINE HCV-1042A, RC-2B MSIV CLOSED.
		• DETERMINE HCV-1042C, RC-2B MSIV Bypass Valve CLOSED.
		• DETERMINE HCV-1105, RC-2A Feed Regulating Bypass Valve CLOSED.
		• DETERMINE HCV-1106, RC-2B Feed Regulating Bypass Valve CLOSED.
		• DETERMINE HCV-1386, RC-2A Feed Header Isolation Valve CLOSED.
		• DETERMINE HCV-1385, RC-2B Feed Header Isolation Valve CLOSED.
		• DETERMINE HCV-1103, RC-2A Feed Regulating Block Valve CLOSED.
		• DETERMINE HCV-1104, RC-2B Feed Regulating Block Valve CLOSED.

Time	Position	Applicant's Actions or Behavior
	CRS	DETERMINE RCS pressure ≥ 1600 psia. [Step 6]
	CRS	DETERMINE Containment pressure < 5 psig. [Step 7]
	CRS	DETERMINE SIAS has NOT actuated. [Step 8]
	ATCO	VERIFY RCP operating parameters: [Step 9]
		• DETERMINE RCP RC-3A TRIPPED and T_{COLD} < 500°F. [Step 9.a]
		• DETERMINE RCS pressure ~1900 psia. [Step 9.b]
		• DETERMINE RCPs subcooling > 20°F. [Step 9.c]
	ATCO	VERIFY normal CCW/RW System operation: [Step 10]
		• DETERMINE at least 2 CCW Pumps are RUNNING. [Step 10.a]
		• DETERMINE CCW Pump discharge pressure ≥ 60 psig. [Step 10.b]
		• ENSURE at least two Raw Water Pumps operating. [Step 10.c]
	ATCO	• START at least one Raw Water Pump.
		• DETERMINE at least three RW/CCW Heat Exchangers in service. [Step 10.d]
		• DETERMINE all RCP cooler CCW Valves OPEN. [Step 10.e]
	CRS	DETERMINE affected SG is RC-2A and SG pressure is < 700 psia. [Step 11]
	CRS	DETERMINE Uncontrolled Heat Extraction has NOT been isolated. [Step 12]
		• [CA] DETERMINE Emergency Boration already in progress. [Step 12.1]
	BOPO	DETERMINE SG RC-2A < 500 psia and SG RC-2B > 500 psia. [Step 13]
	BOPO	DETERMINE Steam Generator RC-2A is most affected SG. [Step 14]
	CRS	DETERMINE Uncontrolled Heat Extraction has NOT been isolated. [Step 15]

Time	Position	Applicant's Actions or Behavior
	CRS	IF RC-2A is most affected, ISOLATE RC-2A by performing HR-19, Isolate/Restore Steam Generator A. [Step 16]
<u>Examiner Note</u>: The following steps are from HR-19, Isolate/Restore Steam Generator A.		
CRITICAL TASK STATEMENT		**Isolate the Affected Steam Generator to Prevent Excess Plant Cooldown and Reactivity Additions Prior to Steam Generator Level = 0% Wide Range Level.**
CRITICAL TASK	BOPO	PERFORM the following to isolate Steam Generator RC-2A: [Step 1]
		• ENSURE all the following valves are CLOSED: [Step 1.a]
	BOPO	• VERIFY HCV-1041A, RC-2A MSIV CLOSED. • VERIFY HCV-1041C, RC-2A MSIV Bypass Valve CLOSED. • VERIFY MS-291, Air Assisted Main Steam Safety Valve CLOSED. • VERIFY FCV-1101, RC-2A Feed Regulating Valve CLOSED. • VERIFY HCV-1105, Feed Regulating Bypass Valve CLOSED. • VERIFY HCV-1386, RC-2A Feed Header Isolation Valve CLOSED. • VERIFY HCV-1103, Feed Regulating Block Valve CLOSED. • VERIFY HCV-1388A, Blowdown Isolation Valve CLOSED. • VERIFY HCV-1388B, Blowdown Isolation Valve CLOSED. • **CLOSE HCV-1107A, AFW Isolation Valve.** • **CLOSE HCV-1107B, AFW Isolation Valve.**
		• CONTACT Auxiliary Operator to CLOSE MS-298, Steam Valves HCV-1041A & 1042A Packing Leakoff Line Isolation Valve in Room 81. [Step 1.b]
		• If sampling is NOT in progress, CLOSE both Sample Valves: [Step 1.c]
		• HCV-2506A, RC-2A Blowdown Sample Isolation Valve
		• HCV-2506B, RC-2B Blowdown Sample Isolation Valve
		• PERFORM the following to CLOSE YCV-1045A: [Step 1.d]
	BOPO	• PLACE ISOLATION VALVE YCV-1045A OVERRIDE SW in OVERRIDE. [Step 1.d.1)]
	BOPO	• PLACE control switch for S/G RC-2A STM TO FW-10 HDR A ISOLATION VALVE YCV-1045A in CLOSE. [Step 1.d.2)]

Operating Test :	NRC	Scenario #	4	Event #	5, 6, & 7	Page	36	of	36
Event Description:	RCP Trip / Automatic Reactor Trip Failure / Instrument Air Compressors Trip / Bearing Cooling Water Pump Trip / Steam Line Break inside Containment								

Time	Position	Applicant's Actions or Behavior
		• CONTACT Auxiliary Operator HANDJACK YCV-1045A, MAIN STEAM LINE "A" TO AUX FEEDWATER PUMP FW-10 SUPPLY VALVE to CLOSE in Room 81. [Step 1.d.3)]
	CRS	VERIFY RC-2A is most affected SG per Attachment HR-18, Most Affected Steam Generator Determination. [Step 2]
When Steam Generator RC-2A is isolated, TERMINATE the scenario.		

Facility: Fort Calhoun Station		Date of Exam: 12/14/15							Operating Test No.: NRC								
APPLICANT	EVENT TYPE	SCENARIOS												TOTAL	MINIMUM(*)		
		FCS #1			FCS #2			FCS #3									
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
SRO-U1	RX				-									0	1	1	0
	NOR				-									0	1	1	1
	I/C				3,4,5									3	4	4	2
	MAJ				6,8									2	2	2	1
	TS				3,4									2	0	2	2
SRO-U2	RX				-									0	1	1	0
	NOR				-									0	1	1	1
	I/C				3,4,5									3	4	4	2
	MAJ				6,8									2	2	2	1
	TS				3,4									2	0	2	2
SRO-I1	RX	-					-		6					1	1	1	0
	NOR	-					-		-					0	1	1	1
	I/C	2,3,4,5					3,5,9		2,4,5					10	4	4	2
	MAJ	6					6,8		7					4	2	2	1
	TS	2,4					-		-					2	0	2	2
SRO-I2	RX		5		-					-				1	1	1	0
	NOR		1		-					1				1	1	1	1
	I/C		2,4,8,9		3,4,5					3,5,8				10	4	4	2
	MAJ		6		6,8					7				4	2	2	1
	TS		-		3,4					-				2	0	2	2
SRO-I3	RX			-		-		-						0	1	1	0
	NOR			-		1		6						2	1	1	1
	I/C			3,5,7		4,7		2,3,4,5						9	4	4	2
	MAJ			6		6,8		7						4	2	2	1
	TS			-		-		4,5						2	0	2	2
SRO-I4	RX	-				-		-						0	1	1	0
	NOR	-				1		6						2	1	1	1
	I/C	2,3,4,5				4,7		2,3,4,5						10	4	4	2
	MAJ	6				6,8		7						4	2	2	1
	TS	2,4				-		4,5						4	0	2	2
SRO-I5	RX	-				-								0	1	1	0
	NOR	-				1								1	1	1	1
	I/C	2,3,4,5				4,7								5	4	4	2
	MAJ	6				6,8								3	2	2	1
	TS	2,4				-								2	0	2	2

Facility: Fort Calhoun Station   Date of Exam: 12/14/15   Operating Test No.: NRC

APPLICANT	EVENT TYPE	SCENARIOS: FCS #1 CREW POSITION			FCS #2 CREW POSITION			FCS #3 CREW POSITION			CREW POSITION			TOTAL	MINIMUM(*)		
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO-1	RX		5				-			-				1	1	1	0
	NOR		1				-			1				2	1	1	1
	I/C		2,4,8, 9				3,5,9			3,5,8				10	4	4	2
	MAJ		6				6,8			7				4	2	2	1
	TS		-				-			-				0	0	2	2
RO-2	RX			-					6					1	1	1	0
	NOR			-					-					0	1	1	1
	I/C			3,5,7					2,4,5					6	4	4	2
	MAJ			6					7					2	2	2	1
	TS			-					-					0	0	2	2
RO-3	RX		5				-							1	1	1	0
	NOR		1				-							1	1	1	1
	I/C		2,4,8, 9				3,5,9							7	4	4	2
	MAJ		6				6,8							3	2	2	1
	TS		-				-							0	0	2	2
SUR.	RX			-													
	NOR			-													
	I/C			3,5,7													
	MAJ			6													
	TS			-													

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls" (ATC) and "balance-of-plant" (BOP) positions; Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-1 *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a 1-for-1 basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Facility: Fort Calhoun Station Date of Exam: 12/14/15 Operating Test No.: NRC

APPLICANT	EVENT TYPE	SCENARIOS												TOTAL	MINIMUM(*)		
		FCS #1			FCS #2			FCS #3									
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO-1	RX		5				-			-				**1**	**1**	**1**	**0**
	NOR		1				-			1				**2**	**1**	**1**	**1**
	I/C		2,4,8, 9				3,5,9			3,5,8				**10**	**4**	**4**	**2**
	MAJ		6				6,8			7				**4**	**2**	**2**	**1**
	TS		-				-			-				**0**	**0**	**2**	**2**
RO-2	RX			-					6					**1**	**1**	**1**	**0**
	NOR			-					-					**0**	**1**	**1**	**1**
	I/C			3,5,7					2,4,5					**6**	**4**	**4**	**2**
	MAJ			6					7					**2**	**2**	**2**	**1**
	TS			-					-					**0**	**0**	**2**	**2**
RO-3	RX		5				-							**1**	**1**	**1**	**0**
	NOR		1				-							**1**	**1**	**1**	**1**
	I/C		2,4,8, 9				3,5,9							**7**	**4**	**4**	**2**
	MAJ		6				6,8							**3**	**2**	**2**	**1**
	TS		-				-							**0**	**0**	**2**	**2**
SUR.	RX			-													
	NOR			-													
	I/C			3,5,7													
	MAJ			6													
	TS			-													

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls" (ATC) and "balance-of-plant" (BOP) positions; Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-1 *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a 1-for-1 basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Facility: FCS	Date of Examination: 12/14/15												Operating Test No. NRC 1/2/3
	Applicants												
Competencies	SROU-1				SROU-2				SRO1-1				
	SCENARIO				SCENARIO				SCENARIO				
	1	2	3		1	2	3		1	2	3		
Interpret/Diagnose Events and Conditions	-	3,4,5,6,8	-		-	3,4,5,6,8	-		2,3,4,5,6	3,5,6,8,9	2,4,5,6,7		
Comply With and Use Procedures (1)	-	2,3,4,5,6,8	-		-	2,3,4,5,6,8	-		2,3,4,5,6	3,5,6,8,9	2,4,5,6,7		
Operate Control Boards (2)	-	N/A	-		-	N/A	-		N/A	3,5,6,8,9	2,4,5,6,7		
Communicate and Interact	-	1,2,3,4,5,6,8	-		-	1,2,3,4,5,6,8	-		1,2,3,4,5,6	3,5,6,8,9	2,4,5,6,7		
Demonstrate Supervisory Ability (3)	-	3,4,5,6,8	-		-	3,4,5,6,8	-		2,3,4,5,6	N/A	N/A		
Comply With and Use Tech. Specs. (3)	-	3,4	-		-	3,4	-		2,4	N/A	N/A		

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.

Facility: FCS	Date of Examination: 12/14/15			Operating Test No. NRC 1/2/3								
	Applicants											
Competencies	SROI-2				SROI-3				SROI-4			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3		1	2	3		1	2	3	
Interpret/Diagnose Events and Conditions	2,4,5,6, 8,9	3,4,5,6, 8	3,5,6, 7,8		3,5,6,7	4,6,7,8	2,3,4,5, 6,7		2,3,4,5, 6	4,6,7,8	2,3,4,5, 6,7	
Comply With and Use Procedures (1)	1,2,4,5, 6,8,9	2,3,4,5, 6,8	3,5,6, 7,8		3,5,6,7	1,4,6,7, 8	2,3,4,5, 6,7		2,3,4,5, 6	1,4,6,7, 8	2,3,4,5, 6,7	
Operate Control Boards (2)	1,2,4,5, 6,8,9	N/A	1,3,5, 6,7,8		3,5,6,7	1,4,6,7, 8	N/A		N/A	1,4,6,7, 8	N/A	
Communicate and Interact	1,2,3,4, 5,6,8,9	1,2,3,4, 5,6,8	1,3,5, 6,7,8		3,5,6,7	1,4,6,7, 8	1,2,3,4, 5,6,7		1,2,3,4, 5,6	1,4,6,7, 8	1,2,3,4, 5,6,7	
Demonstrate Supervisory Ability (3)	N/A	3,4,5,6, 8	N/A		N/A	N/A	2,3,4,5, 6,7		2,3,4,5, 6	N/A	2,3,4,5, 6,7	
Comply With and Use Tech. Specs. (3)	N/A	3,4	N/A		N/A	N/A	4,5		2,4	N/A	4,5	

Notes:

(4) Includes Technical Specification compliance for an RO.

(5) Optional for an SRO-U.

(6) Only applicable to SROs.

Facility: FCS	Date of Examination: 12/14/15			Operating Test No. NRC 1/2/3								
	Applicants											
Competencies	SROI-5											
	SCENARIO											
	1	2	3									
Interpret/Diagnose Events and Conditions	2,3,4,5, 6	4,6,7,8	-									
Comply With and Use Procedures (1)	2,3,4,5, 6	1,4,6,7, 8	-									
Operate Control Boards (2)	N/A	1,4,6,7, 8	-									
Communicate and Interact	1,2,3,4, 5,6	1,4,6,7, 8	-									
Demonstrate Supervisory Ability (3)	2,3,4,5, 6	N/A	-									
Comply With and Use Tech. Specs. (3)	2,4	N/A	-									

Notes:

(7) Includes Technical Specification compliance for an RO.

(8) Optional for an SRO-U.

(9) Only applicable to SROs.

Facility: FCS	Date of Examination: 12/14/15				Operating Test No. NRC 1/2/3							
	Applicants											
Competencies	RO-1				RO-2				RO-3			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3		1	2	3		1	2	3	
Interpret/Diagnose Events and Conditions	2,4,5,6, 8,9	3,5,6,8, 9	3,5,6, 7,8		3,5,6,7	2,4,5, 6,7	-		2,4,5,6, 8,9	3,5,6,8, 9	-	
Comply With and Use Procedures (1)	1,2,4,5, 6,8,9	3,5,6,8, 9	3,5,6, 7,8		3,5,6,7	2,4,5, 6,7	-		1,2,4,5, 6,8,9	3,5,6,8, 9	-	
Operate Control Boards (2)	1,2,4,5, 6,8,9	3,5,6,8, 9	1,3,5, 6,7,8		3,5,6,7	2,4,5, 6,7	-		1,2,4,5, 6,8,9	3,5,6,8, 9	-	
Communicate and Interact	1,2,3,4, 5,6,8,9	3,5,6,8, 9	1,3,5, 6,7,8		3,5,6,7	2,4,5, 6,7	-		1,2,3,4, 5,6,8,9	3,5,6,8, 9	-	
Demonstrate Supervisory Ability (3)	N/A	N/A	N/A		N/A	N/A	-		N/A	N/A	-	
Comply With and Use Tech. Specs. (3)	N/A	N/A	N/A		N/A	N/A	-		N/A	N/A	-	

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.