Facility: CPNPP JPM # NRC RA1 Task #RO1310 K/A #2.1.43 4.1 / 4.3

Title: Determine Reactivity Effects When Starting Positive Displacement Charging Pump

Examinee (Print): ____________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______

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:

- Unit 1 is operating at approximately 3% power.
- The Positive Displacement Charging Pump must be placed in service per SOP-103A, Chemical and Volume Control System.
- The Reactor Coolant System Boron concentration the last time the Positive Displacement Charging Pump was run was 2400 ppm.
- Current Reactor Coolant System Boron concentration is 1545 ppm.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- CALCULATE a Reactivity Evaluation for starting the Positive Displacement Charging Pump per SOP-103A, Chemical and Volume Control System, Steps 5.3.1.C and 5.3.1.D.
- REPORT findings to the Shift Manager.

Task Standard: Calculate the change in boron and resultant change in temperature when placing the Positive Displacement Pump in service per SOP-103A.

Required Materials: SOP-103A, Chemical and Volume Control System, Rev. 17-23.
Reactivity Briefing Sheet for 1545 ppm Reactor Coolant System conditions.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ____________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **SOP-103A, Chemical and Volume Control System, Section 5.3.1.**
    - **INITIALED up to Step 5.3.1.C.**
- **89.8 EFPD Reactivity Briefing Sheet.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from SOP-103A, Step 5.3.1.**
NOTE:	This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.
Perform Step: 1√ 5.3.1.C 1st line + calculation	IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows: • ΔB = RCS Boron Concentration Difference
Standard:	CALCULATED ΔB = RCS Boron Concentration Difference as follows: ΔB = (2400 ppm PDP - 1545 ppm RCS) x 0.00128 = **1.0944 ppm**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of ITC and HFP Differential Boron Worth may occur.**
Perform Step: 2√ 5.3.1.C 2nd line + calculation	On the Reactivity Briefing Sheet get the following information: ITC _____ pcm/ºF HFP Differential Boron Worth _____ pcm/ppm
Standard:	DETERMINED the following from the Reactivity Briefing Sheet: ITC = **- 1.1 ± 0.1 pcm /ºF** HFP Differential Boron Worth = **- 6.9 ± 0.1pcm / ppm**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of ITC / HFP Differential Boron Worth may occur.**
Perform Step: 3√ 5.3.1.C 2nd line + calculation	On the Reactivity Briefing Sheet get the following information: ITC / HFP Differential Boron Worth = ppm / ºF
Standard:	CALCULATED change in ppm / ºF: - 1.1 pcm / ºF / - 6.9 pcm / ppm = **0.1594 ± 0.02 ppm / ºF**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of Δ T_{AVE} may occur.**
Perform Step: 4√ 5.3.1.C 3rd line + calculation	On the Reactivity Briefing Sheet get the following information: Δ T_{AVE} = Δ B / ppm / ºF
Standard:	CALCULATED change in T_{AVE} as follows: Δ T_{AVE} = Δ B / ppm / ºF = 1.0944 ppm / 0.1594 ppm / ºF = **6.87 ± 1.0 ºF**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 5.3.1.D	IF ΔT_{AVE} calculated above is >1ºF, THEN notify Shift Operations Manager to discuss contingency actions.
Standard:	DETERMINED ΔT_{AVE} calculated is greater than 1ºF and NOTIFIED Shift Manager.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 is operating at approximately 3% power.**
- **The Positive Displacement Charging Pump must be placed in service per SOP-103A, Chemical and Volume Control System.**
- **The Reactor Coolant System Boron concentration the last time the Positive Displacement Charging Pump was run was 2400 ppm.**
- **Current Reactor Coolant System Boron concentration is 1545 ppm.**

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

- **CALCULATE a Reactivity Evaluation for starting the Positive Displacement Charging Pump per SOP-103A, Chemical and Volume Control System, Steps 5.3.1.C and 5.3.1.D.**
- **REPORT findings to the Shift Manager.**

Reactivity Briefing Sheet for Stable Operation
FOR SIMULATOR USE ONLY

Calculations based on core design values, and assume:

Burnup = **4001.3** MWD/MTU
89.82 EFPD
Power = **3** RTP
Boron = **1545** ppm
B10 Conc = **0.1776696** w/o
Control Bank D = **100** steps

Burnup in the BOL range

NOTE: Re-create the Briefing Sheet if current values significantly differ from assumed inputs.

Reactivity affects of Control Bank D

HFP Diff Worth @ 100.0 steps = **-4.3** pcm / step

HFP Integral Rod Worth for CBD Step Positions:

Steps	pcm	Steps	pcm	Steps	pcm	Steps	pcm
110	-269.0	103	-294.7	96	-325.2	85	-382.5
109	-272.4	102	-298.7	95	-329.9	80	-411.7
108	-275.9	101	-302.9	94	-334.8	75	-442.7
107	-279.4	100	-307.1	93	-339.8	70	-475.3
106	-283.1	99	-311.5	92	-344.8	65	-509.2
105	-286.9	98	-316.0	91	-350.0	60	-544.2
104	-290.7	97	-320.5	90	-355.2	55	-580.3

Reactivity affects of Boron

HFP Diff Boron Worth @ 1545 ppm = **-6.9** pcm / ppm

1-FK-110 Pot Setting for Blended Flow @ 1545 ppm = **6.34**
(Assuming BAT concentration of 7492.0 ppm)

Reactivity affects of Power

Power Coefficient of Reactivity = **-13.1** pcm / % RTP

Dilution to equal 1% Power Increase = **83.8** gallons RMUW

Boration to equal 1% Power Decrease = **20.9** gallons boric acid

Reactivity affects of RCS Temperature

Temperature Coefficient of Reactivity (ITC) = **-1.1** pcm / °F

Boration to equal 1°F Temperature Decrease = **1.8** gallons boric acid

Dilution to equal 1°F Temperature Increase = **7.3** gallons RMUW

Load Reduction equal to 1°F T_{ave} Increase = **1.0** MWe

COMANCHE PEAK NUCLEAR POWER PLANT

UNIT 1

SYSTEM OPERATING PROCEDURE MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN-19 07-28-2010 1200

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

CHEMICAL AND VOLUME CONTROL SYSTEM

PROCEDURE NO. SOP-103A

REVISION NO. 17

EFFECTIVE DATE: 03-05-2008 1200

PREPARED BY (Print): Brad Hancock Ext: 6769

TECHNICAL REVIEW BY (Print): Lisabeth Donley Ext: 6524

APPROVED BY: Alan Hall for Dave Goodwin Date: 02-19-2008
DIRECTOR, OPERATIONS

5.3 Major Component Operation

5.3.1 Positive Displacement Pump Startup

This section describes the steps to place the PDP in service.

CAUTION: PDP operation may result in high gaseous activity in the PDP Room due to packing leakage.

NOTE:

- PDP run time should be minimized to conserve pump packing. Run PDP only when required (for example - Slave Relay Testing).
- Following loss of Instrument Air, control air to the PDP fluid drive must be reset. This is done by depressing the RESET pushbutton on the instrument air supply to the PDP fluid drive. This RESET is normally accomplished by ABN-301 restoration section 3.0.
- The reactivity impact for starting the PDP pump is typically very small due to diffusion effects between the PDP piping and the RCS. However, assuming no diffusion, the reactivity effects could potentially approach -15 pcm (and -1.5 °F temperature change) with very large (>1000 ppm) boron concentration differences between the PDP piping and RCS. (EVAL-04-0944-04)
- Following several PDP starts, the PDP fluid drive oil may exceed the upper limit mark on the drive unit's sight glass due to priming) the pump (adding oil) over a period of time. The PROMPT Team should be contacted to drain the excess oil. (SMF-01-0600 and 4-03-149515-00)
- With the PDP stopped, oil level should be in the upper 1/4 of the MIN - MAX range, preferably near the MAX level mark. (SMF-05-2603)

☒ A. Ensure the prerequisites in Section 2.5 are met.

B. IF the PDP has not operated for an extended period (month), THEN prime the PDP fluid drive by performing the following:

N/A

☐ 1) IF pump hydraulic fluid level is at the maximum level of the sightglass, THEN instruct a PROMPT member to drain ~1/2 liter of oil into a clean container. (This oil will be added to the fluid drive at step 5.3.1.B. 3)

☐ 2) Remove the pipe plugs from the two priming holes on top of the input end bell (motor side of the fluid drive).

☐ 3) Pour oil (collected in step 5.3.1.B. 1) a) and/or from the approved lubrication list) into either hole until oil rises to the bottom of the other hole and remains there.

☐ 4) Replace and tighten the pipe plugs.

5.3.1

> NOTE: Steps C and D may be considered NA if in a MODE other than MODE 1 or 2.

☐ C. IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows:

> NOTE: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.

ΔB = RCS Boron Concentration Difference

ΔB = (_______ ppm PDP - ________ ppm RCS) x 0.00128

ΔB = _______ ppm

On the Reactivity Briefing Sheet get the following information:

ITC_____________ pcm/°F HFP Differential Boron Worth__________ pcm/ppm

$$\frac{\text{ITC}}{\text{HFP Differential Boron Worth}} = \frac{\text{\_\_\_\_\_\_\_\_\_\_\_\_ pcm/°F}}{\text{pcm/ppm}} = \text{\_\_\_\_\_\_\_\_\_\_\_ ppm/°F}$$

$$\Delta\text{Tave} = \frac{\triangle\text{B}}{\text{ppm/°F}} = \frac{\text{\_\_\_\_\_\_\_\_\_\_\_\_ ppm}}{\text{ppm/°F}} = \text{\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_ °F}$$

☐ D. IF ΔTave calculated above is >1°F, THEN notify Shift Operations Manager to discuss contingency actions.

> NOTE: If the Stuffing Box Coolant Tank is overfilled, the PDP Charging Pump Room will become contaminated.

E. IF Stuffing Box Coolant Tank is low, THEN fill per the following steps:

☐ 1) Slowly crack OPEN 1CS-0119, PD PMP 1-01 STUFFING BOX COOL TK MU ISOL VLV, until desired fill rate is achieved.

☐ 2) When the desired tank level has been established, CLOSE 1CS-0119.

☐ F. Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.

G. OPEN the following valves:

☐ ● 1/1-8202A, VENT VLV (MCB)

☐ ● 1/1-8202B, VENT VLV (MCB)

5.3.1

☐ H. Ensure 1APPD, POSITIVE DISPLACEMENT CHARGING PUMP 1-01 MOTOR BREAKER 1EB1/2B/BKR is racked to the CONNECT position.

☐ I. Place 1-SK-459A, PDP SPD CTRL, in MANUAL with demand at 55%.

> NOTE: The PDP will not start until 1-8109, PD CHRG PMP 1-01 RECIRC VLV, is open and handswitch 1/1-APPD, PDP, is in the START position. Two minutes after the PDP breaker is closed, 1-8109 will automatically close.

☐ J. OPEN 1/1-8109, PDP RECIRC VLV.

> NOTE: PDP speed may have to be raised rapidly when a CCP is also in operation to prevent the PDP from stalling on low oil pressure.

☐ K. WHEN 1/1-8109, PDP RECIRC VLV is open, THEN start the PDP by placing handswitch 1/1-APPD PDP, to the START position.

☐ L. Ensure 1/1-8109, PDP RECIRC VLV, is CLOSED.

> NOTE: During PDP operation the following step may be performed to lower PDP suction stabilizer level.

M. IF 1/1-8204, H2/N2 SPLY VLV indicates OPEN (red light on), THEN perform the following to lower suction stabilizer level:

[C] ☐ ● OPEN 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV for no more than 10 seconds to clear the high level, then close.

N. IF a CCP is in operation AND it is to be placed in standby, THEN perform the following:

☐ 1) Ensure only ONE letdown orifice is in service per Section 5.2.3.

☐ 2 Alternately raise PDP speed using 1-SK-459A, PDP SPD CTRL, and lower CCP flow using 1-FK-121, CCP CHRG FLO CTRL, until 1-FK-121 is at minimum.

☐ 3) Shut down the running CCP per Section 5.3.4.

[IV] ☐ O. IF desired, THEN gradually adjust 1-SK-459A, PDP SPD CTRL, to achieve the required flow rate AND place in AUTO.

☐ P. Adjust 1-LK-459, PRZR LVL CTRL, as necessary to maintain stable Pressurizer level.

COMMENTS ____

____

____

5.3.1

NOTE: Steps C and D may be considered NA if in a MODE other than MODE 1 or 2.

☐ C. IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows:

NOTE: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.

ΔB = RCS Boron Concentration Difference

ΔB = (2400 ppm PDP - 1545 ppm RCS) x 0.00128

ΔB = 1.0944 ppm

On the Reactivity Briefing Sheet get the following information:

ITC -1.1 +/- 0.1 pcm/°F HFP Differential Boron Worth - 6.9 +/- 0.1 pcm/ppm

$$\frac{\text{ITC}}{\text{HFP Differential Boron Worth}} = \frac{-1.1 \text{ +/- } 0.1 \text{ pcm/°F}}{-6.9 \text{ +/- } 0.1 \text{ pcm/ppm}} = 0.1594 \text{ +/- } 0.02 \text{ ppm/°F}$$

$$\Delta\text{Tave} = \frac{\Delta\text{B}}{\text{ppm/°F}} = \frac{1.0944 \text{ ppm}}{0.1594 \text{ +/- } 0.02 \text{ ppm/°F}} = 6.87 \text{ +/- } 1.0 \text{ °F}$$

☐ D. IF ΔTave calculated above is >1°F, THEN notify Shift Operations Manager to discuss contingency actions.

NOTE: If the Stuffing Box Coolant Tank is overfilled, the PDP Charging Pump Room will become contaminated.

E. IF Stuffing Box Coolant Tank is low, THEN fill per the following steps:

☐ 1) Slowly crack OPEN 1CS-0119, PD PMP 1-01 STUFFING BOX COOL TK MU ISOL VLV, until desired fill rate is achieved.

☐ 2) When the desired tank level has been established, CLOSE 1CS-0119.

☐ F. Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.

G. OPEN the following valves:

☐ • 1/1-8202A, VENT VLV (MCB)

☐ • 1/1-8202B, VENT VLV (MCB)

Facility: CPNPP JPM # NRC RA2 Task #RO5115 K/A #2.1.25 3.9 / 4.2
Title: Calculate Pressurizer and Steam Generator Level from Remote Shutdown Panel

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______

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 Unit 1 Control Room has been evacuated.
- ABN-905A, Loss of Control Room Habitability, is in progress.
- A cooldown is in progress from the Remote Shutdown Panel.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- CALCULATE Steam Generator indicated wide range level to maintain Steam Generator actual level between 74% and 83% when at 400ºF.
- CALCULATE Pressurizer indicated level to maintain Pressurizer actual level between 25% and 50% when at 467ºF.

Task Standard: Calculate actual Steam Generator and Pressurizer levels during a cooldown per ABN-905A, Attachments 16 and 17.

Required Materials: ABN-905A, Loss of Control Room Habitability, Rev. 9-6.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **ABN-905A, Loss of Control Room Habitability.**
  - **Attachment 16, SG Level Temperature Correction.**
  - **Attachment 17, PRZR Level Temperature Correction.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**Steam Generator Level Graph is from ABN-905A, Attachment 16.**
Perform Step: 1√	CALCULATE Steam Generator <u>indicated</u> wide range level to maintain Steam Generator <u>actual</u> level between 74% and 83% when at 400ºF.
Standard:	CALCULATED Steam Generator <u>indicated</u> wide range level at **64% ±1%** to maintain Steam Generator <u>actual</u> level at 74% when at 400ºF.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√	CALCULATE Steam Generator <u>indicated</u> wide range level to maintain Steam Generator <u>actual</u> level between 74% and 83% when at 400ºF.
Standard:	CALCULATED Steam Generator <u>indicated</u> wide range level at **71% ±1%** to maintain Steam Generator <u>actual</u> level at 83% when at 400ºF.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Pressurizer Level Graph is from ABN-905A, Attachment 17.**
Perform Step: 3√	CALCULATE Pressurizer <u>indicated</u> level to maintain Pressurizer <u>actual</u> level between 25% and 50% when at 467ºF.
Standard:	CALCULATED Pressurizer <u>indicated</u> level at **24% ±1%** to maintain Pressurizer <u>actual</u> level at 25% when at 467ºF.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√	CALCULATE Pressurizer <u>indicated</u> level to maintain Pressurizer <u>actual</u> level between 25% and 50% when at 467ºF.
Standard:	CALCULATED Pressurizer <u>indicated</u> level at **57% ±1%** to maintain Pressurizer <u>actual</u> level at 50% when at 467ºF.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **The Unit 1 Control Room has been evacuated.**
- **ABN-905A, Loss of Control Room Habitability, is in progress.**
- **A cooldown is in progress from the Remote Shutdown Panel.**

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

- **CALCULATE Steam Generator <u>indicated</u> wide range level to maintain Steam Generator <u>actual</u> level between 74% and 83% when at 400ºF.**
- **CALCULATE Pressurizer <u>indicated</u> level to maintain Pressurizer <u>actual</u> level between 25% and 50% when at 467ºF.**

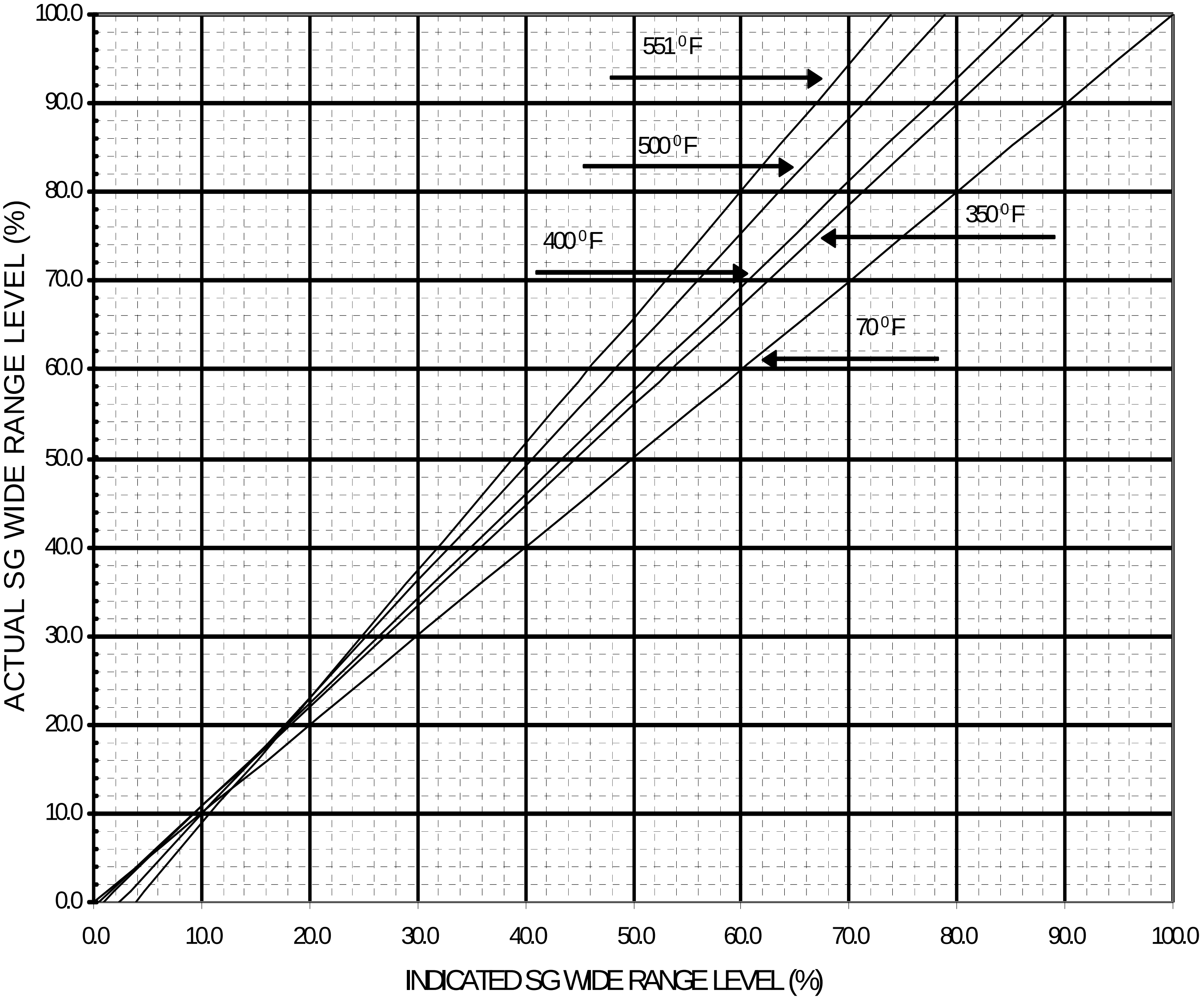
ATTACHMENT 16
PAGE 1 OF 1

SG LEVEL TEMPERATURE CORRECTION

NOTE: Normal SG level for Hot Standby and Cooldown (60 - 75% NR) is between 83% and 90% actual wide range. Operating outside this range could cause uncovering AFW nozzle OR ESF actuation OR moisture carryover. Approximate critical levels (actual wide range) are:

- Lo-Lo (ESF actuation) Unit 1 - 74%
- AFW Nozzle Unit 1 - 83%
- Hi-Hi (moisture carryover) 92%

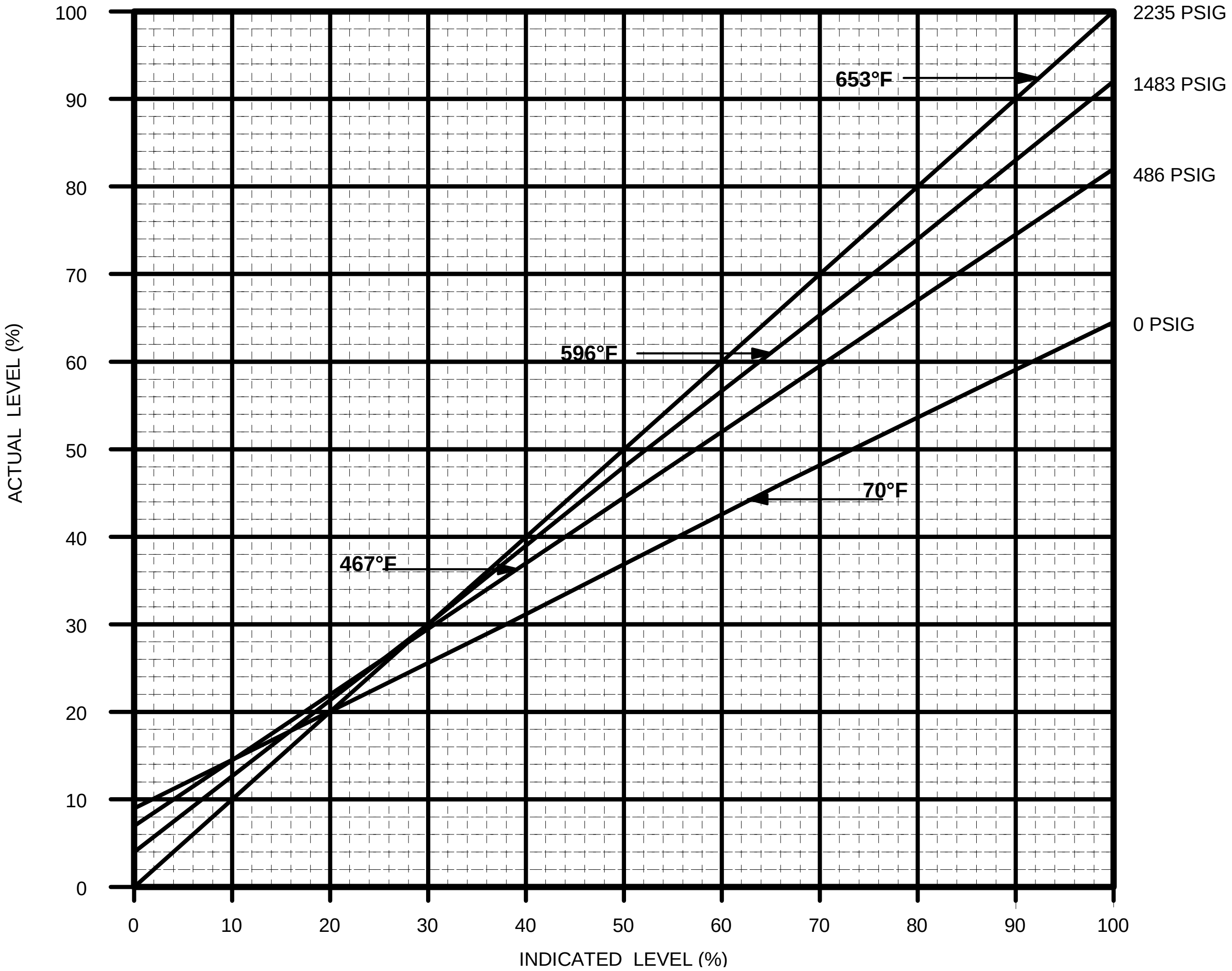
(L)



ATTACHMENT 17
PAGE 1 OF 1

PRZR LEVEL TEMPERATURE CORRECTION

PRESSURIZER LEVEL CHANNEL
(Hot Calibrated)
(LI-459B, LI-460B)



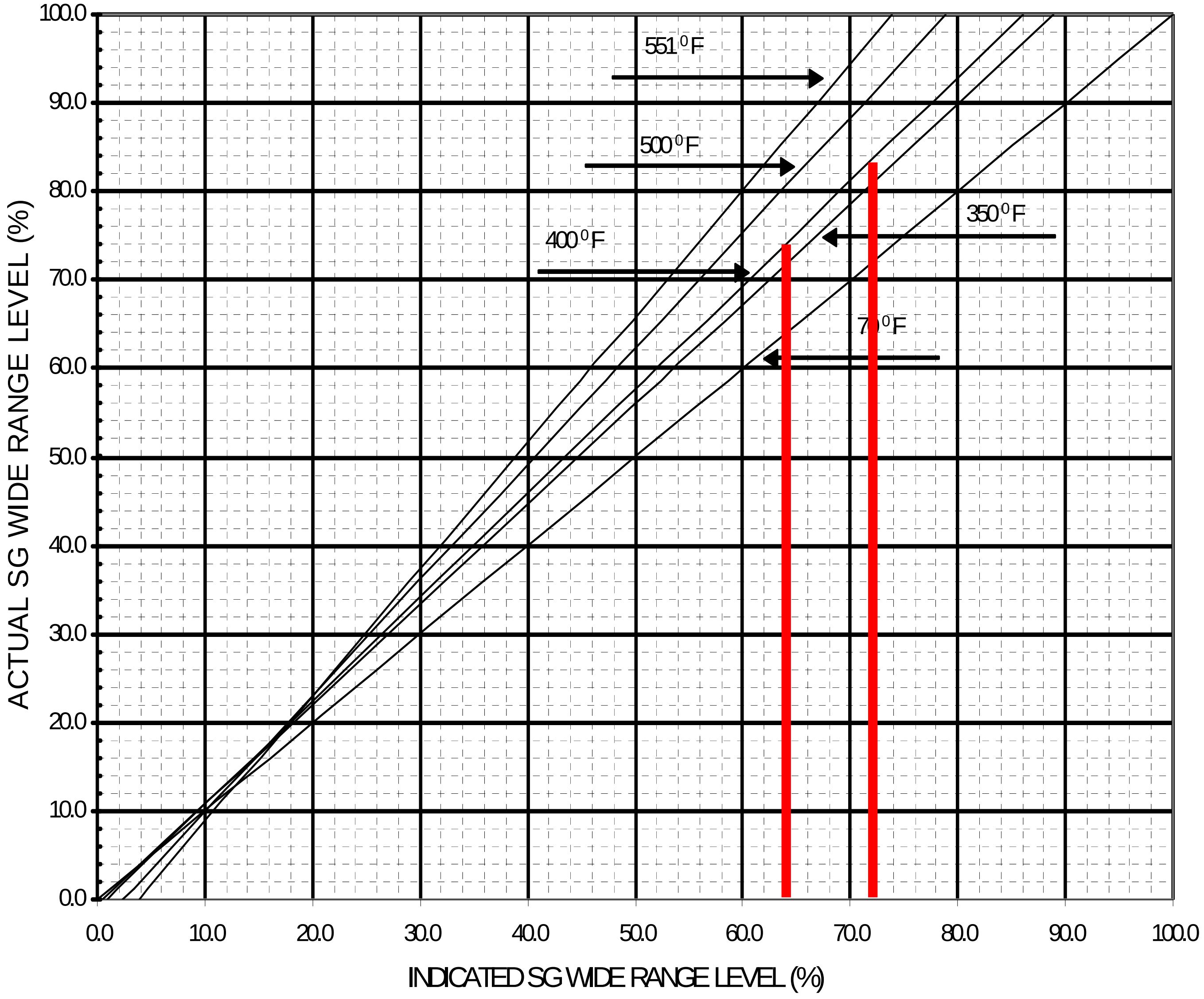
ATTACHMENT 16
PAGE 1 OF 1

SG LEVEL TEMPERATURE CORRECTION

NOTE: Normal SG level for Hot Standby and Cooldown (60 - 75% NR) is between 83% and 90% actual wide range. Operating outside this range could cause uncovering AFW nozzle OR ESF actuation OR moisture carryover. Approximate critical levels (actual wide range) are:

- Lo-Lo (ESF actuation) Unit 1 - 74%
- AFW Nozzle Unit 1 - 83%
- Hi-Hi (moisture carryover) 92%

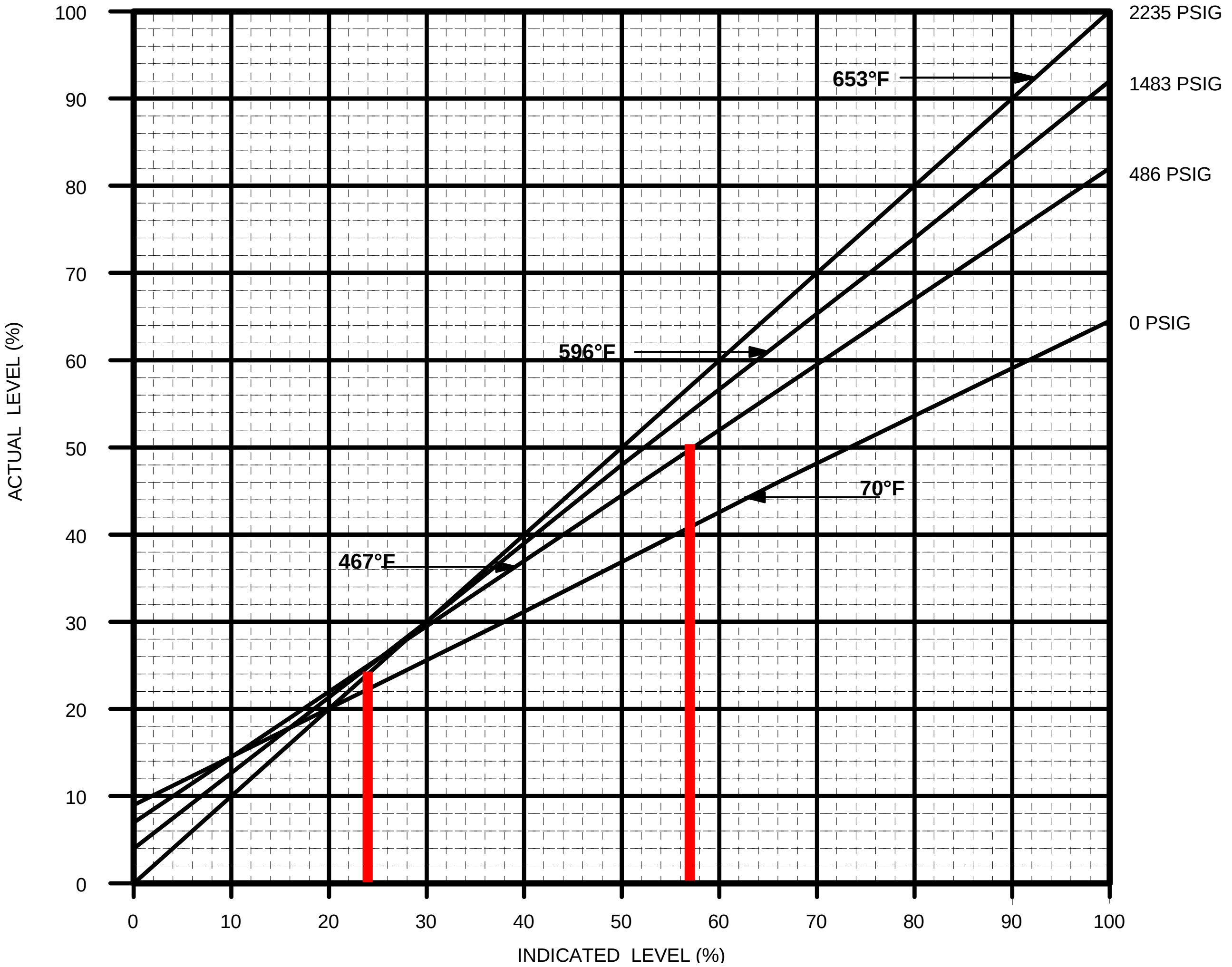
(L)



ATTACHMENT 17
PAGE 1 OF 1

PRZR LEVEL TEMPERATURE CORRECTION

PRESSURIZER LEVEL CHANNEL
(Hot Calibrated)
(LI-459B, LI-460B)



Facility: CPNPP JPM # NRC RA3 Task #RO1808 K/A #2.2.12 3.7 / 4.1

Title: Perform Axial Flux Difference Surveillance

Examinee (Print): ________________________________

Testing Method:

Simulated Performance: ______ Classroom: X

Actual Performance: X Simulator: ______

Alternate Path: ______ Plant: ______

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:

- Unit 1 is at 95% power.
- The Axial Flux Difference (AFD) alarm was declared INOPERABLE over 24 hours ago.
- Power Range Nuclear Instrument AFD data was collected for several hours last shift.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- PERFORM OPT-403, Axial Flux Difference.
- ENTER the Power Range Nuclear Instrument AFD data onto OPT-403-1, AFD Data Sheet.
- RECORD findings in the Discrepancies/Comments Section of OPT-403-1.

TIME	1-NI-41C	1-NI-42C	1-NI-43C	1-NI-44C
0800	9%	10%	11%	9%
0830	9%	11%	12%	10%
0900	10%	11%	12%	11%
0930	11%	14%	12%	14%
1000	11%	13%	12%	13%
1030	12%	14%	13%	14%

Task Standard: Perform Axial Flux Difference surveillance per OPT-403 and record findings on OPT-403-1.

Required Materials: OPT-403, Axial Flux Difference, Rev. 11.

OPT-403-1, AFD Data Sheet, Rev. 10-1.

NUC-204-6, Axial Flux Difference As a Function of Rated Thermal Power, Unit 1 Cycle 15, Rev. 07/26/10.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **OPT-403, Axial Flux Difference.**
- **OPT-403-1, AFD Data Sheet.**
- **NUC-204-6, Axial Flux Difference As a Function of Rated Thermal Power.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OPT-403, Section 8.0 and documented on Form OPT-403-1.**
Perform Step: 1 8.1 & 8.1.1	Record the following data for the affected unit: • Unit 1 or 2 as applicable
Standard:	CIRCLED Unit 1 on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2 8.1 & 8.1.2	Record the following data for the affected unit: • Date
Standard:	ENTERED Date on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3 8.2 & 8.2.1	Record the following data: • Time
Standard:	ENTERED Time on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4√ 8.2 & 8.2.2	Record the following data: • PR Δ FLUX for each operable excore detector
Standard:	RECORDED PR Δ FLUX for each operable excore detector on OPT-403-1 from JPM Cue Sheet.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5√ 8.2 & 8.2.3	Record the following data: • Percent Rated Thermal Power (RTP)
Standard:	RECORDED Percent Rated Thermal Power on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 6√ 8.3 & 8.3.1	Perform the following to determine PR Δ FLUX status and record: • Verify at least 3 of 4 PR Δ FLUX channels are within the Acceptable Operation region (Doghouse Region) of NUC-204-6 "Axial Flux Difference as a Function of Rated Thermal Power."
Standard:	DETERMINED PR Δ FLUX status and RECORDED and INITIALED on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 7√ 8.3 & 8.3.2	Perform the following to determine PR Δ FLUX status and record: • Repeat Steps 8.2 and 8.3 at least once per thirty (30) minutes.
Standard:	REPEATED Steps 8.2 and 8.3 at least once per thirty (30) minutes on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 8	Record findings in the Discrepancies/Comments Section of OPT-403-1.
Standard:	RECORDED findings in the Discrepancies / Comments Section of OPT-403-1.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is at 95% power.**
- **The Axial Flux Difference (AFD) alarm was declared INOPERABLE over 24 hours ago.**
- **Power Range Nuclear Instrument AFD data was collected for several hours last shift.**

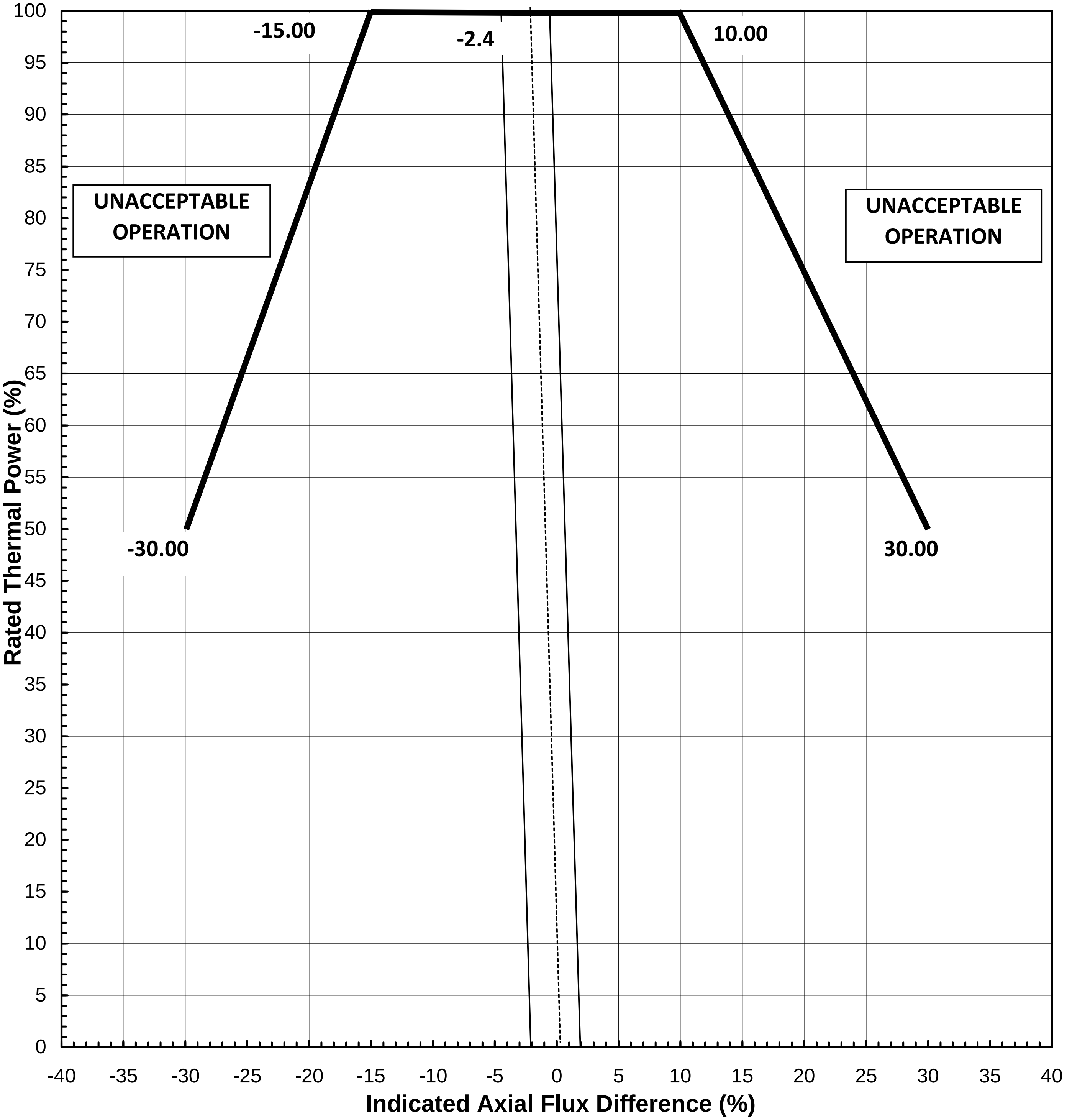
<u>INITIATING</u> <u>CUE</u>: **The Unit Supervisor directs you to PERFORM the following:**

- **PERFORM OPT-403, Axial Flux Difference.**
- **ENTER the Power Range Nuclear Instrument AFD data onto OPT-403-1, AFD Data Sheet.**
- **RECORD findings in the Discrepancies/Comments Section of OPT-403-1.**

TIME	1-NI-41C	1-NI-42C	1-NI-43C	1-NI-44C
0800	9%	10%	11%	9%
0830	9%	11%	12%	10%
0900	10%	11%	12%	11%
0930	11%	14%	12%	14%
1000	11%	13%	12%	13%
1030	12%	14%	13%	14%

FOR SIMULATOR USE ONLY

Axial Flux Difference as a Function of Rated Thermal Power
Unit 1, Cycle 15



Prepared By: **MOL - IC18** Date: ________

Approved By: **SIMULATOR USE ONLY** Date: ________

Core Performance Engineering Supervisor

NUC-204-6
Page 1 of 1
Rev. 2

REFERENCE USE

AFD DATA SHEET

8.1.1 UNIT: (Circle one) 1 2

8.1.2 DATE _____________

NOTE:
- PR Δ FLUX shall be considered outside the Acceptable Operations limits when two or more OPERABLE excore channels indicate Δ FLUX to be outside the Acceptable Operations limits.
- PR Δ FLUX logging frequency may be shortened at the discretion of the Shift Manager.
- Log PR Δ FLUX data at least once per 30 minutes. The 30 minute frequency satisfies the following TRS 13.2.32.1 requirements:
  - Once per hour for the first 24 hours that the AFD Monitor Alarm is inoperable.
  - Once per 30 minutes when the AFD Monitor Alarm is inoperable for > 24 hours.

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

AFD DATA SHEET

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

Additional copies of Page 2 of this data sheet may be used as required to document AFD status. ☐ Continued Next Page

AFD DATA SHEET

DISCREPANCIES / COMMENTS: ____________________

CORRECTIVE ACTIONS: ____________________

PERFORMED BY: ____________________ DATE: __________
SIGNATURE

REVIEWED BY: ____________________ DATE: __________
OPERATIONS MANAGEMENT

COMANCHE PEAK NUCLEAR POWER PLANT

UNIT COMMON

OPERATIONS TESTING MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE________________________

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

AXIAL FLUX DIFFERENCE

PROCEDURE NO. OPT-403

REVISION NO. 11

EFFECTIVE DATE: 10/08/08 1200

PREPARED BY (Print): JIM BRAU Ext: 5443

TECHNICAL REVIEW BY (Print): ROB SLOUGH Ext: 5727

APPROVED BY: D.W. McGAUGHEY for Dave Goodwin Date: 09/19/08
DIRECTOR, OPERATIONS

1.0 PURPOSE

This procedure satisfies Axial Flux Difference (AFD) monitoring when automated monitoring is NOT available. The requirements of TRS 13.2.32.1 is met by monitoring and logging indicated AFD for each OPERABLE excore channel. A <u>frequency of 30 minutes</u> for logging data is used to ensure the requirement is met if the AFD Monitor is inoperable for greater than 24 hours.

The actual TRS frequency requirements are as follows:

<u>NOTE</u>: The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

- Once per hour for the first 24 hours that the AFD Monitor Alarm is inoperable. (TRS 13.2.32.1).
- Once per 30 minutes when the AFD Monitor Alarm is inoperable for >24 hours. (TRS 13.2.32.1).

2.0 ACCEPTANCE CRITERIA

2.1 At > 50% RTP the indicated AFD is within the Acceptable Operation limit of NUC-204-6 “Axial Flux Difference as a Function of Rated Thermal Power.”

3.0 DEFINITIONS/ACRONYMS

3.1 APL^{ND} - Analyzed Power Limit Nuclear Design

3.2 AFD - Axial Flux Difference

3.3 RTP - Rated Thermal Power

3.4 Validated Computer Program - A computer program verified consistent with current procedures and technical data used to calculate AFD and the associated penalty time.

4.0 REFERENCES

4.1 Technical Specification 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)”

4.2 Technical Requirement 13.2.32, "AXIAL FLUX DIFFERENCE (AFD)"

4.3 FSAR Section 4.3, “Nuclear Design”

4.4 COLR, “Core Operating Limits Report”

4.5 NUC-204, "Target Axial Flux Difference"

4.6 NUC-204-6, “Axial Flux Difference as a Function of Rated Thermal Power”

4.7 SOP-906, “Plant Process Computer System Guidelines”

5.0 PRECAUTIONS, LIMITATIONS AND NOTES

5.1 Precautions

None

5.2 Limitations

5.2.1 Core Performance Engineering shall be notified when any acceptance criteria not satisfied.

5.3 Notes

None

6.0 PREREQUISITES

☐ 6.1 MODE 1 at > 50% RTP.

7.0 TEST EQUIPMENT

None

8.0 INSTRUCTIONS

NOTE: Record all data on Form OPT-403-1.

8.1 Record the following data for the affected unit:

☐ 8.1.1 Unit 1 or 2 as applicable

☐ 8.1.2 Date

8.2 Record the following data:

☐ 8.2.1 Time

☐ 8.2.2 PR Δ FLUX for each operable excore detector

☐ 8.2.3 Percent Rated Thermal Power (RTP)

8.3 Perform the following to determine PR Δ FLUX status and record:

☐ A. Verify at least 3 of 4 PR Δ FLUX channels are within the Acceptable Operation region (Doghouse Region) of NUC-204-6 “Axial Flux Difference as a Function of Rated Thermal Power.”

[C] ☐ B. Repeat Steps 8.2 and 8.3 at least once per thirty (30) minutes.

9.0 RESTORATION/POST WORK ACTIVITIES

None

10.0 ATTACHMENTS/FORMS

10.1 Attachments

None

10.2 Forms

10.2.1 OPT-403-1, AFD Data Sheet

AFD DATA SHEET

8.1.1 UNIT: (Circle one) [✓ 1] 2

8.1.2 DATE TODAY

NOTE:
- PR Δ FLUX shall be considered outside the Acceptable Operations limits when two or more OPERABLE excore channels indicate Δ FLUX to be outside the Acceptable Operations limits.
- PR Δ FLUX logging frequency may be shortened at the discretion of the Shift Manager.
- Log PR Δ FLUX data at least once per 30 minutes. The 30 minute frequency satisfies the following TRS 13.2.32.1 requirements:
  - Once per hour for the first 24 hours that the AFD Monitor Alarm is inoperable.
  - Once per 30 minutes when the AFD Monitor Alarm is inoperable for > 24 hours.

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
0800	9%	10%	11%	9%	95	[✓ YES] NO	LZ
0830	9%	11%	12%	10%	95	[✓ YES] NO	LZ
0900	10%	11%	12%	11%	95	[✓ YES] NO	LZ
0930	11%	14%	12%	14%	95	YES [✓ NO]	LZ
1000	11%	13%	12%	13%	95	YES [✓ NO]	LZ
1030	12%	14%	13%	14%	95	YES [✓ NO]	LZ
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

AFD DATA SHEET

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

Additional copies of Page 2 of this data sheet may be used as required to document AFD status. ☐ Continued Next Page

AFD DATA SHEET

DISCREPANCIES / COMMENTS: Acceptance Criteria not met starting at 0930.

CORRECTIVE ACTIONS:

PERFORMED BY: ______________________ DATE: __________
SIGNATURE

REVIEWED BY: ______________________ DATE: __________
OPERATIONS MANAGEMENT

OPT-403-1
PAGE 3 OF 3
R-10

Facility: CPNPP JPM # NRC RA4 Task #RWT029 K/A #2.3.12 3.2 / 3.7
Title: Determine Radiation Levels During Maintenance and Administrative Exposure Limit

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______

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: JPM Cue Sheet #1

Given the following conditions:

- A high dose maintenance activity is scheduled in the Fuel Building.
- The general dose rate in the area is 100 mrem/hour but can be reduced to 25 mrem/hour if lead shielding is installed.
- It will take Nuclear Equipment Operators (NEOs) Alpha & Bravo 30 minutes to install the shielding.
- Independent of the shielding, it will take NEO Alpha two (2) hours or NEOs Alpha & Bravo one and a half (1.5) hours to perform the maintenance.

Initiating Cue:

The Work Control Supervisor directs you to PERFORM the following:

- CALCULATE the dose received when performing the maintenance for each of the following conditions:
  - NEO Alpha **without** shielding. __________ mrem.
  - NEOs Alpha & Bravo **without** shielding. __________ mrem.
  - NEO Alpha **with** shielding. __________ mrem.
  - NEOs Alpha & Bravo **with** shielding. __________ mrem

Initial Conditions: JPM Cue Sheet #2

Given the following conditions:

- It was determined that NEO Alpha received a Total Effective Dose Equivalent (TEDE) of 225 mrem while performing the maintenance task.
- NEO Alpha’s year to date whole body exposure is 1785 mrem.

Initiating Cue:

The Shift Manager directs you to PERFORM the following:

- IDENTIFY if any applicable CPNPP Administrative Exposure Levels have been exceeded.
- REPORT findings to the Shift Manager.

Task Standard:

Calculate the dose received when performing the maintenance and determine if an Administrative Exposure Level was exceeded per STA-655.

Required Materials: STA-655, Exposure Monitoring Program, Rev. 19.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee JPM Cue Sheet #1.

When JPM Cue Sheet #1 is completed, PROVIDE JPM Cue Sheet #2.

ENSURE examinee has a calculator.

ENSURE copy of STA-655, Exposure Monitoring Program is available.

√ - Check Mark Denotes Critical Step

START TIME:	

Perform Step: 1√	Determine total dose to NEO Alpha <u>without</u> shielding.
Standard:	DETERMINED total dose to NEO Alpha <u>without</u> shielding as follows: • 100 mrem/hr x 2 hours = **200 mrem total dose.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√	Determine total combined dose to NEOs Alpha & Bravo <u>without</u> shielding.
Standard:	DETERMINED total combined dose to NEOs Alpha & Bravo <u>without</u> shielding as follows: • 100 mrem/hr x 1.5 hours/NEO x 2 NEOs = **300 mrem total dose.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3	Determine total dose to <u>install</u> shielding.
Standard:	DETERMINED total dose to <u>install</u> shielding as follows: • 100 mrem/hr x 0.5 hours/NEO x 2 NEOs = **100 mrem to install.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√	Determine total dose to NEO Alpha <u>with</u> shielding.
Standard:	DETERMINED total dose to NEO Alpha <u>with</u> shielding as follows: • 25 mrem/hr x 2 hours + 100 mrem = **150 mrem total dose.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√	Determine total combined dose to NEOs Alpha & Bravo <u>with</u> shielding.
Standard:	DETERMINED total combined dose to NEOs Alpha & Bravo <u>with</u> shielding as follows: • 25 mrem/hr x 1.5 hours/NEO x 2 NEOs + 100 mrem = **175 mrem total dose.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Provide the examinee with copy of JPM Cue Sheet #2.**
Perform Step: 6√	Determine if any applicable CPNPP Administrative Exposure Levels have been exceeded.
Standard:	DETERMINED that the CPNPP TEDE Administrative Exposure Limit of **2000 mrem** was exceeded per STA-655, Exposure Monitoring Program. **1785 mrem = 225 mrem = 2010 mrem.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7	Report findings to Shift Manager.
Standard:	REPORTED to Shift Manager that the CPNPP TEDE Administrative Exposure Limit of 2000 mrem was exceeded.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **A high dose maintenance activity is scheduled in the Fuel Building.**
- **The general dose rate in the area is 100 mrem/hour but can be reduced to 25 mrem/hour if lead shielding is installed.**
- **It will take Nuclear Equipment Operators (NEOs) Alpha & Bravo 30 minutes to install the shielding.**
- **Independent of the shielding, it will take NEO Alpha two (2) hours or NEOs Alpha & Bravo one and a half (1.5) hours to perform the maintenance.**

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

- **CALCULATE the dose received when performing the maintenance for each of the following conditions:**
  - **NEO Alpha without shielding. _____ mrem.**
  - **NEOs Alpha & Bravo without shielding. _____ mrem.**
  - **NEO Alpha with shielding. _____ mrem.**
  - **NEOs Alpha & Bravo with shielding. _____ mrem.**

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

- **It was determined that NEO Alpha received a Total Effective Dose Equivalent (TEDE) of 225 mrem while performing the maintenance task.**
- **NEO Alpha's year to date whole body exposure is 1785 mrem.**

<u>INITIATING</u> <u>CUE</u>: **The Shift Manager directs you to PERFORM the following:**

- **IDENTIFY if any applicable CPNPP Administrative Exposure Levels have been exceeded.**
- **REPORT findings to the Shift Manager.**

CPNPP STATION ADMINISTRATION		PROCEDURE NO. STA-655
EXPOSURE MONITORING PROGRAM	REVISION NO. 19	Page 23 of 29

ATTACHMENT 8.A
PAGE 1 OF 2

ADMINISTRATIVE EXPOSURE LEVELS
DEEP DOSE

RADIATION WORKERS

PERIOD	CALCULATION	LEVEL
Annual	TEDE (Total Effective Dose Equivalent)	2000 mrem
Annual	TODE - (The SUM of Deep-Dose Equivalent and Committed Dose Equivalent to any individual organ or tissue other than the lens of the eye).	20,000 mrem

PERIOD	EVENT	LEVEL
Annual	Planned Special Exposure (PSE)	4000 mrem
	NOT TO EXCEED:	
Lifetime	Planned Special Exposure (PSE)	Five times the annual dose limit.

ATTACHMENT 8.A
PAGE 2 OF 2

ADMINISTRATIVE EXPOSURE LEVELS
DEEP DOSE

EMBRYO/FETUS OF DECLARED PREGNANT RADIATION WORKER

PERIOD	RECEPTOR	LEVEL
Gestation	Declared Radiation Worker **OR:** Declared Escorted Radiation Worker	200 mrem (Not to exceed 50mrem/month)

NOTE: **If the dose to the embryo/fetus is found to have exceeded 200 mrem by the time the woman declares pregnancy, then any additional dose should not exceed 50 mrem during the remainder of the pregnancy.**

NOTE: **Administrative Exposure Levels are based on Electronic Dosimeter estimates.**

ESCORTED RADIATION WORKERS

PERIOD	CALCULATION	LEVEL
Monitoring Period	DDE (Deep Dose Equivalent) (with OSL badge)	100 mrem
	With appropriate authorization:	
Annual	DDE (Deep Dose Equivalent) (with OSL badge)	≤ 2000 mrem

MEMBER OF THE PUBLIC (VISTOR)

PERIOD	CALCULATION	LEVEL
Quarter	DDE (Deep Dose Equivalent)	20 mrem

NOTE: **A Visitor is not allowed into a contaminated or airborne area and therefore a committed dose equivalent should not be calculated.**

NOTE: **Administrative Exposure Levels are based on Electronic Dosimeter estimates.**

Facility: CPNPP JPM # NRC SA1 Task #SO1017 K/A #2.1.43 4.1 / 4.3

Title: Determine Reactivity Effects When Starting Positive Displacement Charging Pump

Examinee (Print): ________________________________

Testing Method:

Simulated Performance: ______ Classroom: X

Actual Performance: X Simulator: ______

Alternate Path: ______ Plant: ______

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:

- Unit 1 is operating at approximately 3% power.
- The Positive Displacement Charging Pump must be placed in service per SOP-103A, Chemical and Volume Control System.
- The Reactor Coolant System Boron concentration the last time the Positive Displacement Charging Pump was run was 2400 ppm.
- Current Reactor Coolant System Boron concentration is 1545 ppm.
- Chemistry has just reported the Unit 1 Refueling Water Storage Tank boron concentration is 2700 ppm.

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

- CALCULATE a Reactivity Evaluation for starting the Positive Displacement Charging Pump per SOP-103A, Chemical and Volume Control System, Steps 5.3.1.C and 5.3.1.D.
- REPORT your findings to the Shift Manager.
- IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time associated with the REPORT from Chemistry, if any.
  - LCO ______________
  - REQUIRED ACTION ______________
  - COMPLETION TIME ______________

Task Standard: Calculate the change in boron and resultant change in temperature when placing the Positive Displacement Pump in service per SOP-103A and identify any Technical Specification Limiting Condition for Operation, Required Action and Completion Time.

Required Materials: SOP-103A, Chemical and Volume Control System, Rev. 17-23.

CPNPP Technical Specifications Units 1 and 2, Amendment 152.

Reactivity Briefing Sheet for 1545 ppm Reactor Coolant System conditions.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **SOP-103A, Chemical and Volume Control System, Section 5.3.1.**
  - **INITIALED up to Step 5.3.1.C.**
- **89.8 EFPD Reactivity Briefing Sheet.**
- **CPNPP Technical Specifications - Units 1 and 2.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from SOP-103A, Step 5.3.1.**
NOTE: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.	
Perform Step: 1√ 5.3.1.C 1st line + calculation	IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows: • ΔB = RCS Boron Concentration Difference
Standard:	CALCULATED ΔB = RCS Boron Concentration Difference as follows: ΔB = (500 ppm PDP - 1545 ppm RCS) x 0.00128 = **1.0944 ppm**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of ITC and HFP Differential Boron Worth may occur.**
Perform Step: 2√ 5.3.1.C 2nd line + calculation	On the Reactivity Briefing Sheet get the following information: ITC _____ pcm/ºF HFP Differential Boron Worth _____ pcm/ppm
Standard:	DETERMINED the following from the Reactivity Briefing Sheet: ITC = **- 1.1 ± 0.1 pcm /ºF** HFP Differential Boron Worth = **- 6.9 ± 0.1 pcm / ppm**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of ITC / HFP Differential Boron Worth may occur.**
Perform Step: 3√ 5.3.1.C 2nd line + calculation	On the Reactivity Briefing Sheet get the following information: ITC / HFP Differential Boron Worth = ppm / ºF
Standard:	CALCULATED change in ppm / ºF: - 1.1 pcm / ºF / - 6.9 pcm / ppm = **0.1594 ± 0.02 ppm / ºF**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Rounding off of ΔT_{AVE} may occur.**
Perform Step: 4√ 5.3.1.C 3rd line + calculation	On the Reactivity Briefing Sheet get the following information: $\Delta T_{AVE} = \Delta B$ / ppm / ºF
Standard:	CALCULATED change in T_{AVE} as follows: $\Delta T_{AVE} = \Delta B$ / ppm / ºF = 1.0944 ppm / 0.1594 ppm / ºF = **6.87 ± 1.0 ºF**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 5.3.1.D	<u>IF</u> ΔT_{AVE} calculated above is >1ºF, <u>THEN</u> notify Shift Operations Manager to discuss contingency actions.
Standard:	DETERMINED ΔT_{AVE} calculated is greater than 1ºF and NOTIFIED Shift Operations Manager.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√	Identify Technical Specification Limiting Condition for Operation.
Standard:	RECOGNIZED RWST boron concentration greater than 2600 ppm and DETERMINED the following: • Technical Specification LCO 3.5.4, Refueling Water Storage Tank CONDITION A, RWST boron concentration not within limits.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7√	Identify Technical Specification REQUIRED ACTION and COMPLETION TIME.
Standard:	DETERMINED Technical Specification REQUIRED ACTION and COMPLETION TIME: • 3.5.4.A.1 - Restore RWST to OPERABLE status within 8 hours.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 is operating at approximately 3% power.**
- **The Positive Displacement Charging Pump must be placed in service per SOP-103A, Chemical and Volume Control System.**
- **The Reactor Coolant System Boron concentration the last time the Positive Displacement Charging Pump was run was 2400 ppm.**
- **Current Reactor Coolant System Boron concentration is 1545 ppm.**
- **Chemistry has just reported the Unit 1 Refueling Water Storage Tank boron concentration is 2700 ppm.**

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

- **CALCULATE a Reactivity Evaluation for starting the Positive Displacement Charging Pump per SOP-103A, Chemical and Volume Control System, Steps 5.3.1.C and 5.3.1.D.**
- **REPORT your findings to the Shift Manager.**
- **IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time associated with the REPORT from Chemistry, if any.**
  - **LCO** ________________
  - **REQUIRED ACTION** ________________
  - **COMPLETION TIME** ________________

Reactivity Briefing Sheet for Stable Operation
FOR SIMULATOR USE ONLY

Calculations based on core design values, and assume:

Burnup = **4001.3** MWD/MTU
89.82 EFPD
Power = **3** RTP
Boron = **1545** ppm
B10 Conc = **0.1776696** w/o
Control Bank D = **100** steps

Burnup in the BOL range

NOTE: Re-create the Briefing Sheet if current values significantly differ from assumed inputs.

Reactivity affects of Control Bank D

HFP Diff Worth @ 100.0 steps = **-4.3** pcm / step

HFP Integral Rod Worth for CBD Step Positions:

Steps	pcm
110	-269.0
109	-272.4
108	-275.9
107	-279.4
106	-283.1
105	-286.9
104	-290.7

Steps	pcm
103	-294.7
102	-298.7
101	-302.9
100	-307.1
99	-311.5
98	-316.0
97	-320.5

Steps	pcm
96	-325.2
95	-329.9
94	-334.8
93	-339.8
92	-344.8
91	-350.0
90	-355.2

Steps	pcm
85	-382.5
80	-411.7
75	-442.7
70	-475.3
65	-509.2
60	-544.2
55	-580.3

Reactivity affects of Boron

HFP Diff Boron Worth @ 1545 ppm = **-6.9** pcm / ppm

1-FK-110 Pot Setting for Blended Flow @ 1545 ppm = **6.34**
(Assuming BAT concentration of 7492.0 ppm)

Reactivity affects of Power

Power Coefficient of Reactivity = **-13.1** pcm / % RTP

Dilution to equal 1% Power Increase = **83.8** gallons RMUW

Boration to equal 1% Power Decrease = **20.9** gallons boric acid

Reactivity affects of RCS Temperature

Temperature Coefficient of Reactivity (ITC) = **-1.1** pcm / °F

Boration to equal 1°F Temperature Decrease = **1.8** gallons boric acid

Dilution to equal 1°F Temperature Increase = **7.3** gallons RMUW

Load Reduction equal to 1°F T_{ave} Increase = **1.0** MWe

COMANCHE PEAK NUCLEAR POWER PLANT

UNIT 1

SYSTEM OPERATING PROCEDURE MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN-19 07-28-2010 1200

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

CHEMICAL AND VOLUME CONTROL SYSTEM

PROCEDURE NO. SOP-103A

REVISION NO. 17

EFFECTIVE DATE: 03-05-2008 1200

PREPARED BY (Print): Brad Hancock Ext: 6769

TECHNICAL REVIEW BY (Print): Lisabeth Donley Ext: 6524

APPROVED BY: Alan Hall for Dave Goodwin Date: 02-19-2008
DIRECTOR, OPERATIONS

5.3 Major Component Operation

5.3.1 Positive Displacement Pump Startup

This section describes the steps to place the PDP in service.

CAUTION: PDP operation may result in high gaseous activity in the PDP Room due to packing leakage.

NOTE:

- PDP run time should be minimized to conserve pump packing. Run PDP only when required (for example - Slave Relay Testing).
- Following loss of Instrument Air, control air to the PDP fluid drive must be reset. This is done by depressing the RESET pushbutton on the instrument air supply to the PDP fluid drive. This RESET is normally accomplished by ABN-301 restoration section 3.0.
- The reactivity impact for starting the PDP pump is typically very small due to diffusion effects between the PDP piping and the RCS. However, assuming no diffusion, the reactivity effects could potentially approach -15 pcm (and -1.5 °F temperature change) with very large (>1000 ppm) boron concentration differences between the PDP piping and RCS. (EVAL-04-0944-04)
- Following several PDP starts, the PDP fluid drive oil may exceed the upper limit mark on the drive unit's sight glass due to priming) the pump (adding oil) over a period of time. The PROMPT Team should be contacted to drain the excess oil. (SMF-01-0600 and 4-03-149515-00)
- With the PDP stopped, oil level should be in the upper 1/4 of the MIN - MAX range, preferably near the MAX level mark. (SMF-05-2603)

☑ A. Ensure the prerequisites in Section 2.5 are met.

B. IF the PDP has not operated for an extended period (month), THEN prime the PDP fluid drive by performing the following:

N/A ☐ 1) IF pump hydraulic fluid level is at the maximum level of the sightglass, THEN instruct a PROMPT member to drain ~1/2 liter of oil into a clean container. (This oil will be added to the fluid drive at step 5.3.1.B. 3)

☐ 2) Remove the pipe plugs from the two priming holes on top of the input end bell (motor side of the fluid drive).

☐ 3) Pour oil (collected in step 5.3.1.B. 1) a) and/or from the approved lubrication list) into either hole until oil rises to the bottom of the other hole and remains there.

☐ 4) Replace and tighten the pipe plugs.

5.3.1

> <u>NOTE</u>: Steps C and D may be considered NA if in a MODE other than MODE 1 or 2.

☐ C. <u>IF</u> the RCS boron concentration has changed significantly since the PDP was last operated, <u>THEN</u> determine the impact of water in the PDP piping on reactivity as follows:

> <u>NOTE</u>: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.

ΔB = RCS Boron Concentration Difference

ΔB = (_______ ppm PDP - ________ ppm RCS) x 0.00128

ΔB = _______ ppm

On the Reactivity Briefing Sheet get the following information:

ITC_____________ pcm/°F HFP Differential Boron Worth__________ pcm/ppm

$$\frac{\text{ITC}}{\text{HFP Differential Boron Worth}} = \frac{\_\_\_\_\_\_\_\_\_\_\_\_\text{ pcm/°F}}{\text{pcm/ppm}} = \_\_\_\_\_\_\_\_\_\_\_\text{ ppm/°F}$$

$$\Delta\text{Tave} = \frac{\triangle\text{B}}{\text{ppm/°F}} = \frac{\_\_\_\_\_\_\_\_\_\_\_\_\text{ ppm}}{\text{ppm/°F}} = \_\_\_\_\_\_\_\_\_\_\_\_\_\_\_\text{ °F}$$

☐ D. <u>IF</u> ΔTave calculated above is >1°F, <u>THEN</u> notify Shift Operations Manager to discuss contingency actions.

> <u>NOTE</u>: If the Stuffing Box Coolant Tank is overfilled, the PDP Charging Pump Room will become contaminated.

E. <u>IF</u> Stuffing Box Coolant Tank is low, <u>THEN</u> fill per the following steps:

☐ 1) Slowly crack OPEN 1CS-0119, PD PMP 1-01 STUFFING BOX COOL TK MU ISOL VLV, until desired fill rate is achieved.

☐ 2) When the desired tank level has been established, CLOSE 1CS-0119.

☐ F. Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.

G. OPEN the following valves:

☐ ● 1/1-8202A, VENT VLV (MCB)

☐ ● 1/1-8202B, VENT VLV (MCB)

5.3.1

☐ H. Ensure 1APPD, POSITIVE DISPLACEMENT CHARGING PUMP 1-01 MOTOR BREAKER 1EB1/2B/BKR is racked to the CONNECT position.

☐ I. Place 1-SK-459A, PDP SPD CTRL, in MANUAL with demand at 55%.

NOTE: The PDP will not start until 1-8109, PD CHRG PMP 1-01 RECIRC VLV, is open and handswitch 1/1-APPD, PDP, is in the START position. Two minutes after the PDP breaker is closed, 1-8109 will automatically close.

☐ J. OPEN 1/1-8109, PDP RECIRC VLV.

NOTE: PDP speed may have to be raised rapidly when a CCP is also in operation to prevent the PDP from stalling on low oil pressure.

☐ K. WHEN 1/1-8109, PDP RECIRC VLV is open, THEN start the PDP by placing handswitch 1/1-APPD PDP, to the START position.

☐ L. Ensure 1/1-8109, PDP RECIRC VLV, is CLOSED.

NOTE: During PDP operation the following step may be performed to lower PDP suction stabilizer level.

M. IF 1/1-8204, H2/N2 SPLY VLV indicates OPEN (red light on), THEN perform the following to lower suction stabilizer level:

[C] ☐ ● OPEN 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV for no more than 10 seconds to clear the high level, then close.

N. IF a CCP is in operation AND it is to be placed in standby, THEN perform the following:

☐ 1) Ensure only ONE letdown orifice is in service per Section 5.2.3.

☐ 2 Alternately raise PDP speed using 1-SK-459A, PDP SPD CTRL, and lower CCP flow using 1-FK-121, CCP CHRG FLO CTRL, until 1-FK-121 is at minimum.

☐ 3) Shut down the running CCP per Section 5.3.4.

[IV] ☐ O. IF desired, THEN gradually adjust 1-SK-459A, PDP SPD CTRL, to achieve the required flow rate AND place in AUTO.

☐ P. Adjust 1-LK-459, PRZR LVL CTRL, as necessary to maintain stable Pressurizer level.

COMMENTS ____

____

____

5.3.1

> NOTE: Steps C and D may be considered NA if in a MODE other than MODE 1 or 2.

☐ C. IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows:

> NOTE: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.

ΔB = RCS Boron Concentration Difference

ΔB = (2400 ppm PDP - 1545 ppm RCS) x 0.00128

ΔB = 1.0944 ppm

On the Reactivity Briefing Sheet get the following information:

ITC -1.1 +/- 0.1 pcm/°F

HFP Differential Boron Worth - 6.9 +/- 0.1 pcm/ppm

$$\frac{\text{ITC}}{\text{HFP Differential Boron Worth}} = \frac{-1.1 \text{ +/- } 0.1 \text{ pcm/°F}}{-6.9 \text{ +/- } 0.1 \text{ pcm/ppm}} = 0.1594 \text{ +/- } 0.02 \text{ ppm/°F}$$

$$\Delta\text{Tave} = \frac{\Delta\text{B}}{\text{ppm/°F}} = \frac{1.0944 \text{ ppm}}{0.1594 \text{ +/- } 0.02 \text{ ppm/°F}} = 6.87 \text{ +/- } 1.0 \text{ °F}$$

☐ D. IF ΔTave calculated above is >1°F, THEN notify Shift Operations Manager to discuss contingency actions.

> NOTE: If the Stuffing Box Coolant Tank is overfilled, the PDP Charging Pump Room will become contaminated.

E. IF Stuffing Box Coolant Tank is low, THEN fill per the following steps:

☐ 1) Slowly crack OPEN 1CS-0119, PD PMP 1-01 STUFFING BOX COOL TK MU ISOL VLV, until desired fill rate is achieved.

☐ 2) When the desired tank level has been established, CLOSE 1CS-0119.

☐ F. Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.

G. OPEN the following valves:

☐ ● 1/1-8202A, VENT VLV (MCB)

☐ ● 1/1-8202B, VENT VLV (MCB)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	-----------------------------------NOTE----------------------------------- Only required to be performed when ambient air temperature is < 40°F or > 120°F. --- Verify RWST borated water temperature is ≥ 40°F and ≤ 120°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 473,731 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	7 days

Facility: CPNPP JPM # SRO NRC SA2 Task #SO1005 K/A #2.1.1 3.8 / 4.2
Title: Determine Technical Specification and Event Reportability

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______

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:

- Both Units are at 100% power.
- A prolonged heat wave has raised Station Service Water Intake temperature to 105ºF.

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

- IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time, if any.
  - LCO _______________
  - REQUIRED ACTION _______________
  - COMPLETION TIME _______________
- DETERMINE Oral and Written Reportability Requirements, if any.
  - Oral Reporting Requirement _______________
  - Written Reporting Requirement _______________

Task Standard: Determine Technical Specifications impacted and Reportability Requirements for an INOPERABLE Ultimate Heat Sink per STA-501 and Technical Specifications.

Required Materials: STA-501, Nonroutine Reporting, Rev. 14-5.
CPNPP Technical Specifications Units 1 and 2, Amendment 152.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **STA-501, Nonroutine Reporting.**
- **CPNPP Technical Specifications - Units 1 and 2.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from CPNPP Technical Specifications, Amendment 152.**
Perform Step: 1√	Identify Technical Specification Limiting Condition for Operation.
Standard:	RECOGNIZED Safe Shutdown Impoundment INOPERABLE due to Station Service Water temperatures greater than 102ºF and DETERMINED the following: • Technical Specification LCO 3.7.9, Ultimate Heat Sink, CONDITION B, SSI inoperable for reasons other than Condition A.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√	Identify Technical Specification REQUIRED ACTION and COMPLETION TIME.
Standard:	DETERMINED Technical Specification REQUIRED ACTION and COMPLETION TIME: • 3.7.9.B.1 - Be in MODE 3 within 6 hours, AND • 3.7.9.B.2 - Be in MODE 5 within 36 hours.
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**The following steps are from STA-501, Attachment 8.D/13.**
Perform Step: 3√ Attachment 8.D/13 Page 4 of 16	Determine oral Reporting Requirements per STA-501.
Standard:	DETERMINED oral Reporting Requirements per STA-501: "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications [Note: To Mode 3]." • Oral Report within 4 hours per 10CFR50.72 (b) (2) (i).
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4√ Attachment 8.D/13 Page 4 of 16	Determine written Reporting Requirements per STA-501.
Standard:	DETERMINED written Reporting Requirement per STA-501: "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications [Note: To Mode 3]." • Written Report (LER) within 60 days per 10CFR50.73 (a) (2) (i) (a).
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Both Units are at 100% power.**
- **A prolonged heat wave has raised Station Service Water Intake temperature to 105ºF.**

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

- **IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time, if any.**
  - **LCO** ________________
  - **REQUIRED ACTION** ________________
  - **COMPLETION TIME** ________________
- **DETERMINE Oral and Written Reportability Requirements, if any.**
  - **Oral Reporting Requirement** ________________
  - **Written Reporting Requirement** ________________

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The Safe Shutdown Impoundment (SSI) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SSI level less than required.	A.1 Restore SSI level to within limits.	7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SSI inoperable for reasons other than Condition A.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of SSI is ≥ 770 ft mean sea level.	24 hours
SR 3.7.9.2	Verify station service water intake temperature is ≤ 102°F.	24 hours

ATTACHMENT 8.D/13
PAGE 4 OF 16

10CFR50.72 and 10CFR50.73 MATRIX

10CFR50.72	hrs	10CFR50.73	LER
(a)(1)(i) The declaration of any of the Emergency Classes specified in the Emergency Plan: (after notification of state and local agencies).	Less than an hour	*There is no requirement in 10CFR50.73 to report the declaration of an Emergency Class. However, an event or condition that leads to declaration of an Emergency Class may meet one or more of the specific reporting requirements that are in 10CFR50.73*	
(a)(1)(ii) Those non-emergency events specified in paragraph (b) of this section that occurred within three years of the date of discovery.		*There is usually a parallel reporting requirement in 10CFR50.73 that captures a non-emergency event that is reportable under 10CFR50.72. Exceptions are: a press release; notification to another government agency; transport of a contaminated person offsite; and loss of emergency preparedness capability.*	
(a)(2) If the Emergency Notification System is inoperative, the licensee shall make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Operations Center		There is no corresponding requirement in 10CFR50.73	
(a)(3) The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.		*There is usually a parallel reporting requirement in 10CFR50.73 that captures a non-emergency event that is reportable under 10CFR50.72. Exceptions are: a press release; notification to another government agency; transport of a contaminated person offsite; and loss of emergency preparedness capability.*	
(a)(4) The licensee shall activate the Emergency Response Data System (ERDS) as soon as possible but not later than one hour after declaring an Emergency Class of alert, site area emergency, or general emergency. The ERDS may also be activated by the licensee during emergency drills or exercises if the licensee's computer system has the capability to transmit the exercise data.		*There is usually a parallel reporting requirement in 10CFR50.73 that captures a non-emergency event that is reportable under 10CFR50.72. Exceptions are: a press release; notification to another government agency; transport of a contaminated person offsite; and loss of emergency preparedness capability.*	
(b)(1) Any deviation from the plant's Technical Specifications authorized pursuant to 10CFR50.54(x).	1	**(a)(2)(i)(C)** Any deviation from the plant's Technical Specifications authorized pursuant to 10CFR50.54(x).	60 day LER
(b)(2)(i) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications [Note: To Mode 3]	4	**(a)(2)(i)(A)** The completion of any nuclear plant shutdown required by the plant's Technical Specifications	60 day LER

Facility: CPNPP JPM # NRC SA3 Task #SO1202 K/A #2.2.12 3.7 / 4.1

Title: Perform Axial Flux Difference Surveillance

Examinee (Print): ______________________________

Testing Method:

Simulated Performance: ______ Classroom: X

Actual Performance: X Simulator: ______

Alternate Path: ______ Plant: ______

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:

- Unit 1 is at 95% power.
- The Axial Flux Difference (AFD) alarm was declared INOPERABLE over 24 hours ago.
- Power Range Nuclear Instrument AFD data was collected for several hours last shift.

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

- PERFORM OPT-403, Axial Flux Difference.
- ENTER the Power Range Nuclear Instrument AFD data onto OPT-403-1, AFD Data Sheet.
- IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time, if any.
    - LCO ______________
    - REQUIRED ACTION ______________
    - COMPLETION TIME ______________
- RECORD findings in the Discrepancies/Comments Section of OPT-403-1.

TIME	1-NI-41C	1-NI-42C	1-NI-43C	1-NI-44C
0800	9%	10%	11%	9%
0830	9%	11%	12%	10%
0900	10%	11%	12%	11%
0930	11%	14%	12%	14%
1000	11%	13%	12%	13%
1030	12%	14%	13%	14%

Task Standard: Perform Axial Flux Difference surveillance per OPT-403 and record findings on OPT-403-1 and identify any Technical Specification Limiting Condition for Operation, Required Action and Completion Time.

Required Materials: OPT-403, Axial Flux Difference, Rev. 11.

OPT-403-1, AFD Data Sheet, Rev. 10-1.

NUC-204-6, Axial Flux Difference As a Function of Rated Thermal Power, Unit 1 Cycle 15, Rev. 07/26/10.

CPNPP Technical Specifications Units 1 and 2, Amendment 152.

Validation Time: 20 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **OPT-403, Axial Flux Difference.**
- **OPT-403-1, AFD Data Sheet.**
- **NUC-204-6, Axial Flux Difference As a Function of Rated Thermal Power.**
- **CPNPP Technical Specifications - Units 1 and 2.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OPT-403, Section 8.0 and documented on Form OPT-403-1.**
Perform Step: 1 8.1 & 8.1.1	Record the following data for the affected unit: • Unit 1 or 2 as applicable
Standard:	CIRCLED Unit 1 on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2 8.1 & 8.1.2	Record the following data for the affected unit: • Date
Standard:	ENTERED Date on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3 8.2 & 8.2.1	Record the following data: • Time
Standard:	ENTERED Time on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4√ 8.2 & 8.2.2	Record the following data: • PR Δ FLUX for each operable excore detector
Standard:	RECORDED PR Δ FLUX for each operable excore detector on OPT-403-1 from JPM Cue Sheet.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5√ 8.2 & 8.2.3	Record the following data: • Percent Rated Thermal Power (RTP)
Standard:	RECORDED Percent Rated Thermal Power on OPT-403-1.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 6√ 8.3 & 8.3.1	Perform the following to determine PR Δ FLUX status and record: • Verify at least 3 of 4 PR Δ FLUX channels are within the Acceptable Operation region (Doghouse Region) of NUC-204-6 "Axial Flux Difference as a Function of Rated Thermal Power."
Standard:	DETERMINED PR Δ FLUX status and RECORDED and INITIALED on OPT-403-1.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7√ 8.3 & 8.3.2	Perform the following to determine PR Δ FLUX status and record: • Repeat Steps 8.2 and 8.3 at least once per thirty (30) minutes.
Standard:	REPEATED Steps 8.2 and 8.3 at least once per thirty (30) minutes on OPT-403-1.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 8√	Identify Technical Specification Limiting Condition for Operation.
Standard:	DETERMINED Axial Flux Difference not within limits at 0930. • Technical Specification LCO 3.2.3, Axial Flux Difference, CONDITION A, AFD not within limits.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9√	Identify Technical Specification REQUIRED ACTION and COMPLETION TIME.
Standard:	DETERMINED Technical Specification REQUIRED ACTION and COMPLETION TIME: • 3.2.3.A.1 - Restore THERMAL POWER to < 50% RTP within 30 minutes.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 10	Record findings in the Discrepancies/Comments Section of OPT-403-1.
Standard:	RECORDED findings in the Discrepancies / Comments Section of OPT-403-1.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 is at 95% power.**
- **The Axial Flux Difference (AFD) alarm was declared INOPERABLE over 24 hours ago.**
- **Power Range Nuclear Instrument AFD data was collected for several hours last shift.**

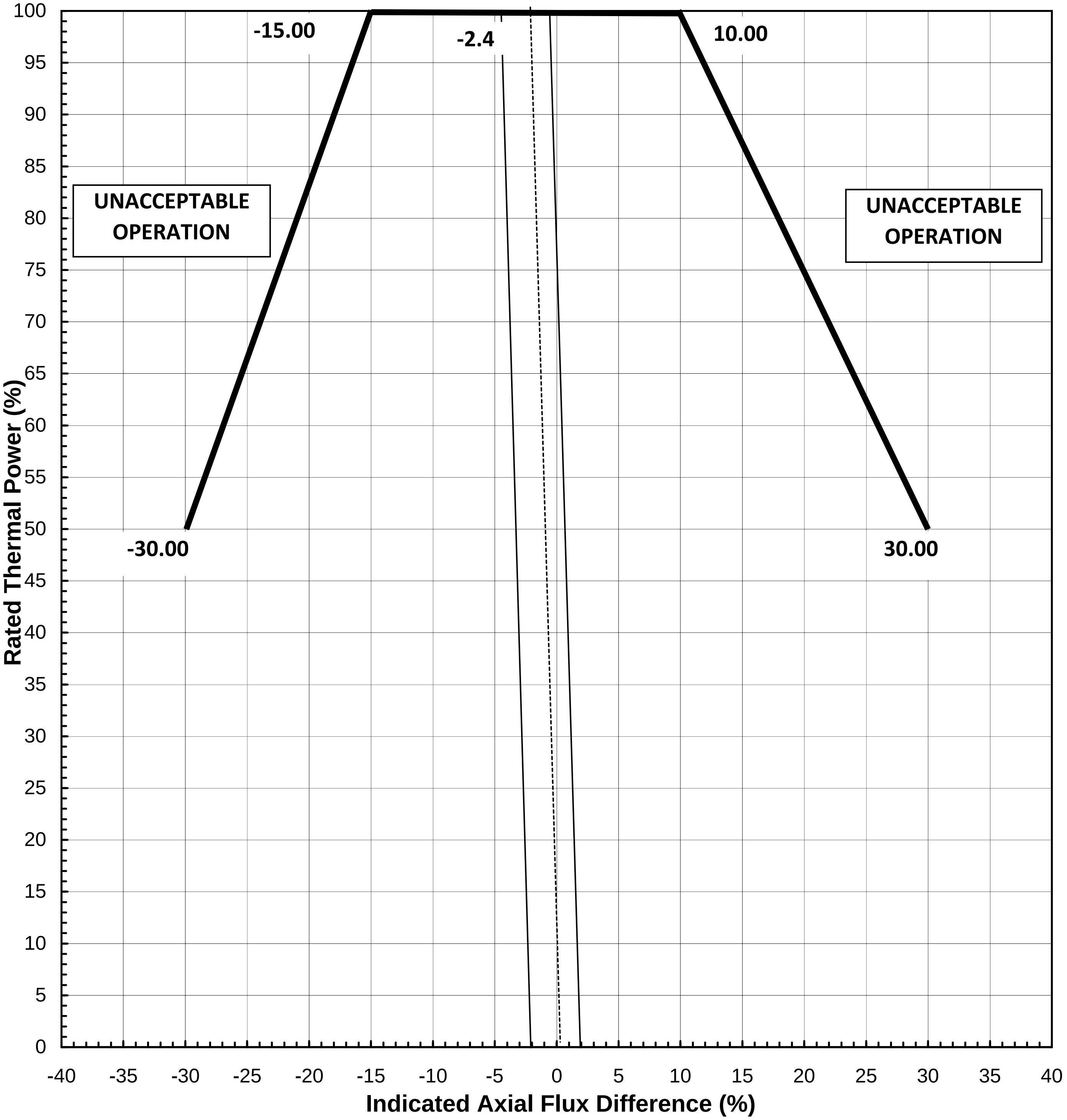
INITIATING CUE: **The Shift Manager directs you to PERFORM the following:**

- **PERFORM OPT-403, Axial Flux Difference.**
- **ENTER the Power Range Nuclear Instrument AFD data onto OPT-403-1, AFD Data Sheet.**
- **IDENTIFY any Technical Specification Limiting Condition for Operation (LCO), Required Action, and Completion Time, if any.**
    - **LCO** _______________
    - **REQUIRED ACTION** _______________
    - **COMPLETION TIME** _______________
- **RECORD findings in the Discrepancies/Comments Section of OPT-403-1.**

TIME	1-NI-41C	1-NI-42C	1-NI-43C	1-NI-44C
0800	9%	10%	11%	9%
0830	9%	11%	12%	10%
0900	10%	11%	12%	11%
0930	11%	14%	12%	14%
1000	11%	13%	12%	13%
1030	12%	14%	13%	14%

FOR SIMULATOR USE ONLY

Axial Flux Difference as a Function of Rated Thermal Power
Unit 1, Cycle 15



Prepared By: **MOL - IC18** Date: ____

Approved By: **SIMULATOR USE ONLY** Date: ____

Core Performance Engineering Supervisor

NUC-204-6
Page 1 of 1
Rev. 2

REFERENCE USE

COMANCHE PEAK NUCLEAR POWER PLANT

UNIT COMMON

OPERATIONS TESTING MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE____________________

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

AXIAL FLUX DIFFERENCE

PROCEDURE NO. OPT-403

REVISION NO. 11

EFFECTIVE DATE: 10/08/08 1200

PREPARED BY (Print): JIM BRAU Ext: 5443

TECHNICAL REVIEW BY (Print): ROB SLOUGH Ext: 5727

APPROVED BY: D.W. McGAUGHEY for Dave Goodwin Date: 09/19/08
DIRECTOR, OPERATIONS

1.0 PURPOSE

This procedure satisfies Axial Flux Difference (AFD) monitoring when automated monitoring is NOT available. The requirements of TRS 13.2.32.1 is met by monitoring and logging indicated AFD for each OPERABLE excore channel. A frequency of 30 minutes for logging data is used to ensure the requirement is met if the AFD Monitor is inoperable for greater than 24 hours.

The actual TRS frequency requirements are as follows:

NOTE: The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

- Once per hour for the first 24 hours that the AFD Monitor Alarm is inoperable. (TRS 13.2.32.1).
- Once per 30 minutes when the AFD Monitor Alarm is inoperable for >24 hours. (TRS 13.2.32.1).

2.0 ACCEPTANCE CRITERIA

2.1 At > 50% RTP the indicated AFD is within the Acceptable Operation limit of NUC-204-6 “Axial Flux Difference as a Function of Rated Thermal Power.”

3.0 DEFINITIONS/ACRONYMS

3.1 APL^{ND} - Analyzed Power Limit Nuclear Design

3.2 AFD - Axial Flux Difference

3.3 RTP - Rated Thermal Power

3.4 Validated Computer Program - A computer program verified consistent with current procedures and technical data used to calculate AFD and the associated penalty time.

4.0 REFERENCES

4.1 Technical Specification 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)”

4.2 Technical Requirement 13.2.32, "AXIAL FLUX DIFFERENCE (AFD)"

4.3 FSAR Section 4.3, “Nuclear Design”

4.4 COLR, “Core Operating Limits Report”

4.5 NUC-204, "Target Axial Flux Difference"

4.6 NUC-204-6, “Axial Flux Difference as a Function of Rated Thermal Power”

4.7 SOP-906, “Plant Process Computer System Guidelines”

5.0 PRECAUTIONS, LIMITATIONS AND NOTES

5.1 Precautions

None

5.2 Limitations

5.2.1 Core Performance Engineering shall be notified when any acceptance criteria not satisfied.

5.3 Notes

None

6.0 PREREQUISITES

☐ 6.1 MODE 1 at > 50% RTP.

7.0 TEST EQUIPMENT

None

CPNPP OPERATIONS TESTING MANUAL	UNIT COMMON	PROCEDURE NO. OPT-403
AXIAL FLUX DIFFERENCE	REVISION NO. 11	PAGE 4 OF 4

8.0 INSTRUCTIONS

NOTE: Record all data on Form OPT-403-1.

8.1 Record the following data for the affected unit:

☐ 8.1.1 Unit 1 or 2 as applicable

☐ 8.1.2 Date

8.2 Record the following data:

☐ 8.2.1 Time

☐ 8.2.2 PR Δ FLUX for each operable excore detector

☐ 8.2.3 Percent Rated Thermal Power (RTP)

8.3 Perform the following to determine PR Δ FLUX status and record:

☐ A. Verify at least 3 of 4 PR Δ FLUX channels are within the Acceptable Operation region (Doghouse Region) of NUC-204-6 “Axial Flux Difference as a Function of Rated Thermal Power.”

[C] ☐ B. Repeat Steps 8.2 and 8.3 at least once per thirty (30) minutes.

9.0 RESTORATION/POST WORK ACTIVITIES

None

10.0 ATTACHMENTS/FORMS

10.1 Attachments

None

10.2 Forms

10.2.1 OPT-403-1, AFD Data Sheet

AFD DATA SHEET

8.1.1 UNIT: (Circle one) [✓ 1] 2

8.1.2 DATE TODAY

NOTE:
- PR Δ FLUX shall be considered outside the Acceptable Operations limits when two or more OPERABLE excore channels indicate Δ FLUX to be outside the Acceptable Operations limits.
- PR Δ FLUX logging frequency may be shortened at the discretion of the Shift Manager.
- Log PR Δ FLUX data at least once per 30 minutes. The 30 minute frequency satisfies the following TRS 13.2.32.1 requirements:
  - Once per hour for the first 24 hours that the AFD Monitor Alarm is inoperable.
  - Once per 30 minutes when the AFD Monitor Alarm is inoperable for > 24 hours.

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
0800	9%	10%	11%	9%	95	[✓ YES] NO	LZ
0830	9%	11%	12%	10%	95	[✓ YES] NO	LZ
0900	10%	11%	12%	11%	95	[✓ YES] NO	LZ
0930	11%	14%	12%	14%	95	YES [✓ NO]	LZ
1000	11%	13%	12%	13%	95	YES [✓ NO]	LZ
1030	12%	14%	13%	14%	95	YES [✓ NO]	LZ
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

AFD DATA SHEET

TIME	PR ΔFLUX				% RTP	3 of 4 PR WITHIN ACCEPTABLE OPERATION	INITIAL
	u -NI-41C	u -NI-42C	u -NI-43C	u -NI-44C			
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	
						YES NO	

Additional copies of Page 2 of this data sheet may be used as required to document AFD status. ☐ Continued Next Page

OPT-403-1
PAGE 2 OF 3
R-10

AFD DATA SHEET

DISCREPANCIES / COMMENTS: Acceptance Criteria not met starting at 0930.

CORRECTIVE ACTIONS:

PERFORMED BY: ______________________ DATE: __________
SIGNATURE

REVIEWED BY: ______________________ DATE: __________
OPERATIONS MANAGEMENT

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

---NOTE---
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Restore THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days

Facility: CPNPP JPM # NRC SA4 Task #SO1112 K/A #2.3.12 3.2 / 3.7
Title: Determine Radiation Levels and Reporting Requirements

Examinee (Print): ________________________________

Testing Method:

Simulated Performance:	______	Classroom:	X
Actual Performance:	X	Simulator:	______
Alternate Path:	______	Plant:	______

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: JPM Cue Sheet #1

Given the following conditions:

- A high dose maintenance activity is scheduled in the Fuel Building.
- The general dose rate in the area is 100 mrem/hour but can be reduced to 25 mrem/hour if lead shielding is installed.
- It will take Nuclear Equipment Operators (NEOs) Alpha & Bravo 30 minutes to install the shielding.
- Independent of the shielding, it will take NEO Alpha two (2) hours or NEOs Alpha & Bravo one and a half (1.5) hours to perform the maintenance.

Initiating Cue:

The Work Control Supervisor directs you to PERFORM the following:

- CALCULATE the dose received when performing the maintenance for each of the following conditions:
  - NEO Alpha **without** shielding. __________ mrem.
  - NEOs Alpha & Bravo **without** shielding. __________ mrem.
  - NEO Alpha **with** shielding. __________ mrem.
  - NEOs Alpha & Bravo **with** shielding. __________ mrem.

Initial Conditions: JPM Cue Sheet #2

Given the following conditions:

- The Shift Manager was notified by Radiation Protection that an individual received 5.5 REM TEDE in a three (3) hour period.

Initiating Cue:

The Shift Manager directs you to PERFORM the following:

- DETERMINE Oral and Written Reportability Requirements, if any.
  - Oral Reporting Requirement _______________
  - Written Reporting Requirement _______________

Task Standard:

Calculate the dose received when performing the maintenance and determine oral and written Reporting Requirements for an overexposure event per STA-501.

Required Materials: STA-501, Nonroutine Reporting, Rev. 14-5.

Validation Time: 25 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee JPM Cue Sheet #1.

When JPM Cue Sheet #1 is completed, PROVIDE JPM Cue Sheet #2 and a copy of:

- **STA-501, Nonroutine Reporting.**

ENSURE examinee has a calculator.

√ - Check Mark Denotes Critical Step

START TIME:	

Perform Step: 1√	Determine total dose to NEO Alpha <u>without</u> shielding.
Standard:	DETERMINED total dose to NEO Alpha <u>without</u> shielding as follows: • 100 mrem/hr x 2 hours = **200 mrem total dose.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√	Determine total combined dose to NEOs Alpha & Bravo <u>without</u> shielding.
Standard:	DETERMINED total combined dose to NEOs Alpha & Bravo <u>without</u> shielding as follows: • 100 mrem/hr x 1.5 hours/NEO x 2 NEOs = **300 mrem total dose.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3	Determine total dose to <u>install</u> shielding.
Standard:	DETERMINED total dose to <u>install</u> shielding as follows: • 100 mrem/hr x 0.5 hours/NEO x 2 NEOs = **100 mrem to install.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4√	Determine total dose to NEO Alpha <u>with</u> shielding.
Standard:	DETERMINED total dose to NEO Alpha <u>with</u> shielding as follows: • 25 mrem/hr x 2 hours + 100 mrem = **150 mrem total dose.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5√	Determine total combined dose to NEOs Alpha & Bravo <u>with</u> shielding.
Standard:	DETERMINED total combined dose to NEOs Alpha & Bravo <u>with</u> shielding as follows: • 25 mrem/hr x 1.5 hours/NEO x 2 NEOs + 100 mrem = **175 mrem total dose.**
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Provide the examinee with copy of JPM Cue Sheet #2 and STA-501.**
Examiner Note:	**The following steps are from STA-501, Attachment 8.D/4.**
Perform Step: 6√ Attachment 8.D/4 Page 2 of 11 <u>or</u> 2 of 11	Determine oral Reporting Requirements per STA-501.
Standard:	DETERMINED oral Reporting Requirements per STA-501: "An individual to receive, in a period of 24 hours: TEDE ≥ 5 rems." • Oral Report within 24 hours via the ENS per 10CFR20.2202 (b).
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7√ Attachment 8.D/4 Page 2 of 11 <u>or</u> 7 of 11	Determine written Reporting Requirements per STA-501.
Standard:	DETERMINED written Reporting Requirement per STA-501: "Any incident for which notification is required per 10CFR20.2202." • Written Report (LER) within 30 days per 10CFR20.2203 (a) (1) and (2).
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **A high dose maintenance activity is scheduled in the Fuel Building.**
- **The general dose rate in the area is 100 mrem/hour but can be reduced to 25 mrem/hour if lead shielding is installed.**
- **It will take Nuclear Equipment Operators (NEOs) Alpha & Bravo 30 minutes to install the shielding.**
- **Independent of the shielding, it will take NEO Alpha two (2) hours or NEOs Alpha & Bravo one and a half (1.5) hours to perform the maintenance.**

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

- **CALCULATE the dose received when performing the maintenance for each of the following conditions:**
  - **NEO Alpha without shielding. _____ mrem.**
  - **NEOs Alpha & Bravo without shielding. _____ mrem.**
  - **NEO Alpha with shielding. _____ mrem.**
  - **NEOs Alpha & Bravo with shielding. _____ mrem.**

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

- **The Shift Manager was notified by Radiation Protection that an individual received 5.5 REM TEDE in a three (3) hour period.**

<u>INITIATING</u> <u>CUE</u>: **The Shift Manager directs you to PERFORM the following:**

- **DETERMINE Oral and Written Reportability Requirements, if any.**
  - **Oral Reporting Requirement** _______________
  - **Written Reporting Requirement** _______________

ATTACHMENT 8.D/4
PAGE 2 OF 11

From 10CFR20.2202(b)

Each Licensee shall within 24 hours of discovery of the event, report any event involving licensed material possessed by the Licensee that may have caused or threatens to cause:

1) An individual to receive, in a period of 24 hours:
   a) TEDE ≥ 5 rems
   b) DE ≥ 15 rems (eye)
   c) SE ≥ 50 rems (skin or extremity)

2) The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, could have received an intake in excess of one occupational annual limit on intake from 10CFR20, appendix B, Table 1, Column 2.

From 10CFR20.2203(a)(1)and(2)

Report in writing within 30 days of occurrence the following types of incidents:

1. Any incident for which notification is required by 10CFR20.2202.

2. Doses in excess of any of the following:

   (i) The occupational dose limits for adults in 10CFR20.1201; or
   (ii) The occupational dose limits for a minor in 10CFR20.1207; or
   (iii) The limits for an embryo/fetus of a declared radiation worker in 10CFR20.1208; or

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-501
NONROUTINE REPORTING	REVISION NO. 14	PAGE 68 OF 168

ATTACHMENT 8.D/4
PAGE 7 OF 11

TABLE NR-4a:SUMMARY OF RADIOLOGICAL EXPOSURE REPORTING REQUIREMENTS

CONDITION	EXPOSURE	SOURCE	TYPE OF REPORT
Occupational Dose Limits for Minors	The annual occupational dose limit for minors is 10% of the annual dose limit specified for adult workers in 10CFR20.1201	Requirement: 10CFR20.1207 Report: 10CFR20.2203(a)(2)(ii)	30 day LER (OL)
Individual exposed to licensed material within a restricted area	(1) Annual Limit, whichever is more limiting: 5 rem TEDE or 50 rem CDE (organ or tissue)	Requirement: 10CFR20.1201(a) Report: 10CFR20.2203(a)(2) (i)	30 day LER (OL)
	(2) Annual Limit 15 rem (eye) 50 rem (skin or any one extremity		
	(3) Individual's accumulated occupational dose is documented with licensee		
Any event involving licensed material possessed that may have caused or threatens to cause exposure to individual	≥ 5 rem TEDE ≥15 rem (eye) ≥50 rem (skin or any one extremity)	Requirement: 10CFR20.2202 Report: 10CFR20.2203	24 hour notification via ENS, AND 30 day LER (OL)
Event involving byproduct, source, or special nuclear material that may have caused or threatens to cause exposure to individual	≥ 25 rem TEDE ≥ 75 rem (eye) ≥250 rad (skin or any one extremity)	Requirement: 10CFR20.2202 Report: 10CFR20.2203	1 hour notification via ENS, AND 30 day LER (OL)
Limits for members of the public	Annual Limit: 100 mrem TEDE Unrestricted Area Dose: 2 mrem in any one hour	Requirement: 10CFR20.1301 Report: 10CFR20.2203	30 day LER (OL)
Limits for Embryo/Fetus of a declared pregnant radiation worker	Gestation Period Limit: 500 mrem TEDE (with a uniform monthly exposure rate)	Requirement: 10CFR20.1208 Report: 10CFR20.2203	30 day LER (OL)

Facility: CPNPP JPM # NRC SA5 Task #SO1136 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: ______

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 Loss of All Offsite Power occurred on Unit 1 30 minutes ago.
- Both Safeguards Buses are deenergized.
- Train A Emergency Diesel Generator was shut down following turbocharger failure.
- Train B Emergency Diesel Generator will not start.
- Pressurizer level is 0%.
- Reactor Coolant System pressure is 30 PSIG and stable.
- Core Exit Thermocouple temperatures are 780ºF and rising.
- Containment pressure is 30 PSIG and stable.
- Steam Generator wide range levels are 30% and slowly lowering.
- No Reactor Vessel Level Indication System lights are lit.
- Turbine Driven Auxiliary Feedwater Pump has tripped.

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

- DETERMINE the Emergency Action Level Group / Category, Subcategory, and Event Classification per EPP-201, Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation.

Task Standard: Determine the Event Category and Event Classification using the Emergency Action Level Hot & Cold Classification Charts per EPP-201.

Required Materials: EPP-201, Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation, Rev. 12.

CPNPP Emergency Action Level Hot & Cold Classification Charts, Rev. 0.

EOP Critical Safety Function Status Trees, Rev. 8.

Validation Time: 20 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: _________________

CLASSROOM SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **EPP-201, Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation.**
- **CPNPP Emergency Action Level Hot & Cold Classification Charts.**
- **EOP Critical Safety Function Status Trees.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from CPNPP Emergency Action Levels Hot.**
Perform Step: 1√	DETERMINE the Event Category.
Standard:	REFERRED to CPNPP Emergency Action Levels Hot and Cold and DETERMINED the following chart is applicable: • **CPNPP EAL Hot Conditions**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√	MATCH plant conditions in the EAL Group / Category.
Standard:	IDENTIFIED EAL Group / Category as **System Malfunctions (S)**.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√	MATCH plant conditions in the selected EAL Subcategory.
Standard:	IDENTIFIED EAL Subcategory as **Loss of AC Power (1)**.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√	Classify the event.
Standard:	CLASSIFIED the event as a **GENERAL EMERGENCY (SG1.1).**
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **A Loss of All Offsite Power occurred on Unit 1 30 minutes ago.**
- **Both Safeguards Buses are deenergized.**
- **Train A Emergency Diesel Generator was shut down following turbocharger failure.**
- **Train B Emergency Diesel Generator will not start.**
- **Pressurizer level is 0%.**
- **Reactor Coolant System pressure is 30 PSIG and stable.**
- **Core Exit Thermocouple temperatures are 780ºF and rising.**
- **Containment pressure is 30 PSIG and stable.**
- **Steam Generator wide range levels are 30% and slowly lowering.**
- **No Reactor Vessel Level Indication System lights are lit.**
- **Turbine Driven Auxiliary Feedwater Pump has tripped.**

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

- **DETERMINE the Emergency Action Level Group / Category, Subcategory, and Event Classification per EPP-201, Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation.**

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

Comanche Peak Nuclear Power Plant
Emergency Action Level Matrix

Revision 12
Effective Date 11/04/2010

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
R **Abnorm. Rad Release / Rad Effluent**	**1** **Offsite Rad Conditions**	Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology 1 2 3 4 5 6 DEF **RG1.1 (Bases Page 66)** Valid reading on any radiation monitors > Table R-1 column "GE" for ≥15 min. (Note 1) **RG1.2 (Bases Page 69)** Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the Exclusion Area Boundary **RG1.3 (Bases Page 71)** Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥60 min. at or beyond the Exclusion Area Boundary (Note 1) **OR** Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the Exclusion Area Boundary	Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release 1 2 3 4 5 6 DEF **RS1.1 (Bases Page 60)** Valid reading on any radiation monitors > Table R-1 column "SAE" for ≥15 min. (Note 1) **RS1.2 (Bases Page 63)** Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the Exclusion Area Boundary **RS1.3** Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥60 min. at or beyond the Exclusion Area Boundary (Note 1) **OR (Bases Page 64)** Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the Exclusion Area Boundary	**Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer 1 2 3 4 5 6 DEF **RA1.1 (Bases Page 51)** Valid reading on any Gaseous monitors > Table R-1 column "Alert" for ≥15 min. (Note 2) **RA1.2 (Bases Page 55)** Valid reading on any Liquid monitors > Table R-1 column "Alert" for ≥15 min. (Note 2) **RA1.3 (Bases Page 58)** Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥15 min.	**Any** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer 1 2 3 4 5 6 DEF **RU1.1 (Bases Page 41)** Valid reading on any Gaseous monitors > Table R-1 column "UE" for ≥60 min. (Note 2) **RU1.2 (Bases Page 45)** Valid reading on any Liquid monitors > Table R-1 column "UE" for ≥60 min. (Note 2) **RU1.3 (Bases Page 49)** Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥60 min.
	2 **Onsite Rad Conditions & Spent Fuel Events**	(Table R-1)	(Table R-1)	Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel 1 2 3 4 5 6 DEF **RA2.1 (Bases Page 77)** Damage to irradiated fuel OR loss of water level **AND** A valid High alarm on any of the following: • RFCu10, LRAM W REFUEL CAV860 (u-RE-6251) • RFCu12, LRAM E REFUEL CAV 860 (u-RE-6253) • CAPu98, CNTMT AIR PIG PART (u-RE-5502) • CAIu99, CNTMT AIR PIG IODINE (u-RE-5566) • CAGu97, CNTMT AIR PIG GAS (u-RE-5503) • FBV088, FB VENT EXH (X-RE-5700) • SFP001, LRAM SFP 2 E WALL (X-RE-6272) • SFP002, LRAM SFP 2 N WALL (X-RE-6273) • SFP003, LRAM SFP 1 E WALL (X-RE-6274) • SFP004, LRAM SFP 1 S WALL (X-RE-6275) **RA2.2 (Bases Page 79)** A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered	Unplanned rise in plant radiation levels 1 2 3 4 5 6 DEF **RU2.1 (Bases Page 73)** Unplanned water level drop in a reactor refueling pathway as indicated by EITHER of the following: • < 857' 3¼" in spent fuel pool or fuel transfer canal • < 856' 11" in refueling cavity (407" above core plate) **AND** Valid area radiation monitor reading rise on any of the following: • RFCu10, LRAM W REFUEL CAV860 (u-RE-6251) • RFCu12, LRAM E REFUEL CAV 860 (u-RE-6253) • SFP001, LRAM SFP 2 E WALL (X-RE-6272) • SFP002, LRAM SFP 2 N WALL (X-RE-6273) • SFP003, LRAM SFP 1 E WALL (X-RE-6274) • SFP004, LRAM SFP 1 S WALL (X-RE-6275) **RU2.2 (Bases Page 76)** Unplanned valid area radiation monitor reading rises by a factor of 1000 over normal levels* * Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value
	3 **CR/CAS Rad**	None	None	Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions 1 2 3 4 5 6 DEF **RA3.1 (Bases Page 81)** Dose rates > 15 mRem/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (X-RE-6281 or X-RE-6282) **OR** CAS (by survey)	None
H **Hazards**	**1** **Natural or Destructive Phenomena**	None	(Table H-1)	Natural or destructive phenomena affecting Vital Areas 1 2 3 4 5 6 DEF **HA1.1 (Bases Page 154)** Seismic event > OBE as indicated by annunciator 2A-3.1, OBE EXCEEDED, or yellow OBE light on Seismic Monitoring system panel **AND** Earthquake confirmed by **any** of the following: • Earthquake felt in plant • National Earthquake Information Center (Note 8) • Control Room indication of degraded performance of systems required for the safe shutdown of the plant **HA1.2 (Bases Page 157)** Tornado striking or sustained high winds > 80 mph resulting in **EITHER**: • Visible damage to **any** Table H-1 structures • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown **HA1.3 (Bases Page 160)** Internal flooding in the Safeguards Building or Turbine Building resulting in **EITHER**: • An electrical shock hazard that precludes access to operate or monitor systems required to establish or maintain safe shutdown • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown **HA1.4 (Bases Page 162)** Turbine failure-generated projectiles resulting in **EITHER**: • Visible damage to or penetration of **any** Table H-1 structures • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown **HA1.5 (Bases Page 165)** Safe Shutdown Impoundment level > 796.0 ft (lake) **OR** Safe Shutdown Impoundment level < 761.5 ft (inside traveling screens) **HA1.6 (Bases Page 167)** Vehicle crash resulting in **EITHER**: • Visible damage to **any** Table H-1 structures • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown	Natural or destructive phenomena affecting the Protected Area 1 2 3 4 5 6 DEF **HU1.1 (Bases Page 143)** Seismic event identified by **any two** of the following: • Annunciator 2A- 2.1, SEISMIC MONITORING SYSTEM ACTIVATION, received • Earthquake felt in plant • National Earthquake Information Center (Note 8) **HU1.2 (Bases Page 145)** Tornado striking within the Protected Area boundary **OR** Sustained high winds > 80 mph **HU1.3 (Bases Page 147)** Internal flooding that has the potential to affect safety-related equipment required by Technical Specifications for the current operating mode in the Safeguards Building or Turbine Building **HU1.4 (Bases Page 149)** Turbine failure resulting in casing penetration or damage to turbine or generator seals **HU1.5 (Bases Page 151)** Safe Shutdown Impoundment level > 794.7 ft (lake) **OR** Safe Shutdown Impoundment level < 769.5 ft (inside traveling screens)
	2 **Fire or Explosion**	(Note 9)	None	Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown 1 2 3 4 5 6 DEF **HA2.1 (Bases Page 173)** Fire or explosion resulting in **EITHER**: • Visible damage to any Table H-1 structures • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown (Note 9)	Fire within the Protected Area not extinguished within 15 min. of detection or explosion within the Protected Area 1 2 3 4 5 6 DEF **HU2.1 (Bases Page 169)** Fire **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **any** Table H-1 area (Note 4) **HU2.2 (Bases Page 171)** Explosion of sufficient force to damage permanent structures or equipment within the Protected Area (Note 9)
	3 **Hazardous Gas**	None	(Note 3)	Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor 1 2 3 4 5 6 DEF **HA3.1 (Bases Page 178)** Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 3)	Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations 1 2 3 4 5 6 DEF **HU3.1 (Bases Page 175)** Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations **HU3.2 (Bases Page 177)** Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event
	4 **Security**	Hostile action resulting in loss of physical control of the facility 1 2 3 4 5 6 DEF **HG4.1 (Bases Page 186)** A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain **any** of the following safety functions: • Reactivity control • RCS inventory • Secondary heat removal **HG4.2 (Bases Page 187)** A hostile action has caused failure of Spent Fuel Cooling systems **AND** Imminent fuel damage is likely for a freshly off-loaded reactor core in pool	Hostile action within the Protected Area 1 2 3 4 5 6 DEF **HS4.1 (Bases Page 184)** A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor	Hostile action within the Owner Controlled Area or airborne attack threat 1 2 3 4 5 6 DEF **HA4.1 (Bases Page 182)** A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor **OR** A validated notification from NRC of an airliner attack threat within 30 min. of the site	Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant 1 2 3 4 5 6 DEF **HU4.1 (Bases Page 180)** A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor **OR** A credible site-specific security threat notification **OR** A validated notification from NRC providing information of an aircraft threat
	5 **Control Room Evacuation**	None	Control Room evacuation has been initiated and plant control **cannot** be established 1 2 3 4 5 6 DEF **HS5.1 (Bases Page 189)** Control Room evacuation has been initiated **AND** Control of the plant **cannot** be established within 15 min.	Control Room evacuation has been initiated 1 2 3 4 5 6 DEF **HA5.1 (Bases Page 188)** Control Room evacuation has been initiated	None
	6 **Judgment**	Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of General Emergency 1 2 3 4 5 6 DEF **HG6.1 (Bases Page 197)** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) offsite for more than the immediate site area	Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of Site Area Emergency 1 2 3 4 5 6 DEF **HS6.1 (Bases Page 195)** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. **Any** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the Exclusion Area Boundary	Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of an Alert 1 2 3 4 5 6 DEF **HA6.1 (Bases Page 193)** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE).	Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE 1 2 3 4 5 6 DEF **HU6.1 (Bases Page 191)** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs
E **ISFSI**		None	None	None	Damage to a loaded cask Confinement Boundary 1 2 3 4 5 6 DEF **EU1.1 (Bases Page 248)** Damage to a loaded cask Confinement Boundary

Note 1: The Emergency Coordinator should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time
If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2). Do **not** delay declaration awaiting dose assessment

Note 2: The Emergency Coordinator should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent** PVG384 + PVG385	X-RE-5567 A + B	---------	---------	1.3E-1 µCi/cc	1.3E-3 µCi/cc
	Plant Vent (WRGM) PVF684 + PVF685	X-RE-5570 A + B	1.7E+7 µCi/sec	1.7E+6 µCi/sec	8.0E+5µCi/sec	8.0E+4µCi/sec
	Main Steam MSLu78 MSLu79 MSLu80 MSLu81	u-RE-2325 u-RE-2326 u-RE-2327 u-RE-2328	27 µCi/cc	2.7 µCi/cc	10 x high alarm setpoint	2 x high alarm setpoint
Liquid	**Liquid Waste** LWE-076	X-RE-5253	-------	-------	200 X high alarm setpoint*	2 X high alarm setpoint*
	Service Water SSWu65 SSWu66	u-RE-4269 u-RE-4270	-------	-------	200 X high alarm setpoint	2 X high alarm setpoint

* With effluent discharge **not** isolated

Note 8: Web address for National Earthquake Information Center is:
www.earthquake.usgs.gov/earthquakes/recenteqsww/Quakes/quakes_all.php

Table H-1 Structures Containing Systems Needed for Safe Shutdown

- u-Containment
- u-Safeguards Building
- X-Auxiliary Building
- X-Electrical & Control Building
- X-Fuel Building
- X-Service Water Intake Structure
- u-Diesel Generator Building
- u-Normal switchgear rooms
- u-CST
- u-RWST

Note 9: Explosion is defined as a rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL.

Note 3: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

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HOT CONDITIONS (RCS > 200°F)

Revision 12
Effective Date 11/04/2010

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S System Malfunct.	**1** Loss of AC Power	Prolonged loss of **all** offsite and **all** onsite AC power to safeguard buses 1 2 3 4 **SG1.1** (Bases Page 211) Loss of **all** offsite and **all** onsite AC power to 6.9 KV safeguard buses uEA1 and uEA2 **AND EITHER**: • Restoration of at least one safeguard bus within 4 hours is **not** likely • CSFST Core Cooling - RED or ORANGE path	Loss of **all** offsite and **all** onsite AC power to safeguard buses for ≥15 min. 1 2 3 4 **SS1.1** (Bases Page 208) Loss of **all** offsite and **all** onsite AC power to 6.9 KV safeguard buses uEA1 and uEA2 for ≥15 min. (Note 4)	AC power capability to safeguard buses reduced to a single power source for ≥15 min. such that **any** additional single failure would result in a loss of all AC power to safeguard buses 1 2 3 4 **SA1.1** (Bases Page 205) AC power capability to 6.9 KV safeguard buses uEA1 and uEA2 reduced to a single power source for ≥15 min. (Note 4) **AND** **Any** additional single power source failure will result in loss of **all** AC power to 6.9 KV safeguard buses uEA1 and uEA2 (Table S-3)	Loss of **all** offsite AC power to safeguard buses for ≥15 min. 1 2 3 4 **SU1.1** (Bases Page 202) Loss of **all** offsite AC power to 6.9 KV safeguard buses uEA1 and uEA2 for ≥15 min. (Note 4)
	2 Loss of DC Power	None	Loss of **all** vital DC power for ≥15 min. 1 2 3 4 **SS2.1** (Bases Page 216) < 105 VDC on **all** 125 VDC safeguard buses uED1, uED2, uED3 and uED4 for ≥15 min. (Note 4)	None	None
	3 Criticality & RPS Failure	Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists 1 2 **SG3.1** (Bases Page 226) An automatic trip failed to shut down the reactor **AND** **All** manual actions do **not** shut down the reactor as indicated by reactor power ≥5% **AND EITHER**: • CSFST Core Cooling - RED • CSFST Heat Sink - RED	Automatic trip fails to shut down the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor 1 2 **SS3.1** (Bases Page 224) An automatic trip failed to shut down the reactor **AND** Manual actions taken at the reactor control console (Note 6) do **not** shut down the reactor as indicated by reactor power ≥5%	Automatic trip fails to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor 1 **SA3.1** (Bases Page 220) An automatic trip failed to shut down the reactor **AND** Manual actions taken at the reactor control console (Note 6) successfully shut down the reactor as indicated by reactor power < 5%	Inadvertent criticality 3 4 **SU3.1** (Bases Page 219) An unplanned sustained positive startup rate observed on nuclear instrumentation
	4 Inability to Reach or Maintain Shutdown Conditions	None	None	None	Inability to reach required shutdown within Technical Specification limits 1 2 3 4 **SU4.1** (Bases Page 230) Plant is not brought to required operating mode within Technical Specifications LCO action statement time
	5 Instr.	None	Inability to monitor a significant transient in progress 1 2 3 4 **SS5.1** (Bases Page 235) Loss of approximately 75% (or more) of annunciation or indication on CB-01 through CB-09 and CB-11 for ≥15 min. (Note 4) **AND** A significant transient is in progress, Table S-1 **AND** Compensatory indications are unavailable	Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable 1 2 3 4 **SA5.1** (Bases Page 233) Unplanned loss of approximately 75% (or more) of annunciation or indication on CB-01 through CB-09 and CB-11 for ≥15 min. (Note 4) **AND EITHER:** • A significant transient is in progress, Table S-1 • Compensatory indications are unavailable	Unplanned loss of safety system annunciation or indication in the Control Room for ≥15 min. 1 2 3 4 **SU5.1** (Bases Page 231) Unplanned loss of approximately 75% (or more) of annunciation or indication associated with safety systems on CB-01 through CB-09 and CB-11 for ≥15 min. (Note 4)
	6 Comm.	None	None	None	Loss of **all** onsite or offsite communications capabilities 1 2 3 4 **SU6.1** (Bases Page 238) Loss of **all** Table S-2 onsite (internal) communication methods affecting the ability to perform routine operations **OR** Loss of **all** Table S-2 offsite (external) communication methods affecting the ability to perform offsite notifications
	7 Fuel Clad Degradation	None	None	None	Fuel clad degradation 1 2 3 4 **SU7.1** (Bases Page 240) Reactor coolant Dose Equivalent I-131 specific activity > 60 µCi/gm **OR** Reactor coolant Dose Equivalent XE-133 specific activity > 500 µCi/gm **SU7.2** (Bases Page 242) Gross Failed Fuel Monitor, FFLu60 (u-RE-0406), High Alarm (RED)
	8 RCS Leakage	None	None	None	RCS leakage 1 2 3 4 **SU8.1** (Bases Page 244) Unidentified or pressure boundary leakage > 10 gpm (Note 7) **OR** Identified leakage > 25 gpm
F Fission Product Barriers		1 2 3 4 **FG1.1** (Bases Page 256) Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	1 2 3 4 **FS1.1** (Bases Page 254) Loss or potential loss of **any** two barriers (Table F-1)	1 2 3 4 **FA1.1** (Bases Page 253) **Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)	1 2 3 4 **FU1.1** (Bases Page 252) **Any** loss or **any** potential loss of Containment (Table F-1)

Table F-1 Fission Product Barrier Matrix

	Fuel Cladding Barrier		Reactor Coolant System Barrier		Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. CSFST	1. CSFST Core Cooling-RED entry conditions met (Bases Page 260)	1. CSFST Core Cooling-ORANGE entry conditions met **OR** CSFST Heat Sink-RED entry conditions met and heat sink required (Bases Page 263)	None	1. CSFST RCS Integrity - RED entry conditions met **OR** CSFST Heat Sink - RED entry conditions met and heat sink required (Bases Page 280)	None	1. CSFST Containment - RED entry conditions met (Bases Page 298)
B. Core Exit T/Cs	2. Core exit TCs > 1,200°F (Bases Page 266)	2. Core exit TCs > 750°F (Bases Page 267)	None	None	None	2. Core exit TCs > 1,200°F **AND** Restoration procedures **not** effective within 15 min. (Bases Page 301) 3. **All** of the following: • Core exit TCs > 750°F • RVLIS 11 in. above plate light **not** lit • Restoration procedures **not** effective within 15 min. (Bases Page 303)
C. Radiation	3. Containment radiation > 400 R/hr CTEu16 Containment HRRM (u-RE-6290A), or CTWu17 Containment HRRM (u-RE-6290B) (Bases Page 268) 4. Gross Failed Fuel Monitor, (FFLu60) u-RE-0406, radiation > 3.7E04 µCi/cc (Bases Page 270)	None	1. Containment radiation > 5 R/hr CTEu16 Containment HRRM (u-RE-6290A), or CTWu17 Containment HRRM (u-RE-6290B) (Bases Page 285)	None	None	4. Containment radiation > 4,000 R/hr CTEu16 Containment HRRM (u-RE-6290A), or CTWu17 Containment HRRM (u-RE-6290B) (Bases Page 306)
D. Inventory	None	3. RVLIS 11 in. above plate light **not** lit (Bases Page 274)	2. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (≤ 25°F [55°F]) (Bases Page 288) 3. Ruptured SG results in an ECCS (SI) actuation (Bases Page 290)	2. RCS leak rate > the capacity of one charging pump in the normal charging mode with letdown isolated: • Positive Displacement: 98 gpm • Centrifugal: 150 gpm (Bases Page 291)	1. Containment pressure rise followed by a rapid unexplained drop in Containment pressure (Bases Page 308) 2. Containment pressure or sump level response **not** consistent with LOCA conditions (Bases Page 309) 3. Ruptured SG is also faulted outside of Containment (Bases Page 311) 4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable steam release from affected SG to the environment (Bases Page 313)	5. Containment pressure 50 psig and rising (Bases Page 315) 6. Containment hydrogen concentration > 4% (Bases Page 316) 7. Containment pressure > 18 psig with **neither** Containment Spray system train operating (Bases Page 318)
E. Other	5. Coolant activity > 300 µCi/cc I-131 Dose Equivalent (Bases Page 275)	None	None	None	5. Failure of **all** valves in **any one** line to close **AND** Direct downstream pathway to the environment exists after Containment isolation signal (Bases Page 320)	None
F. Judgment	6. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier (Bases Page 277)	4. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier (Bases Page 278)	4. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier (Bases Page 295)	3. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier (Bases Page 296)	6. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier (Bases Page 322)	8. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier (Bases Page 323)

Note 4: The Emergency Coordinator should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Note 5: Applicable on Cold Condition Chart only.

Note 6: For manual trip, the MCB reactor trip switches and deenergizing uB3 and uB4 are the only methods applicable to EALs SA3.1 and SS3.1

Note 7: Use Category F EALs for escalation due to RCS leakage

Table S-1 Significant Transients

- Electrical load rejection > 25% full electrical load
- Reactor trip
- Runback > 25% reactor power
- ECCS injection
- Reactor power oscillations > 10%

Table S-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Gai-Tronics Page/party system (Public Address System)	X	
Plant Radio System	X	
PABX (Private Automatic Branch Exchange System)	X	X
Public Telephone System	X	X
Federal Telephone System (FTS)		X

Table S-3 AC Power Sources

Offsite:
- 138 KV switchyard circuit
- 345 KV switchyard circuit

Onsite:
- uEG1
- uEG2

EAL Identifier

XXX.X

Category (R, H, E, S, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

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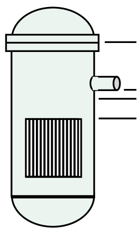
COLD CONDITIONS (RCS ≤ 200°F)

Revision 12
Effective Date 11/04/2010

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
C **Cold SD/ Refueling System Malfunct.**	**1** **Loss of AC Power**	None	None	Loss of **all** offsite and **all** onsite AC power to safeguard buses 5 6 DEF **CA1.1 (Bases Page 88)** Loss of **all** offsite and **all** onsite AC power to 6.9 KV safeguard buses uEA1 and uEA2 for ≥15 min. (Note 4)	AC power capability to safeguard buses reduced to a single power source for ≥15 min. such that **any** additional single failure would result in a loss of all AC power to safeguard buses 5 6 **CU1.1 (Bases Page 85)** AC power capability to 6.9 KV safeguard buses uEA1 and uEA2 reduced to a single power source for ≥15 min. (Note 4) **AND** **Any** additional single power source failure will result in loss of **all** AC power to 6.9 KV safeguard buses uEA1 and uEA2 (Table C-5)
	2 **Loss of DC Power**	None	None	None	Loss of required DC power for ≥15 min. 5 6 **CU2.1 (Bases Page 91)** < 105 VDC on **required** 125 VDC safeguard buses (uED1, uED2, uED3, uED4) for ≥15 min. (Note 4)
	3 **RCS Level**	Loss of RCS inventory affecting fuel clad integrity with Containment challenged 5 6 **CG3.1 (Bases Page 118)** RCS level < 0 in. above upper core plate (top) for ≥30 min. (Note 4) **AND** **Any** Containment challenge condition, Table C-4 **CG3.2 (Bases Page 122)** RCS level **cannot** be monitored for ≥30 min. (Note 4) with a loss of RCS inventory indicated by **any** of the following: • > 20,000 R/hr on **any** of the following: - CTEu16, Containment HRRM (u-RE-6290A) - CTWu17, Containment HRRM (u-RE-6290B) • Erratic source range monitor indication • Unexplained level rise in **any** Table C-1 sump / tank level **AND** Any Containment challenge condition, Table C-4	Loss of RCS inventory affecting core decay heat removal capability 5 6 **CS3.1 (Bases Page 108)** With Containment closure **not** established, RCS level < 27.25 in. above upper core plate (top) **CS3.2 (Bases Page 111)** With Containment closure established, RCS level < 0 in. above upper core plate (top) **CS3.3 (Bases Page 114)** RCS level **cannot** be monitored for ≥30 min. (Note 4) with a loss of RCS inventory indicated by **any** of the following: • > 20,000 R/hr on **any** of the following: - CTEu16, Containment HRRM (u-RE-6290A) - CTWu17, Containment HRRM (u-RE-6290B) • Erratic source range monitor indication • Unexplained level rise in **any** Table C-1 sump / tank level	Loss of RCS inventory 5 6 **CA3.1 (Bases Page 104)** RCS level < 33.25 in. above upper core plate (top) **OR** RCS level **cannot** be monitored for ≥15 min. (Note 4) with a loss of RCS inventory as indicated by an unexplained level rise in **any** Table C-1 sump / tank level	RCS leakage 5 **CU3.1 (Bases Page 94)** RCS leakage results in the inability to maintain or restore **EITHER** of the following for ≥15 min. (Note 4): • Pressurizer level > 17% • Above the low end of the target level control band (If pressurizer level was intentionally lowered < 17%) RCS leakage 6 **CU3.2 (Bases Page 96)** Unplanned RCS level drop below **EITHER** of the following for ≥15 min. (Note 4): • Reactor Vessel flange (when the level band is established **above** the flange) • Target band (when the level band is established **below** the flange) **CU3.3 (Bases Page 100)** RCS level **cannot** be monitored **AND** Loss of RCS inventory as indicated by an unexplained level rise in **any** Table C-1 sump / tank level
	4 **RCS Temp.**	None	None	Inability to maintain plant in cold shutdown 5 6 **CA4.1 (Bases Page 134)** An unplanned event results in **EITHER**: • RCS temperature > 200°F (Note 5) for > Table C-3 duration • RCS pressure rise > 10 psig due to a loss of RCS cooling (this condition is not applicable in solid plant conditions)	Unplanned loss of decay heat removal capability with irradiated fuel in the Reactor Vessel 5 6 **CU4.1 (Bases Page 129)** Unplanned event results in RCS temperature > 200°F (Note 5) **CU4.2 (Bases Page 131)** Loss of **all** RCS temperature and RCS level indication for ≥15 min. (Note 4)
	5 **Comm.**	None	None	None	Loss of **all** onsite or offsite communications capabilities 5 6 DEF **CU5.1 (Bases Page 138)** Loss of **all** Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations **OR** Loss of **all** Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications
	6 **Inadvertent Criticality**	None	None	None	Inadvertent criticality 5 6 **CU6.1 (Bases Page 140)** An unplanned sustained positive startup rate observed on nuclear instrumentation

Reactor Vessel Threshold Values



	EWR	Plant El.	EAL(s)
Reactor Vessel Flange	132.50“	834‘ 0.50“	CU3.2
Bottom of Hotleg	33.25“	825‘ 9.25“	CA3.1
6 in. < Bottom of Hotleg	27.25“	825‘ 3.25“	CS3.1
Top of Core Plate	0.00“	823‘ 0.00“	CS3.2 CG3.1

Note 4: The Emergency Coordinator should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Note 5: Begin monitoring hot condition EALs concurrently

Table C-1 Sumps / Tanks

- Containment Sump 1
- Containment Sump 2
- CCW Surge Tank A
- CCW Surge Tank B
- PRT
- RCDT

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Gai-Tronics Page/party system (Public Address System)	X	
Plant Radio System	X	
PABX (Private Automatic Branch Exchange System)	X	X
Public Telephone System	X	X
Federal Telephone System (FTS)		X

Table C-3 RCS Reheat Duration Thresholds

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable

RCS Status	Containment Closure Status	Duration
Intact (but **NOT** reduced inventory)	N/A	60 min.*
Not intact **OR** reduced inventory	Established	20 min.*
	NOT established	0 min.

Table C-4 Containment Challenge Conditions

- Containment closure **not** established
- Containment hydrogen concentration > 4%
- Unplanned pressure rise that can breach the Containment barrier

Table C-5 AC Power Sources

Offsite:
- 138 KV switchyard circuit
- 345 KV switchyard circuit

Onsite:
- uEG1
- uEG2

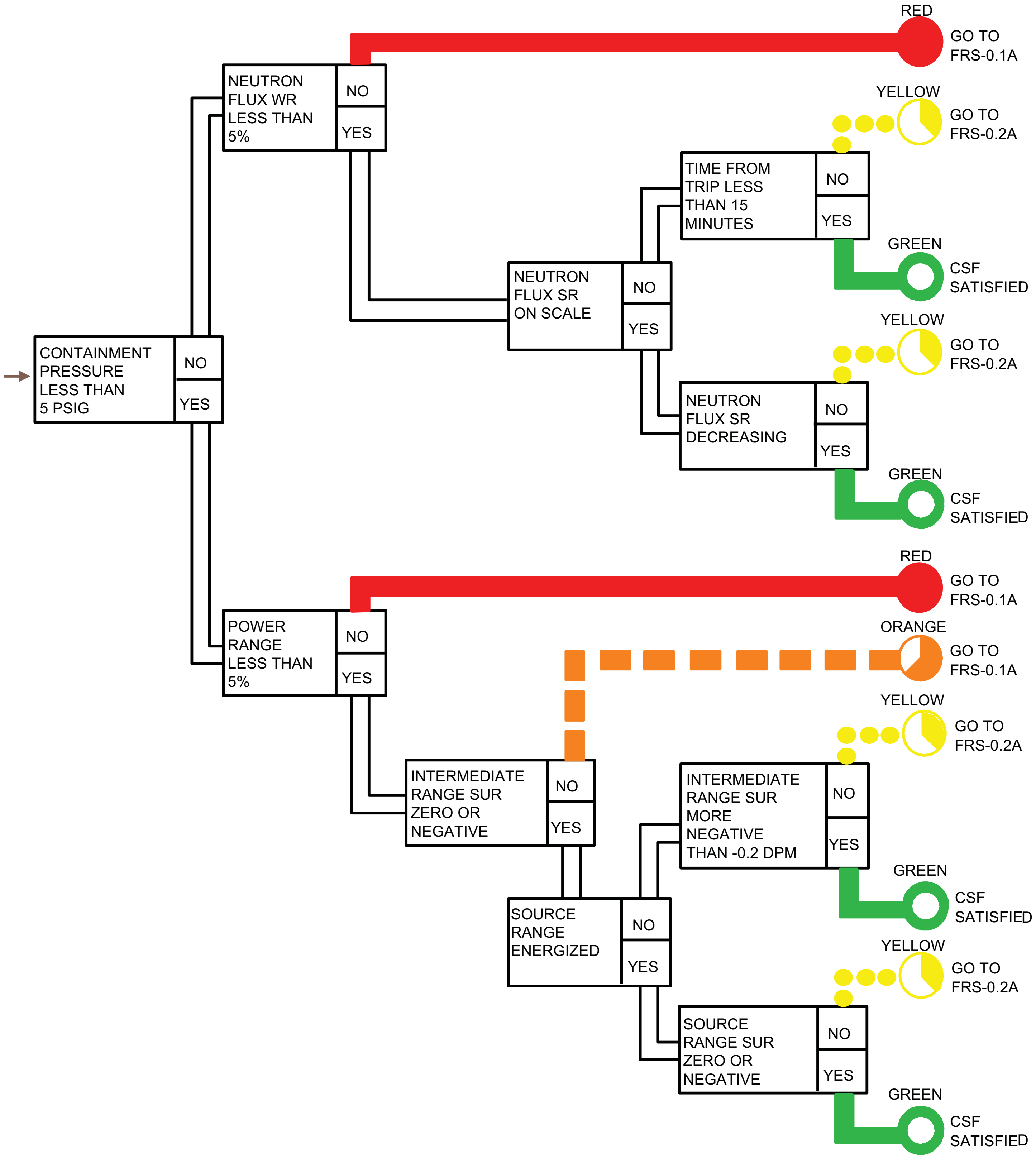
EAL Identifier

XXX.X

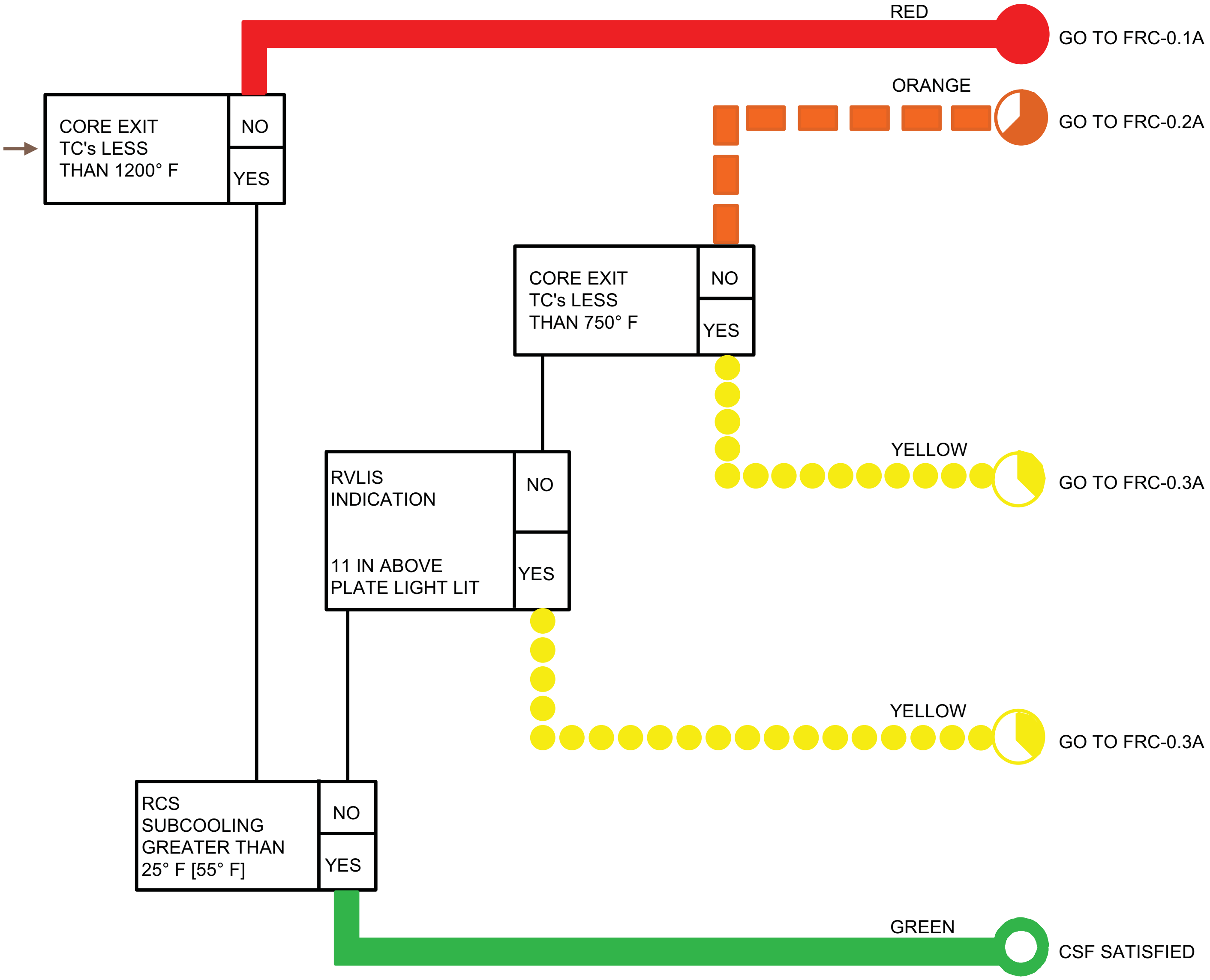
Category (R, H, E, S, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

Prepared for Luminant by: Operations Support Services, Inc. - www.ossi-net.com

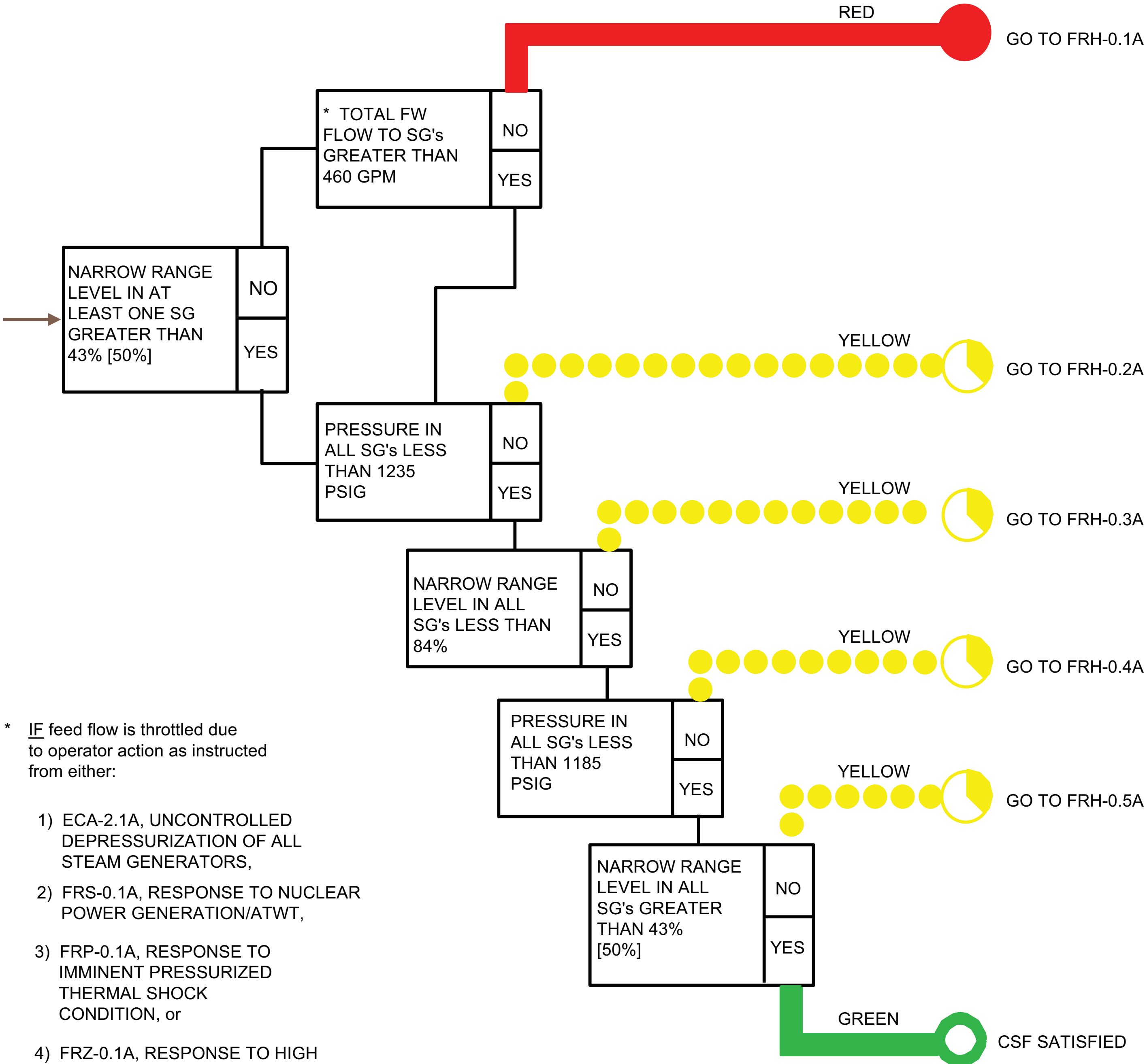
SUBCRITICALITY



CORE COOLING



HEAT SINK

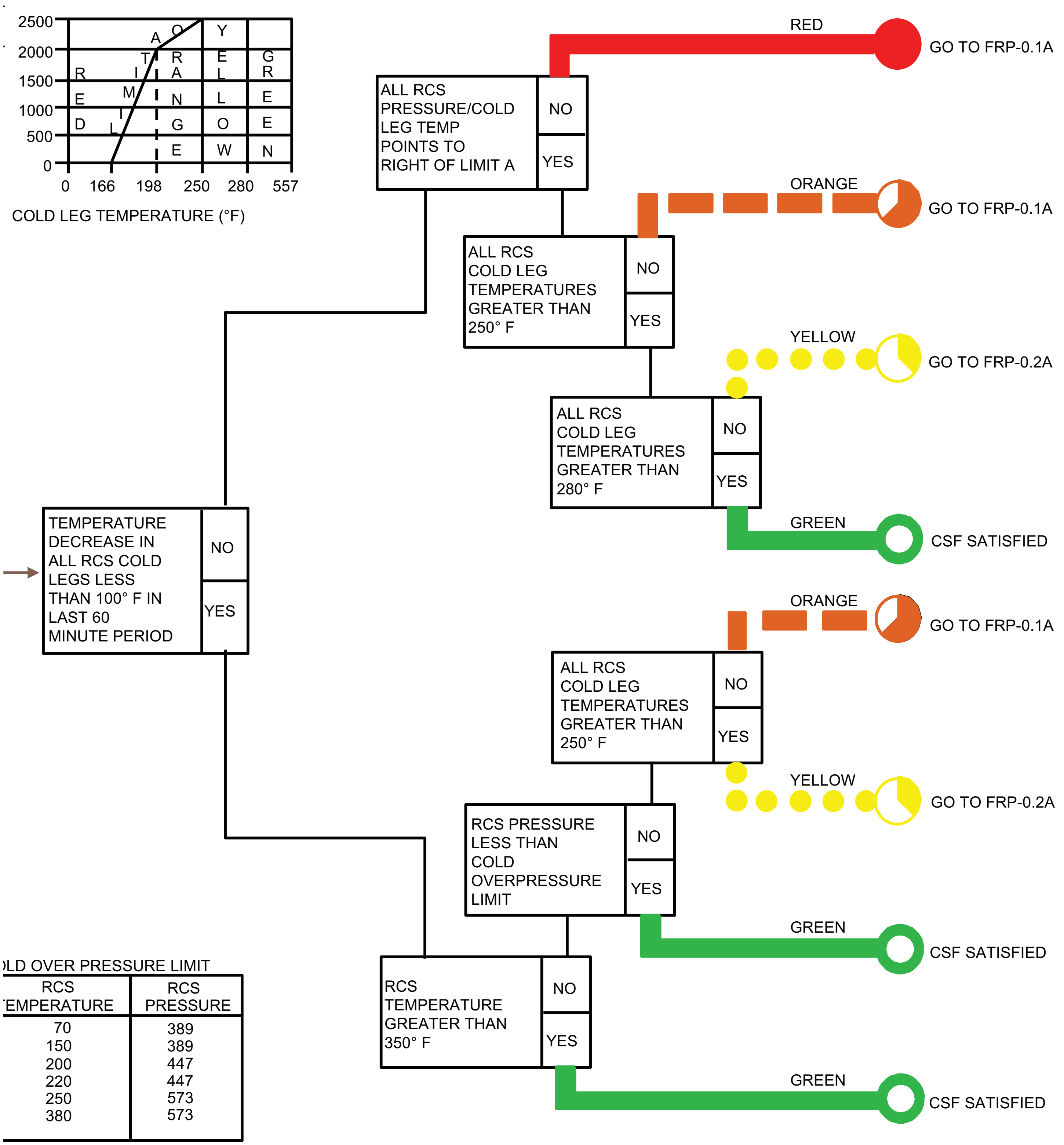


* <u>IF</u> feed flow is throttled due to operator action as instructed from either:

1) ECA-2.1A, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS,
2) FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT,
3) FRP-0.1A, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, or
4) FRZ-0.1A, RESPONSE TO HIGH CONTAINMENT PRESSURE,

<u>THEN</u> FRH-0.1A does not need to be implemented.

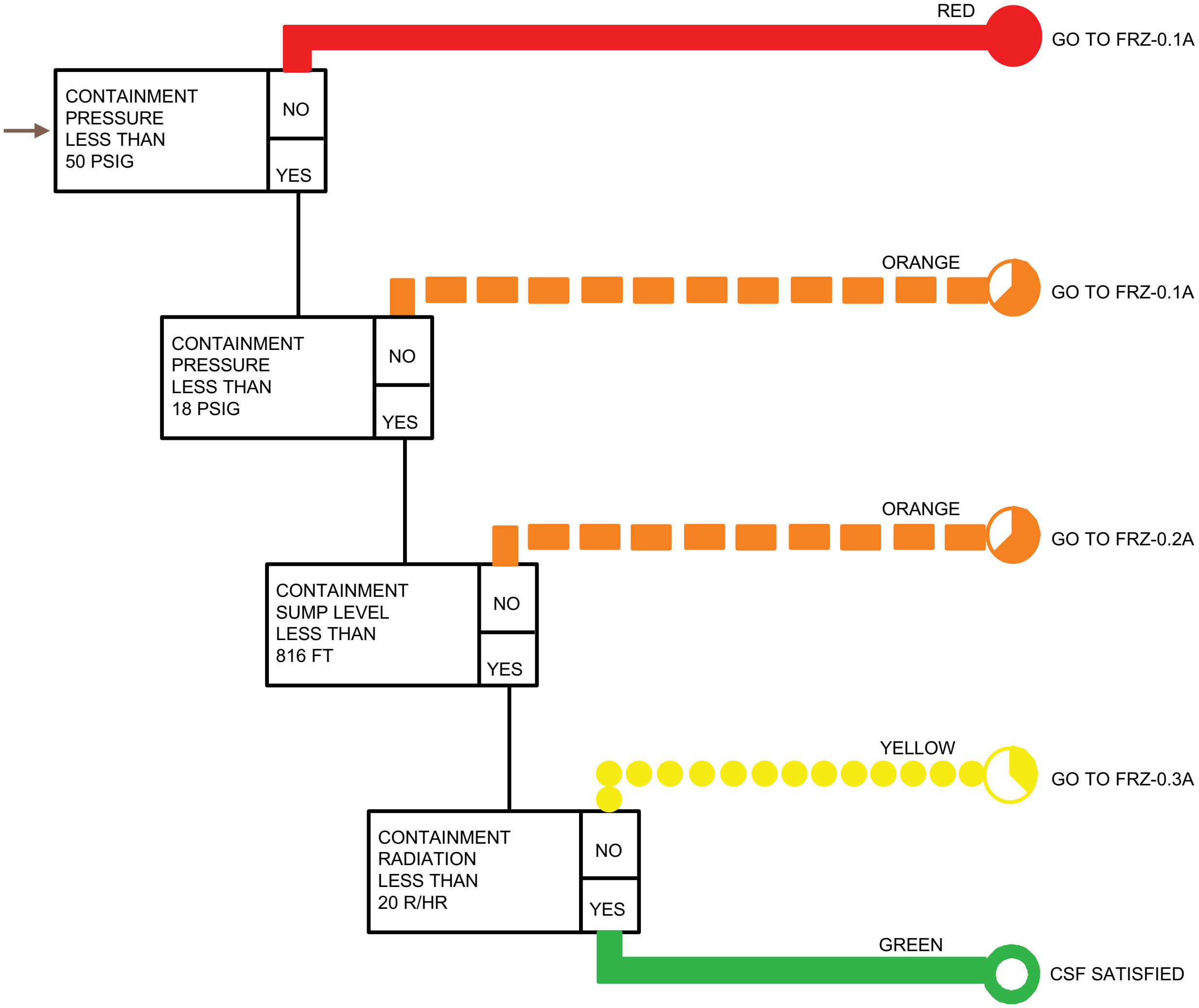
INTEGRITY



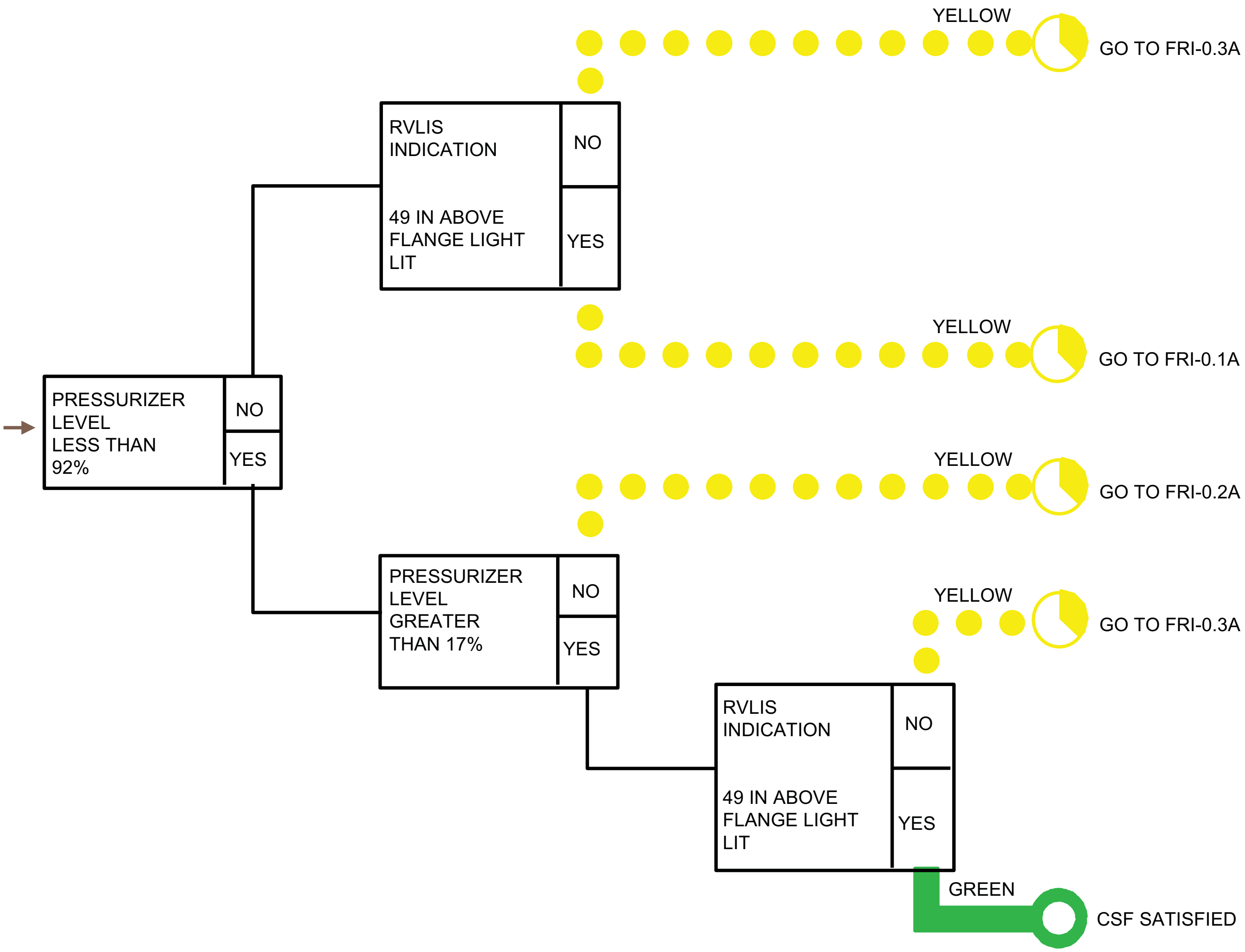
LD OVER PRESSURE LIMIT

RCS EMPERATURE	RCS PRESSURE
70	389
150	389
200	447
220	447
250	573
380	573

CONTAINMENT



INVENTORY


YELLOW
GO TO FRI-0.3A
RVLIS INDICATION
NO
49 IN ABOVE FLANGE LIGHT LIT
YES
YELLOW
GO TO FRI-0.1A
PRESSURIZER LEVEL LESS THAN 92%
NO
YES
YELLOW
GO TO FRI-0.2A
PRESSURIZER LEVEL GREATER THAN 17%
NO
YES
YELLOW
GO TO FRI-0.3A
RVLIS INDICATION
NO
49 IN ABOVE FLANGE LIGHT LIT
YES
GREEN
CSF SATISFIED

COMANCHE PEAK NUCLEAR POWER PLANT

EMERGENCY PLAN MANUAL

LEVEL OF USE:
INFORMATION USE

ASSESSMENT OF EMERGENCY ACTION LEVELS EMERGENCY CLASSIFICATION AND PLAN ACTIVATION

PROCEDURE NO. EPP-201

REVISION NO. 12

SORC Meeting No.: 10-018 Date: 10-14-2010

EFFECTIVE DATE: 11-04-2010 12:00

MAJOR REVISON

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PREPARED BY: (Print): Gary Wiechering EXT: 0180

TECHNICAL REVIEW BY (Print) Kelly Faver EXT: 5628

APPROVED BY: Steve Sewell DATE: 14-Oct-2010

PLANT MANAGER

1.0 PURPOSE (C-01882)

This procedure provides guidance to the Shift Manager, TSC Manager, or EOF Manager to assist in the classification of an emergency as either an “Unusual Event”, “Alert”, “Site Area Emergency”, or “General Emergency”.

2.0 APPLICABILITY

This procedure applies to the Shift Manager, TSC Manager, or EOF Manager in the event of an emergency situation at CPNPP.

3.0 DEFINITIONS/ACRONYMS

3.1 Alert - Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency (EPA) Protection Action Guideline exposure levels. It is the lowest level of classification where near-site or offsite emergency response may be anticipated. For most Alert events, the plant would be brought to a safe condition, and radioactive releases, if any, would be minimal. [C-05703]

3.2 Emergency Action Levels (EALs) – A Pre-determined, Site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter; a discrete, observable event; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

3.3 Emergency Classification - A classification system of emergency severity based on projected or confirmed initiating conditions/emergency action levels. The classes, from least to most severe, are: Unusual Event, Alert, Site Area Emergency and General Emergency.

3.4 Emergency Conditions - Situations which occur that can cause or may threaten to cause hazards affecting the health and safety of employees or the public, or which may result in damage to property.

3.5 General Emergency - The General Emergency classification reflects accident situations involving actual or imminent substantial core degradation or melting with the potential for loss of containment integrity or the potential loss of reactor coolant system integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. [C-05705]

3.6 Site Area Emergency - Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except within the site boundary. [C-05704]

3.7 Unusual Event - Unusual events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected for this classification unless further degradation of safety systems occur. [C-05702]

4.0 INSTRUCTIONS

4.1 General Instructions

NOTE: For the purposes of this procedure, the title Emergency Coordinator is used generically to refer to the position with responsibility for emergency classifications, even though the Emergency Coordinator may not always have this responsibility.

4.1.1 In most cases the decision to declare, upgrade, or proceed to recovery/closeout of an emergency rests with the Emergency Coordinator. When the EOF Manager is the Emergency Coordinator, he may elect to have the TSC Manager retain responsibility for assessing, classifying, and declaring an emergency condition.

4.1.2 The “Emergency Action Level Technical Bases Document” and the “Emergency Action Level Classification Matrix”, cites specific conditions that denote whether the emergency is to be classified as an Unusual Event, Alert, Site Area Emergency or General Emergency. The Emergency Action Level Classification Matrix is provided as guidance to assist the Emergency Coordinator in making that decision. In many cases, a very general statement has been used to denote the emergency action level (EAL) on the Emergency Action Level Classification Matrix. This was done to allow the Emergency Coordinator flexibility to assess any undefinable parameters which may exist. [C-05327]

4.1.3 Plant-specific operator actions required to mitigate the emergency condition are prescribed in the appropriate Abnormal Conditions Procedures or Emergency Operating Procedures (ABN's or EOP's) and are independent of any actions required by this Emergency Plan Procedure.

4.2 Use of the EAL Classification Matrix

4.2.1 The CPNPP EAL scheme includes the following features:

4.2.1.1 Division of the EAL set into three broad groups:

- EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
- EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.
- EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

4.2.1.2 The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

4.2.1.3 Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user.

4.2.1.4 Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CPNPP EAL categories/subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Rad Release / Rad Effluent	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions & Spent Fuel Events 3 – CR/CAS Rad
H – Hazards	1 – Natural or Destructive Phenomena 2 – Fire or Explosion 3 – Hazardous Gas 4 – Security 5 – Control Room Evacuation 6 – Judgment
E – ISFSI	None
Hot Conditions:	
S – System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – Criticality & RPS Failure 4 – Inability to Reach or Maintain Shutdown Conditions 5 – Instrumentation 6 –Communications 7 – Fuel Clad Degradation 8 – RCS Leakage
F – Fission Product Barrier Degradation	None
Cold Conditions:	
C – Cold Shutdown / Refueling System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – RCS Level 4 – RCS Temperature 5 – Communications 6 – Inadvertent Criticality

4.2.1.5 The primary tool for determining the emergency classification level is the EAL Classification Matrix.

- To help in determining the EAL Classification, color coded copies of the EAL Classification Matrix are maintained in the Control Room, the Technical Support Center, and the Emergency Operations Facility and selected other locations.

4.2.1.6 The user of the EAL Classification Matrix may consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration.

4.2.2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

4.2.2.1 If a cell in the Fission Product Barrier Matrix contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

4.2.2.2 Subdivision of the Fission Product Barrier Matrix by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

4.2.2.3 When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of the Fission Product Barrier Matrix, locates the likely category and then reads across the row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category

4.2.2.4 If the EAL-user determines that a Loss threshold has been exceeded, a check mark or circle may be placed in or around the threshold box for the Loss column. This signifies that the threshold barrier is lost. Similarly, this is done for a Potential Loss threshold that has been exceeded.

4.2.2.5 The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if Containment radiation is sufficiently high (i.e., greater than 4,000 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

4.2.3 Classifying Transient Events

4.2.3.1 The key consideration during a Transient Event is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWT event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

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

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

4.2.4 Multiple Events and Classification Upgrading/Downgrading

4.2.4.1 When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached.

- For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Emergency classification level upgrading for multi-unit stations such as CPNPP with shared safety-related systems and functions must also consider the effects of a loss of a common system on more than one unit (e.g. potential for radioactive release from more than one core at the same site).

4.3 Emergency Classification Initial Actions [C-08621]

NOTE: Once indication of an abnormal condition is available, classification declaration must be made within 15 minutes. This time is available to ensure that the classification and subsequent actions associated with the classification, if warranted, are appropriate. It does not allow a delay of 15 minutes if the classification is recognized to be necessary.

It is meant to provide sufficient time to accurately assess the emergency conditions and then evaluate the need for an emergency classification based on the assessment performed. The decision to terminate the event or enter Recovery is NOT time independent.

NOTE: IF a higher classification is made prior to transmitting an event notification, THEN notification for the higher classification can supersede the event notification, provided that it can be performed within the 15-minute timeframe of the previous event. IF the notification of the higher classification cannot be performed within the 15-minute timeframe of the previous event classification, THEN the previous event notification is required within its 15-minute timeframe, and the subsequent event notification is required within its 15-minute timeframe.

CAUTION: Shutdown and outage conditions should be given special consideration since they will likely create abnormalities such as the loss of containment integrity or loss of the RCS pressure boundary (refueling, mid-loop operations, equipment hatch open, etc.). These types of boundary breaches combined with a plant transient (loss of AC power, etc.) may create a worse situation than would be expected if the Unit was at power.

4.3.1 Upon recognition that an abnormal or emergency condition exists, the Shift Manager shall be immediately notified.

4.3.2 Operators shall refer to the appropriate ABN's or EOP's and take actions based upon the indicated symptoms.

4.3.3 The Shift Manager shall evaluate the conditions to determine the need for classifying into one of the four (4) Emergency Classification levels.

4.3.4 If the conditions do not fit any of the general descriptions on the EAL Classification Matrix, the Shift Manager should evaluate the conditions and, if appropriate, classify the emergency based upon professional judgment. If classification is not warranted, no further action is required except to continue monitoring the event.

4.3.5 If the on-duty Shift Manager determines that the conditions fit one or more of the Emergency Classifications shown on the matrix, the Shift Manager shall assume the role of Emergency Coordinator as prescribed in Procedure EPP-109, "Duties and Responsibilities of the Emergency Coordinator/Recovery Manger" and consult his Position Assistant Document (PAD) for further actions. [C-05687, 01278]

4.3.6 When an abnormal or emergency condition is being evaluated, **REFER** to the EAL Classification Matrix and **PERFORM** the following:

- **IDENTIFY** the Unit Mode for the state of the plant prior to the abnormal condition (Operating Modes are identified in respective EALs).
- **REVIEW** the Initiating Conditions applicable to the operating mode as follows.
  - Starting with the highest (General Emergency) classification level on the left side of the matrix and continue to the lowest (Unusual Event) classification level on the right side of the matrix.
  - **If** more than one Initiating Conditions applies to the event, **THEN SELECT** the Initiating Conditions for the highest classification (from all of the Initiating Conditions that were determined to have been met).

- **REVIEW** the EAL Threshold Values for the Initiating Conditions.
  - **If** the EAL Threshold Values have been met or exceeded, **THEN**:
    - **NOTE** the EAL number associated with the Initiating Conditions.
    - **DECLARE** the event. For events affecting both Units, the highest classification on either Unit shall be declared.

4.4 Subsequent Actions [C-05701]

The Shift Manager or Emergency Coordinator shall continually monitor plant conditions and compare the current plant conditions to the EAL Classification Matrix to determine whether a change in emergency classification is warranted and whether to escalate the emergency classification or proceed to EPP-121, “Reentry, Recovery and Closeout”.

5.0 REFERENCES

5.1 Emergency Action Level Technical Bases Document

5.2 CPNPP Emergency Plan, Section 2.0

5.3 EPP-109, "Duties and Responsibilities of the Emergency Coordinator/Recovery Manager"

5.4 NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"

5.5 10CFR, Part 50.72, "Notification of Significant Events"

5.6 NEI 99-01 Rev. 5

5.7 CPSES FSAR Chapter 15

5.8 EPP-121, "Reentry, Recovery and Closeout"

6.0 ATTACHMENTS/FORMS

6.1 Attachments

- 6.1.1 Attachment 1, "Initiating Condition Table"
- 6.1.2 Attachment 2, "EAL Classification Matrix"
- 6.1.3 Attachment 3, "EAL Technical Bases Document"

6.2 Forms

None

Attachment 1
Initiating Condition Table
[C-05327, 05701, 05702, 05703, 05704, 05705, 09308, 26728]

Page 1 of 2

Categories	**GE**	**SAE**	**Alert**	**UE**
ALL Modes				
Abnormal Rad Release / Rad Effluent (R)	Offsite Rad Conditions	Offsite Rad Conditions	Offsite Rad Conditions	Offsite Rad Conditions
			Onsite Rad Conditions & Spent Fuel Events	Onsite Rad Conditions & Spent Fuel Events
			CR/CAS Rad	
Hazards (H)			Natural or Destructive Phenomena	Natural or Destructive Phenomena
			Fire or Explosion	Fire or Explosion
			Hazardous Gas	Hazardous Gas
	Security	Security	Security	Security
		Control Room Evacuation	Control Room Evacuation	
	Judgment	Judgment	Judgment	Judgment
ISFSI				ISFSI

Categories	**GE**	**SAE**	**Alert**	**UE**
HOT Conditions				
System Malfunctions (S)	Loss of AC Power	Loss of AC Power	Loss of AC Power	Loss of AC Power
		Loss of DC Power		
	Criticality & RPS Failure	Criticality & RPS Failure	Criticality & RPS Failure	Criticality & RPS Failure
				Inability to Reach or Maintain Shutdown Conditions
		Instrumentation	Instrumentation	Instrumentation
				Communication
				Fuel Clad Degradation
				RCS Leakage
Fission Product Barriers (F)	Fission Product Barriers	Fission Product Barriers	Fission Product Barriers	Fission Product Barriers

Attachment 1
Initiating Condition Table

[C-05327, 05701, 05702, 05703, 05704, 05705, 09308, 26728]

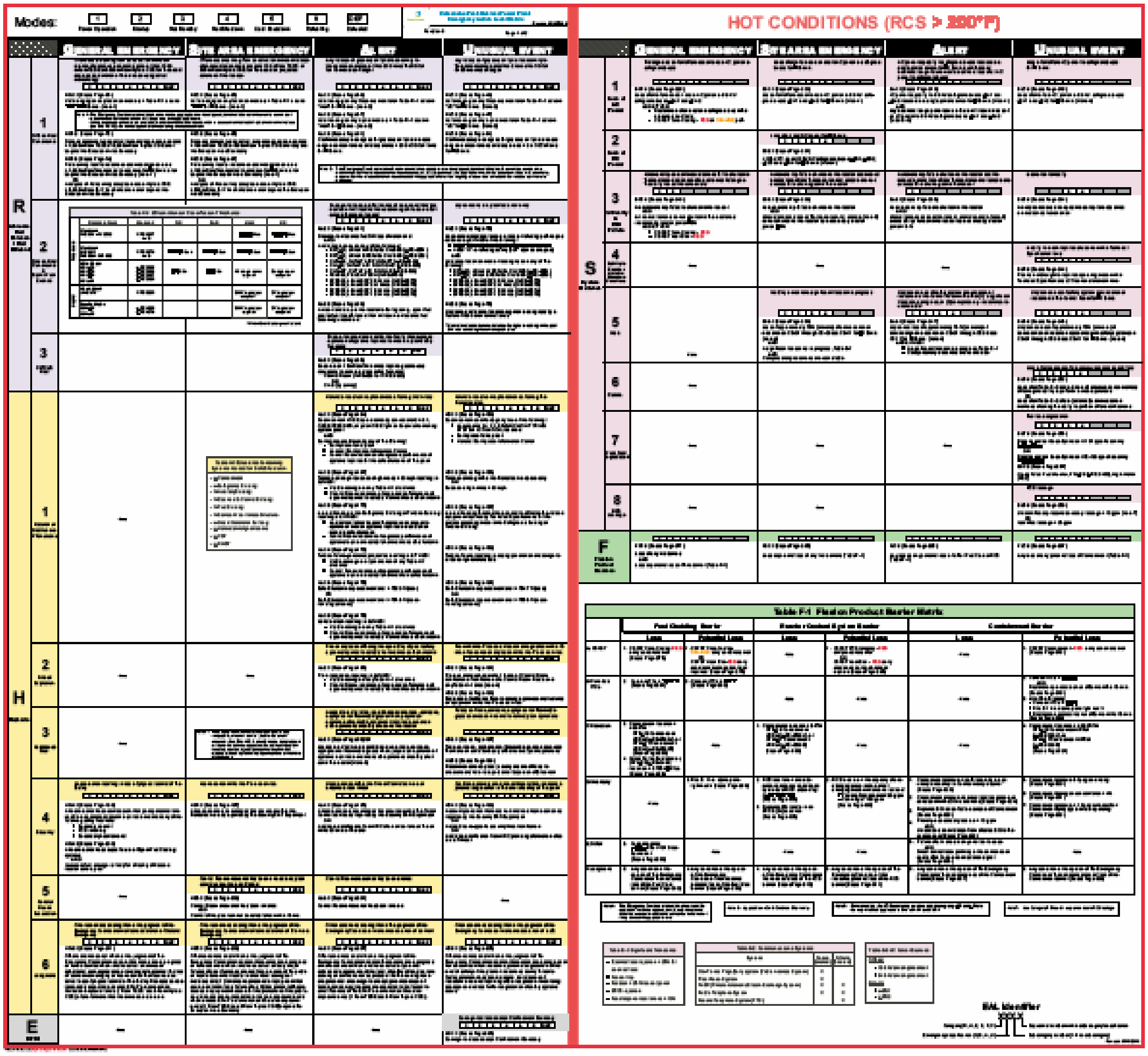
Page 2 of 2

Categories	GE	SAE	Alert	UE
COLD Conditions				
Cold SD / Refueling System Malfunct. (C)			Loss of AC Power	Loss of AC Power
				Loss of DC Power
	RCS Level	RCS Level	RCS Level	RCS Level
			RCS Temp.	RCS Temp.
				Communication
				Inadvertent Criticality

Attachment 2
EAL Classification Matrix
Page 1 of 2

The following EAL Classification Matrix is an example only. There are larger versions of EAL Classification Matrix, which are color coded to help in the determination of the classifications in the Control Room, Technical Support Center, Emergency Operations Facility and selected other locations.

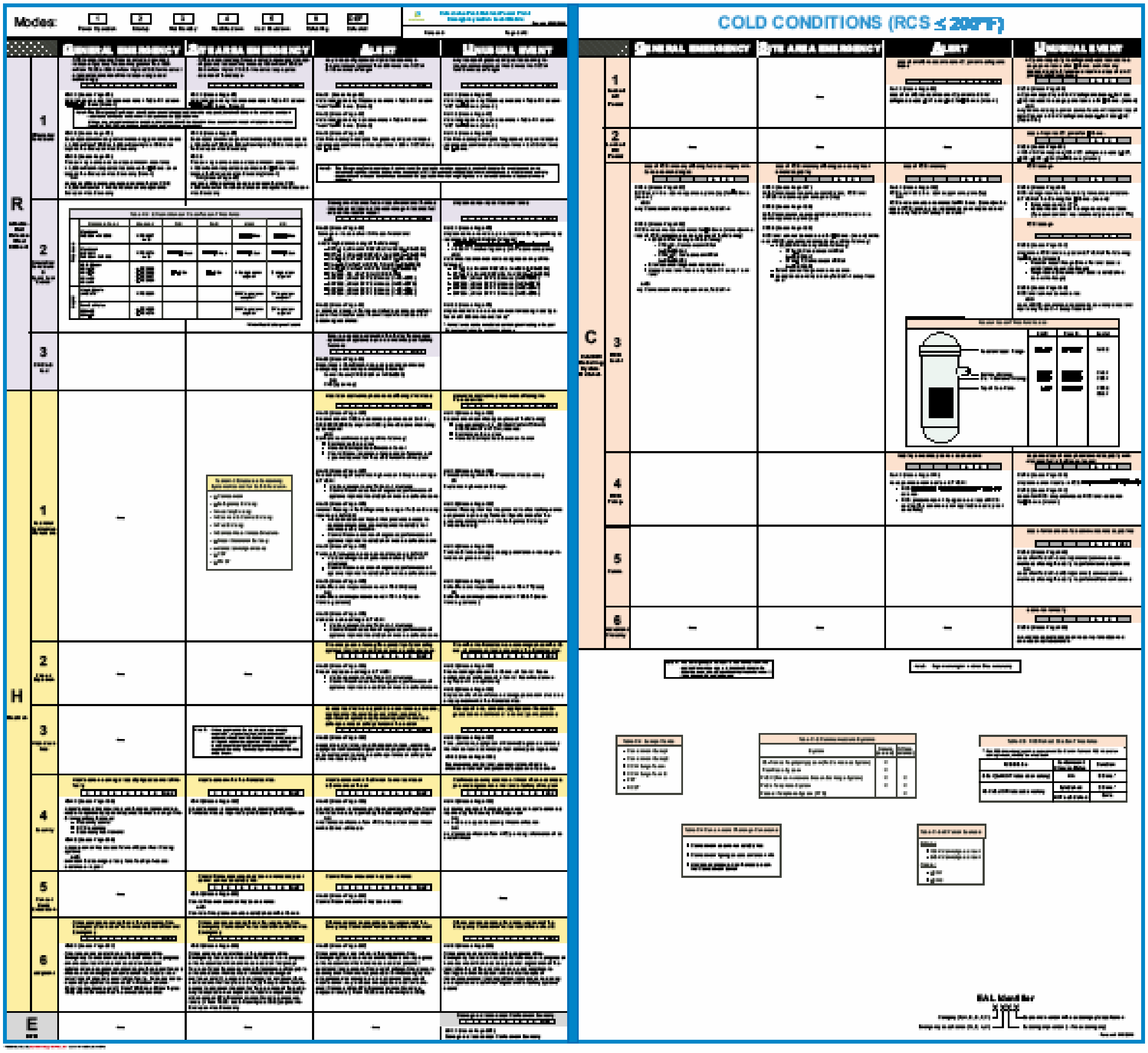
EXAMPLE
All Modes / Hot Conditions



HOT CONDITIONS (RCS > 200°F)

Attachment 2
EAL Classification Matrix
Page 2 of 2

EXAMPLE
All Modes / Cold Conditions



Attachment 3
EAL Technical Bases Document
Page 1 of 1

The EAL Technical Bases Document is a stand alone document that provides an explanation and rationale for each Emergency Action Level (EAL). Decision-makers responsible for implementation of EPP-201, “Assessment of Emergency Action Levels, Emergency Classification and Plan Activation,” may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. Below is the “Table of Contents” for the EAL Technical Bases Document

Section

Facility: CPNPP JPM # NRC S-1 Task #RO1014 K/A #001.A2.17 3.3 / 3.8 SF-1

Title: Respond to a Dropped Control Rod

Examinee (Print): ________________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	______

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:

- Unit 1 is in MODE at 1 at 100% power.
- Control Rod H8 has dropped into the core.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- RESPOND to the dropped Control Rod per ABN-712, Rod Control System Malfunction.

Task Standard: Perform initial actions for a dropped Control Rod and trip the Reactor when a second Control Rod drops per ABN-712.

Required Materials: ABN-712, Rod Control System Malfunction, Rev. 10-2.
EOP-0.0A, Reactor Trip or Safety Injection, Rev. 8-5.

Validation Time: 4 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ________________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-33 or any at 100% power Initial Condition and PERFORM the following:

- **VERIFY Rod Control is in AUTO.**
- **EXECUTE the following malfunctions:**
  - **RD03H8, Dropped Control Rod.**
  - **RP01, Automatic Reactor Trip Failure.**
  - **RP13A, Manual Reactor Trip Switch at CB-07 DISABLED.**
  - **RD03B06, Dropped Control Rod when Rod Control Bank Select Switch is taken to MANUAL.**

BOOTH OPERATOR NOTE:

- **After each JPM:**
  - **VERIFY Rod Control is in AUTO.**
  - **RESET Rod Bank Update when Simulator is INITIALIZED.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **ABN-712, Rod Control System Malfunction.**
  - **Section 3.0, Dropped or Misaligned Rod in MODE 1 or 2.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from ABN-712, Step 3.3.**
Perform Step: 1 1	Verify Number of Rods Misaligned from Step Counter by > 12 steps - ≤ One.
Standard:	DETERMINED only Control Rod B12 Rod bottom light is lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 2	Check Reactor - CRITICAL AND Less than or equal to 100% on highest reading NI AND No Reactor Startup in progress.
Standard:	OBSERVED Nuclear Instruments on CB-07 and DETERMINED Reactor is critical and less than 100%.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**When Rod Control is placed in MANUAL, a 2nd Control Rod will drop.**
Perform Step: 3√ 3	Ensure 1/1-RBSS, CONTROL ROD BANK SELECT - NOT IN AUTO.
Standard:	PLACED 1/1-RBSS, Control Rod Bank Select Switch in MANUAL.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4	RESPOND to Multiple Rod Control System Alarms.
Standard:	DETERMINED Control Rod B06 has dropped and REFERRED to Step 1 RNO.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 5√ 1.a) RNO	IF two or more rods dropped, THEN trip Reactor AND GO TO E0P-0.0A/B.
Standard:	At CB-07, PLACED 1/1-RTC, RX TRIP Switch in TRIP and DETERMINED Reactor is NOT tripped.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 1.a) RNO	<u>IF</u> two or more rods dropped, <u>THEN</u> trip Reactor <u>AND</u> GO TO E0P-0.0A/B.	
Standard:	At CB-10, PLACED 1/1-RT, RX TRIP Switch in TRIP and DETERMINED Reactor is tripped.	
Comment:		**SAT** ☐ **UNSAT** ☐

Examiner Note:	**The following steps are from EOP-0.0A.**	
Perform Step: 7 1, 1.a, & 1st bullet	Verify Reactor Trip: • Verify the following: Reactor trip breakers - AT LEAST ONE OPEN	
Standard:	OBSERVED 1/1-RTBAL & 1/1-RTBBL, RX TRIP BKR green OPEN lights lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Perform Step: 8 1, 1.a, & 2nd bullet	Verify Reactor Trip: • Verify the following: Neutron flux - DECREASING	
Standard:	OBSERVED 1-NI-35B, IR CURRENT CHAN I and 1-NI-36B, IR CURRENT CHAN II are lowering.	
Comment:		**SAT** ☐ **UNSAT** ☐

Perform Step: 9 1 & 1.b	Verify Reactor Trip: • All control rod position rod bottom lights - ON	
Standard:	OBSERVED all Control Rods INSERTED on CTRL ROD POSN bezel.	
Terminating Cue:	**This JPM is complete.**	
Comment:		**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is in MODE at 1 at 100% power.**
- **Control Rod H8 has dropped into the core.**

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

- **RESPOND to the dropped Control Rod per ABN-712, Rod Control System Malfunction.**

3.0 DROPPED OR MISALIGNED ROD IN MODE 1 OR 2

3.1 Symptoms

a. Annunciator Alarms

● PR CHAN DEV	(6D-3.4)
● DRPI ROD DEV	(6D-3.5)
● ANY ROD AT BOT	(6D-3.7)
● ≥2 ROD AT BOT	(6D-4.7)
● QUADRANT PWR TILT	(6D-4.10)

b. Plant Indications

- Plant parameters changing abnormally during rod position changes

NOTE:

- A dropped rod will distort the symmetrical flux distribution of the reactor core. This distortion will be reflected as a deviation in the power range and N16 indications monitored by OPT-102A/B (SR 3.3.1.1.2.a; 3.3.1.1.2.b.;3.3.1.1.6; 3.3.1.1.7). The power range and N16 instrumentation need not be declared inoperable if indications were within the required deviation prior to the event and no other influence has occurred. (SMF-2007-003427)
- For the 12 hour shifty surveillance while in the abnormal condition of a dropped rod, an assessment should be performed that the channels are indicating as expected for the condition of an asymmetrical flux pattern. Since the dropped rod may cause the channels to deviate beyond the normal Channel Check criteria, an assessment is required that the channels are as expected for the plant condition. If required, additional resources (e.g. Core Performance Engineering) may be consulted to assist with the assessment. (SMF-2007-003427)

- NIS Power Range instruments power or AFD indications disagree
- DRPI Rod Bottom Light(s) lit for rods which should be withdrawn
- DRPIs in a bank disagree by greater than 12 steps
- DRPI disagrees with its group step counter by greater than 12 steps

3.2 Automatic Actions

- Possible Reactor trip
- Automatic control rod motion

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify Number of Rods Misaligned from Step Counter by >12 steps - ≤ ONE	a) IF two or more rods **dropped**, THEN trip Reactor AND GO TO EOP-0.0A/B. b) Within 1 hour verify SDM OR initiate boration to restore SDM. c) Within 6 hours place unit in HOT STANDBY per IPO-003A/B (TS 3.1.4).
☐ 2 Check Reactor - CRITICAL AND Less than or equal to 100% on highest reading NI AND No Reactor Startup in progress	Reduce load to less or equal to 1100 MW. IF rod(s) misaligned greater than 12 steps during Reactor startup, THEN perform the following: a. Within 1 hour, insert ALL Control Banks to Control Bank Offset Position. b. Log entry into MODE 3. c. Initiate Tracking LCOAR, as necessary (TS 3.1.4). d. Within 1 hour AND once per 12 hours thereafter, ensure adequate Shutdown Margin per TS 3.1.4: 1) Perform OPT-301. 2) Document per OPT-102A/B. e. Initiate repair per STA-606. f. WHEN all RCCAs are returned to operable status, THEN Perform the following: 1) Reference and position affected rod(s) per IPO-002A/B. 2) Document rod operability per OPT-106A/B.
☐ 3 Ensure 1/u-RBSS, CONTROL ROD BANK SELECT - NOT IN AUTO.	

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 Verify Reactor - STABLE ● Tave-Tref - WITHIN 1°F ● Reactor Power - STABLE	Control Tave AND Reactor Power by controlling the following, as necessary: ● Turbine Power ● Boration ● Dilution ● Steam dumps ● Steam Generator Atmospheric Relief Valves
☐ 5 Verify AXIAL FLUX DIFFERENCE (AFD) - WITHIN LIMITS	Restore ΔFlux to within limits or reduce power within 30 minutes. Refer to TS 3.2.3
☐ 6 Verify "QUADRANT PWR TILT" Alarm (6D-4.10) - DARK	IF Reactor Power is greater than 50% RTP, THEN perform OPT-302 (TS 3.2.4).
☐ 7 Within ONE Hour, Determine Cause of Abnormal Rod Position.	Ensure TS 3.1.4 requirements implemented per LCOAR.

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 8 Check Plant Parameters Indicate ACTUAL Dropped or Misaligned Rod: ● Tave ● AFD ● QPTR ● NIS ● Review Plant Computer CET map for any abnormal indications.	IF DRPI malfunction indicated, THEN GO TO Section 4.0, this procedure.
9 Perform the following: ☐ ● Initiate Repair per STA-606. ☐ ● Direct Chemistry to perform shiftly analysis for fuel defects until plant restored to stable conditions.	

NOTE:
- Either of two realignment methods may be used. The DRPI method is less accurate but may allow quicker realignment with less rod movement. The referencing method is more accurate but requires stepping affected rod full out and may have more adverse effect on flux shape.
- A rod may be recovered within the first 6 hours of the event with no restrictions on rod recovery rate or plant operation (EVAL-2006-0003933-04).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ [C] 10 Contact Reactor and System Engineering and Plant Management prior to realigning Rods. ● Determine if any rod recovery restrictions apply. ● Determine recovery method.	
☐ 11 Within 1 Hour AND Once per 12 Hours Thereafter, Perform OPT-301 to Verify Shutdown Margin (TS 3.1.4).	

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 12 Reduce Turbine and Reactor Power to Level at which Control Banks are LESS THAN OR EQUAL TO Position Prior to Transient OR to Level Sufficient to Withdraw Affected Rod, as determined by Engineering.	

CAUTION: The affected rod(s) shall be realigned to within ±12 steps of group step counter demand position within 1 hour or requirements of Tech Spec 3.1.4 implemented (LCOAR).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 13 Verify DRPI Realignment Method Chosen.	GO TO step 16 for referencing realignment method.
☐ 14 Transfer 1/u-RBSS, CONTROL ROD BANK SELECT, as follows: ● Control Rod affected - MANUAL ● Shutdown Rod Affected - SELECT AFFECTED BANK	

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- Affected rod withdrawal should only be performed after fuel conditioning requirements have been met unless approved by Engineering.
- Do NOT withdraw an RCCA that has been misaligned for greater than 6 hours during power operation without Engineering guidance.

NOTE: The last movement of affected rod should be in the SAME direction as the last movement of affected group.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15 Restore Rod to OPERABLE Status by Realigning as follows within 1 Hour:	Ensure TS 3.1.4 requirements implemented per LCOAR.

☐ a. Move affected group to desired DRPI Light.

☐ b. WHEN stable operating conditions have been established, THEN transfer 1/u-RBSS, CONTROL ROD BANK SELECT to affected bank.

CAUTION: Do NOT allow P/A Converter Auto-Manual selector switch to spring return to automatic until directed by this procedure.

☐ c. Locally maintain P/A Converter Auto-Manual selector switch (SFGD 832 Rm u-096) - MANUAL

"Step continued next page"

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Do NOT make any changes in plant operations during realignment of the affected rod that would require a change in bank position.

15 ☐ d. Record positions for affected rod:

Affected Rod (DRPI) ____

Bank (DRPI) ____

Group 1 step counter ____

Group 2 step counter ____

☐ e. Place all lift coil disconnect switches for affected bank, groups 1 AND 2, EXCEPT for affected rod - ROD DISCONNECTED

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- When moving affected rod, a CONTROL ROD CTRL URGENT FAIL alarm will be received in control room and at power cabinet containing the other group of affected bank. This is normal and will prevent the other group's step counter from operating.
- At low RCS boron concentration, excessive boration may delay return to desired power level after rod recovery.

15 ☐ f. WHEN moving affected rod for realignment, THEN perform the following:

1) Maintain Tave within 2°F of Tref by controlling the following, as necessary:
   - Turbine Power
   - Steam Dumps
   - Boration
   - Dilution
2) Verify that only affected rod is moving.
3) Ensure last movement of affected rod is in same direction as last movement of affected group.

☐ g. Slowly move affected rod until aligned with its group by DRPI indication.

☐ h. Place all lift coil disconnect switches - ROD CONNECTED

“Step continued next page”

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

15 ☐ I. Reset affected bank demand step counter to value recorded in Step 15d.

☐ j. Transfer bank selector switch to MANUAL.

☐ k. Place P/A Converter Auto-Manual selector switch - AUTO

☐ l. GO TO Step 17.

CAUTION: Do NOT make any changes in plant operation during realignment that would require a change in bank position.

16 Perform Referencing Realignment Method as follows:

☐ a. Transfer 1/u-RBSS, CONTROL ROD BANK SELECT to affected bank.

NOTE: Rod Groupings are listed on Attachment 1.

☐ b. Record positions for affected rod:

Affected Rod (DRPI) ____

Bank (DRPI) ____

Group 1 step counter ____

Group 2 step counter ____

"Step continued next page"

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16 ☐ c. Place all lift coil disconnect switches for affected bank, groups 1 AND 2, EXCEPT for affected rod - ROD DISCONNECTED

[C] ☐ d. WHILE performing the following steps, verify that ONLY affected rod moves.

☐ e. Reset affected rod group demand step counter to zero steps.

NOTE: At low RCS boron concentration, excessive boration may delay return to desired power level after rod recovery.

☐ f. Maintain Tave within 2°F of Tref by controlling the following, as necessary:

- Turbine Power
- Steam Dumps
- Boration
- Dilution

"Step continued next page"

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: When moving affected rod, a CONTROL ROD CTRL URGENT FAIL alarm will be received in control room and at power cabinet containing the other group of affected bank. This is normal and will prevent the other group' step counter from operating.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16 ☐ g. Reset P/A converter for affected bank - ZERO per SOP-702A/B	
☐ h. Over a 15 or 30 minute period OR as specified by Engineering, withdraw affected rod until operating step counter is at 232 steps.	
☐ I. Adjust affected step counter to 231 steps.	
☐ j. Insert rod to position recorded in Step 16b, affected Group Step Counter.	
☐ k. Verify P/A converter for affected bank reads value recorded in Step 16b Group Step Counter.	k. Manually reset P/A converter per SOP-702A/B to value recorded in step 16b Group Step Counter.
☐ l. Verify affected rod is at same position as its bank (DRPI).	
☐ m. Place all lift coil disconnect switches - ROD CONNECTED	
☐ n. Transfer 1/u-RBSS, CONTROL ROD BANK SELECT - MANUAL.	

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION: Resetting the Urgent Failure Alarm removes the reduced current applied to movable and stationary grippers. IF cause of alarm has NOT been corrected, THEN resetting alarm may result in dropping rod(s).	
17 Clear the Rod Control Urgent Failure alarm as follows:	
☐ a. Ensure only lift reg white light on designated circuit card in affected cabinet (See ALB-6D 1.6 logic diagram) - LIT	
☐ b. DEPRESS 1/u-RCAR, CONTROL ROD CTRL ALARM RESET	
☐ c. Ensure ALL white lights on designated circuit card in affected cabinet (See ALB-6D 1.6 logic diagram) - DARK	
☐ 18 Adjust Tave to within 1°F of Tref	
☐ 19 Place 1/u-RBSS CONTROL ROD BANK SELECT - AUTO if desired	
NOTE: Verification of OPT-106A/B requirement may be satisfied by documenting rod motion during realignment, at discretion of Shift Manager.	
☐ 20 Verify Rod Restored to OPERABLE status WITHIN 1 HOUR from Time Rod Was Misaligned: • Perform OPT-106A/B OR • Document rod motion greater than or equal to 10 steps in one direction in Unit Log.	Initiate actions of TS 3.1.4B AND initiate LCOAR.

3.3 <u>Operator Actions</u>

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 21 Verify Rod Position Indicators - MATCH ACTUAL POSITIONS ● DRPI ● Step Counters ● P/A Converter ● Bank Overlap Unit ● Plant Computer	Consult Engineering as necessary to determine actual position(s) <u>AND</u> adjust affected indicators to agree (Refer to SOP-702A/B).
22 Verify Rod Position Deviation Monitor - OPERABLE	Initiate LCOAR (TS 3.1.4, 3.1.7, TR 13.1.37).
☐ a. Check "DRPI ROD DEV" (6D-3.5) alarm matches actual conditions: ● ALL shutdown rods greater than 210 steps <u>AND</u> ALL DRPIs within ±12 steps of their group position - WINDOW DARK ● Any shutdown rod less than or equal to 210 steps <u>OR</u> any DRPI greater than or equal to ±12 steps from its group demand position - WINDOW LIT	
☐ b. Check "DRPI URGENT FAIL" (6D-3.6) alarm - DARK	
☐ 23 Refer to EPP-201.	
☐ 24 Initiate a SMART Form per STA-421, as applicable.	

END OF SECTION

Facility: CPNPP JPM # NRC S-2 Task #RO1310 K/A #004.A4.08 3.8 / 3.4 SF-2

Title: Start the PDP and Secure the CCP

Examinee (Print): ________________________________

Testing Method:

Simulated Performance: ______ Classroom: ______

Actual Performance: X Simulator: X

Alternate Path: ______ Plant: ______

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 with Unit 1 in MODE 3:

- Centrifugal Charging Pump (CCP) 1-01 is running.
- A Plant Equipment Operator is at the Positive Displacement Pump (PDP) and has verified stuffing box level is satisfactory and fluid drive is primed.
- Prerequisites of Section 2.5 are met and there has been no significant change in boron concentration since the PDP was last operated.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- START the PDP per SOP-103A, Chemical and Volume Control System, Section 5.3.1, Positive Displacement Pump Startup, START at Step 5.3.1.F.
- SHUTDOWN the CCP per SOP-103A, Chemical and Volume Control System, Section 5.3.4, Centrifugal Charging Pump Shutdown.

Task Standard: Start the PDP and shutdown the running CCP per SOP-103A.

Required Materials: SOP-103A, Chemical and Volume Control System, Rev. 17-19.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ________________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-38 or any at power Initial Condition and PERFORM the following:

- **VERIFY only one Letdown Orifice is in service.**
- **ENSURE 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **SOP-103A, Chemical and Volume Control System.**
    - **Section 5.3.1, Positive Displacement Pump Startup.**
    - **Section 5.3.4, Centrifugal Charging Pump Shutdown.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from SOP-103A, Step 5.3.1.**
Perform Step: 1 5.3.1.F	Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.
Standard:	CONTACTED the NEO and VERIFIED 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.
Examiner Cue:	**The discharge valve is open.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2√ 5.3.1.G & 1st bullet	OPEN the following valves: • 1/1-8202A, VENT VLV (MCB)
Standard:	PLACED 1/1-8202A, VENT VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3√ 5.3.1.G & 2nd bullet	OPEN the following valves: • 1/1-8202B, VENT VLV (MCB)
Standard:	PLACED 1/1-8202B, VENT VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 4 5.3.1.H	Ensure 1APPD, POSITIVE DISPLACEMENT CHARGING PUMP 1-01 MOTOR BREAKER 1EB1/2B/BKR is racked to the CONNECT position.
Standard:	OBSERVED 1APPD, Positive Displacement Charging Pump 1-01 lights and DETERMINED breaker is in the CONNECT position.
Examiner Cue:	**If asked, REPORT breaker in the CONNECT position.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5 5.3.1.I	PLACE 1-SK-459A, PDP SPD CTRL, in MANUAL with demand at 55%.
Standard:	VERIFIED 1-SK-459A, PDP SPD CTRL, in MANUAL with demand at 55%.
Comment:	**SAT** ☐ **UNSAT** ☐

NOTE: The PDP will not start until 1-8109, PD CHRG PMP 1-01 RECIRC VLV, is open and handswitch 1/1-APPD, PDP, is in the START position. Two minutes after the PDP breaker is closed, 1-8109 will automatically close.

Perform Step: 6√ 5.3.1.J	OPEN 1/1-8109, PDP RECIRC VLV.
Standard:	PLACED 1/1-8109, PDP RECIRC VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

NOTE: PDP speed may have to be raised rapidly when a CCP is also in operation to prevent the PDP from stalling on low oil pressure.

Perform Step: 7√ 5.3.1.K	WHEN 1/1-8109, PDP RECIRC VLV is open, THEN start the PDP by placing handswitch 1/1-APPD PDP, to the START position.
Standard:	PLACED 1/1-APPD, PDP in START and OBSERVED red PUMP and FAN lights lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 8 5.3.1.L	Ensure 1/1-8109, PDP RECIRC VLV, is CLOSED.
Standard:	PLACED 1/1-8109, PDP RECIRC VLV in CLOSE and OBSERVED green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

NOTE: During PDP operation the following step may be performed to lower PDP suction stabilizer level.

Perform Step: 9 5.3.1.M & bullet	IF 1/1-8204, H2/N2 SPLY VLV indicates OPEN (red light on), THEN perform the following to lower suction stabilizer level: • OPEN 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV for no more than 10 seconds to clear the high level, then close.
Standard:	OBSERVED 1/1-8204, H2/N2 SPLY VLV green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 10 5.3.1.N & 5.3.1.N.1)	IF a CCP is in operation AND it is to be placed in standby, THEN perform the following: • Ensure only ONE letdown orifice is in service per Section 5.2.3.	
Standard:	DETERMINED only one Letdown Orifice is in service.	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 11√ 5.3.1.N & 5.3.1.N.2)	IF a CCP is in operation AND it is to be placed in standby, THEN perform the following: • Alternately raise PDP speed using 1-SK-459A, PDP SPD CTRL, and lower CCP flow using 1-FK-121, CCP CHRG FLO CTRL, until 1-FK-121 is at minimum.	
Standard:	DEPRESSED 1-FK-121, CCP CHRG FLO CTRL amber MAN pushbutton then alternately RAISED PDP speed using 1-SK-459A, PDP SPD CTRL, and LOWERED CCP flow using 1-FK-121, CCP CHRG FLO CTRL, until 1-FK-121 is at minimum.	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 12 5.3.1.N & 5.3.1.N.3)	IF a CCP is in operation AND it is to be placed in standby, THEN perform the following: • Shut down the running CCP per Section 5.3.4.	
Standard:	REFERRED to Section 5.3.4 to SECURE CCP 1-01.	
Comment:		**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps are from SOP-103A, Step 5.3.4.**	
Perform Step: 13 5.3.4.A	If only one CCP is in operation, place 1-FK-121, CCP CHRG FLO CTRL in MANUAL AND slowly reduce to 0% demand.	
Standard:	VERIFIED 1-FK-121, CCP CHRG FLO CTRL in MANUAL with 0% demand and green output (▼) light lit.	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 14√ 5.3.4.B & 1st bullet	STOP the selected CCP. • 1/1-APCH1, CCP 1
Standard:	PLACED 1/1-APCH1, CCP 1 in STOP and OBSERVED green PUMP and red FAN lights lit.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

INITIAL CONDITIONS: **Given the following conditions with Unit 1 in MODE 3:**

- **Centrifugal Charging Pump (CCP) 1-01 is running.**
- **A Plant Equipment Operator is at the Positive Displacement Pump (PDP) and has verified stuffing box level is satisfactory and fluid drive is primed.**
- **Prerequisites of Section 2.5 are met and there has been no significant change in boron concentration since the PDP was last operated.**

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

- **START the PDP per SOP-103A, Chemical and Volume Control System, Section 5.3.1, Positive Displacement Pump Startup, START at Step 5.3.1.F.**
- **SHUTDOWN the CCP per SOP-103A, Chemical and Volume Control System, Section 5.3.4, Centrifugal Charging Pump Shutdown.**

COMANCHE PEAK NUCLEAR POWER PLANT

UNIT 1

SYSTEM OPERATING PROCEDURE MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN-19 07-28-2010 1200

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

CHEMICAL AND VOLUME CONTROL SYSTEM

PROCEDURE NO. SOP-103A

REVISION NO. 17

EFFECTIVE DATE: 03-05-2008 1200

PREPARED BY (Print): Brad Hancock Ext: 6769

TECHNICAL REVIEW BY (Print): Lisabeth Donley Ext: 6524

APPROVED BY: Alan Hall for Dave Goodwin Date: 02-19-2008
DIRECTOR, OPERATIONS

2.5 PDP Startup

☒ • The CVCS is aligned for operation per Section 5.1.1.

☒ • The PDP oil cooler has CCW flow.

☒ • Demineralized Water is available to the PDP stuffing box coolant tank.

2.6 CCP Startup

N/A

☐ • The CVCS is aligned for operation per Section 5.1.1.

☐ • The CCP Lube oil coolers have SSW flow.

2.7 Placing Demineralizers in service.

☐ • Normal letdown AND charging are in service.

☐ • The demineralizer to be placed in service has been filled with resin.

☐ • The demineralizer is filled and vented.

☐ • Notify Chemistry to determine sample requirements. Except for the Cation Demineralizer boron concentration is required prior to placing a demineralizer in service with results logged in the unit log (may be previously determined).

☐ • No resin transfers or flush operations are in progress for the demineralizers selected.

2.8 Shutting Down CVCS For Outage Work.

☐ • RCS level is stable at a level either above or below the RCP seal package

• Verify that the Number 1 Seal Leakoff Isolation Valve for any RCP NOT on its backseat is CLOSED.

☐ 1/1-8141A, RCP 1 SEAL 1 LKOFF VLV	☐ 1/1-8141C, RCP 3 SEAL 1 LKOFF VLV
☐ 1/1-8141B, RCP 2 SEAL 1 LKOFF VLV	☐ 1/1-8141D, RCP 4 SEAL 1 LKOFF VLV

☐ • To prevent damage to the Reactor Cooling Pump seal packages while seal injection is secured, ensure that Standard Clearance 05450 has been hung.

3.0 PRECAUTIONS

- An explosive mixture of oxygen and hydrogen in the Volume Control Tank and/or PDP suction stabilizer should be avoided at all times. Oxygen content in the tank and stabilizer should not exceed 5% by volume when hydrogen is present.

- During normal operation Volume Control Tank pressure should be maintained high enough to provide a minimum back pressure of 15 psig on the Reactor Coolant Pump Seals. During degas operation, VCT pressure shall be maintained $\geq$10 psig to prevent reverse pressurization of the RCP number 2 seals. Reverse pressurization could result in RCP seal damage.

- After any significant change in letdown and charging flow, the reactor coolant pump seal injection flows should be checked and adjusted if necessary.

- To avoid thermal shock of the reactor coolant piping when operating at elevated temperature, charging flow should first be preheated in the regenerative heat exchanger. Letdown flow should not be stopped without also reducing charging flow to maintain RCP seal injection only when RCS cold leg temperature is > 350°F.

- Pressure downstream of the letdown orifices should be maintained greater than saturation pressure to preclude flashing of the letdown coolant before it enters the letdown heat exchanger.

- When placing a standby demineralizer in service, care should be taken to avoid the insertion of positive reactivity due to absorption of boron in the bed.

[C]
- Except as provided for in EVAL-2007-002946-01, RCP seal injection shall be maintained any time RCS level is above the seal package (84 inches above core plate 830'0") for any RCP not on its backseat.

- Demineralizer resins should be maintained wet per RWS-302.

- The CCP alternate miniflow piping must be filled and vented to ensure the relief valves are not damaged by water hammer in the event of an SI actuation.

- Operation of Demineralizers and associated flow paths has the potential to change RCS Boron Concentration which directly affects Reactivity. Prior to performing evolutions affecting Demineralizers and associated flow paths, ensure all potential effects of the evolution (including potential dilution or boration) are considered. Except for the Cation Demineralizer boron concentration is required prior to placing a demineralizer in service with results logged in the unit log (may be previously determined).

- When placing a Demineralizer in service, minor RCS temperature changes of approximately 0.5°F may be expected. Minor changes in temperature may occur even for a saturated demin which has recently been in service. This is due to the daily change in RCS boron concentration and the minor delta that develops to the demin piping boron.

- Charging pump suction should normally remain aligned to the VCT due to dissolved oxygen concerns when suction comes from the RWST. When entering a plant outage, suctions should NOT be rolled to the RWST prior to crud burst. When time allows, Chemistry should be notified prior to rolling suction to the RWST.

4.0 LIMITATIONS AND NOTES

4.1 Limitations

- During normal operation, maintain VCT pressure between 15 psig and 60 psig.
- During degas operation, maintain VCT at a minimum pressure of 10 psig to prevent damage to the RCP number 2 seals.
- [C] Letdown temperature should not exceed 140°F to the demineralizers.
- Two boron injection subsystems shall be OPERABLE in Modes 1, 2, 3 and 4. (TR 13.1.31)
- One ECCS train shall be OPERABLE in Mode 4. (TS 3.5.3)
- At least one boron injection subsystem shall be OPERABLE and capable of being powered from an OPERABLE emergency power source in MODES 5 and 6. (TR 13.1.32)
- CCP Motor Starting Duty
    1. Motor at ambient temperature: 2 consecutive starts.
    2. Motor at operating temperature: 1 consecutive start.
- Minimum time between starts following conditions 1 or 2.
    a. Motor running between starts - 15 minutes.
    b. Motor standing between starts - 45 minutes.
- [C] The PDP suction stabilizer gas supply and vent valves should be closed and the PDP should be stopped IF the charging pump suction is switched from the VCT to the RWST due to VCT low-low level or operator action. This is applicable when VCT pressure is greater than RWST pressure. Higher VCT pressure will disable the PDP stabilizer vent path and may cause gas binding of the CCP's if 1CS-8200, PD CHRG PMP 1-01 SUCT STAB VNT CHK VLV leaks.
- [C] When the PDP is running and 1/1-8204, H2/N2 SPLY VLV indicates open (red light on), 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV may be opened no more than 10 seconds to clear the high level (1/1-8204 green light on). When 1-ALB-6A, 1.8 "PDP SUCT STAB LVL HI-HI" alarms, operator actions will provide steps to start a CCP and stop the PDP.
- Charging flow through the Regenerative Heat Exchanger is limited to 300 gpm. Due to indication (1-FI-121A), flow is limited to 270 gpm.
- The minimum charging flow from the CCP's with 1-FK-121 in AUTO is 55 gpm. Any charging flow less than 55 gpm will require placing 1-FK-121 in MANUAL.
- Seal injection to the RCP No. 1 seals should not exceed 130°F.
- Seal injection to any RCP No. 1 seal should not exceed 13 gpm.
- [C] Seal injection to any RCP No. 1 seal shall not be less than 6 gpm.
- When RCS temp is $\geq$ 500 degrees, letdown flow is limited to 140 gpm with the 45 gpm orifice and ONE 75 gpm orifice in service.

4.1 Limitations (continued)

- Letdown flow is limited to 170 gpm (when RCS temp is < 500 degrees) with 1 Mixed Bed Demineralizer in service. (Reference EVAL-2005-001409-01-00)
- Letdown flow is limited to 195 gpm (when RCS temp is < 500 degrees) when 2 demineralizers are in service. (Reference FDA-2007-001435-01-00)
- [C] Seal injection to a coupled RCP may be secured if RCS level is above or below the seal package provided that the following actions are implemented to minimize the exposure to risk associated with this configuration:
  - Seal injection should be in service any time RCS level is moving through the seal package.
  - The #1 seal leak off isolation valves should be closed.
  - No pump should be rotated while seal injection is secured; this will prevent cycling water through the shaft alley and seal package.
  - The time with seal injection secured should be limited to the time required to perform maintenance on the Chemical and Volume Control System (CVCS) and testing / surveillances that require seal injection to be secured.
  - The RCP Oil Lift system should remain secured during the time that seal injection is isolated to prevent movement of the shaft and possibly cycling water through the shaft and seal package.
  - The pumps shall be hand rotated with the RCS at Low Pressure and Seal Injection in service to assist in dislodging any debris/deposits, prior to pump operation.
  - A flush of the seals at a higher seal injection flow rate may be used to purge any debris or unfiltered water from the seal package and shaft alley, if necessary.

  Although additional risk is incurred by securing seal injection to a coupled pump, this added risk to the seals may be mitigated by implementing the above actions. (Reference EVAL-2007-002946-01-0)

- During certain conditions, it may be necessary to start the CCP before an operator can be dispatched to locally start the Aux Lube Oil Pump. The start of a CCP without starting the Aux Lube Oil Pump is classified as an emergency start and the following limitations apply:
  - Any emergency start of a CCP should be recorded in the Unit Log.
  - WHEN the Aux Lube Oil Pump has been operated within the last 30-day period THEN CCP bearings retain sufficient lubrication for a CCP start without prior start of the Aux Lube Oil Pump.
  - ABNs and ERGs have been evaluated to determine which instructions for a CCP start are considered to be an “emergency” start of the CCP. WHEN ABN instructions reference that a CCP start be performed per SOP-103A, THEN the Aux Lube Oil Pump is expected to be started prior to starting the CCP. ABN instructions that initiate start of a CCP WITHOUT reference to SOP-103A can be performed as an emergency start. Any CCP start within the ERGs is considered an emergency start.

4.2 Notes

- Attachment 1 and 2 can be used to verify valve and control switch lineups with the system in normal operation.

- The symbol [R] has been located throughout this procedure where real or potential radiation hazards are positively identified. This identification technique should not preclude the worker from following good radiation work practices throughout the task to ensure his/her occupational exposure is maintained As Low As Reasonably Achievable (ALARA).

- The symbol [IV] and [CV] have been located throughout this procedure to identify those steps requiring verification. Initial performance and verification (Independent Verification [IV] or Concurrent Verification [CV]) of these steps shall be documented on the Verification Log Sheet (STA-694-1).

- Following boron saturation of a new demineralizer, RCS boron can be expected to drop. A reduction of RCS boron by 10 to 15 ppm is not unusual. This change is the result of boron being removed from the letdown stream during the saturation evolution and blended flow replacing the boron with boron-10.

- When stopping a CCP, lube oil pressure to the pump bearings will be reduced as the shaft-driven lube oil pump coasts down. When lube oil pressure reduces to < 13 psig, the Aux Lube Oil Pump automatically starts and will automatically stop as the lube oil pressure exceeds > 18 psig. The Aux Lube Oil Pump may cycle a few times (normally 3 to 5 times) before remaining on.

- Modifying notes in attachments appear on the bottom of the applicable page and again on the last page of the attachment.

- The interaction of controllers FK-121, LK-459, & SK-459A is complex. When alternating between PDP & CCP operation, LK-459 must be adjusted to accomplish smooth operation. IF a CCP is operating, steady state demand on LK-459 will be ~1/3 the indicated flow of FI-121A. IF the PDP is operating, the demand of LK-459 will be ~matched to the output of SK-459A. These values assume previous steady state, automatic, 100% power operation, but can be used as guidance for manual adjustments.

[C]
- When Excess Letdown flow is aligned to the top of the VCT, the potential exists to bypass the VCT, supplying non-degassed coolant through the Charging Pump Suction Vent Line to the Charging Pump suctions. Therefore, the Charging Pump Suction Vent Line is isolated prior to aligning the Excess Letdown flow to the VCT. Additionally, since no constant vent path is available in this line-up, a LCOAR is entered and ultrasonic monitoring of the Charging Pump Suction Vent Line initiated. Excess Letdown flow is normally aligned to the suction of the charging pumps.

- Proper oil level for the PDP Fluid Drive Unit is as follows (SMF-05-2603):
  - <u>With the PDP stopped</u>, oil level should be in the upper 1/4 of the MIN - MAX range, not to exceed the MAX oil level mark.
  - <u>With the PDP running</u>, oil level should NOT drop below the MIN oil level mark.

  Overfilling can cause unnecessary heating of the oil, and increased load on the motor

- After each PDP run, the PDP boron placard should be updated to ensure the next PDP run will have current boron concentration information.

5.3 Major Component Operation

5.3.1 Positive Displacement Pump Startup

This section describes the steps to place the PDP in service.

CAUTION: PDP operation may result in high gaseous activity in the PDP Room due to packing leakage.

NOTE:

- PDP run time should be minimized to conserve pump packing. Run PDP only when required (for example - Slave Relay Testing).
- Following loss of Instrument Air, control air to the PDP fluid drive must be reset. This is done by depressing the RESET pushbutton on the instrument air supply to the PDP fluid drive. This RESET is normally accomplished by ABN-301 restoration section 3.0.
- The reactivity impact for starting the PDP pump is typically very small due to diffusion effects between the PDP piping and the RCS. However, assuming no diffusion, the reactivity effects could potentially approach -15 pcm (and -1.5 °F temperature change) with very large (>1000 ppm) boron concentration differences between the PDP piping and RCS. (EVAL-04-0944-04)
- Following several PDP starts, the PDP fluid drive oil may exceed the upper limit mark on the drive unit's sight glass due to priming) the pump (adding oil) over a period of time. The PROMPT Team should be contacted to drain the excess oil. (SMF-01-0600 and 4-03-149515-00)
- With the PDP stopped, oil level should be in the upper 1/4 of the MIN - MAX range, preferably near the MAX level mark. (SMF-05-2603)

☒ A. Ensure the prerequisites in Section 2.5 are met.

B. IF the PDP has not operated for an extended period (month), THEN prime the PDP fluid drive by performing the following:

N/A

☐ 1) IF pump hydraulic fluid level is at the maximum level of the sightglass, THEN instruct a PROMPT member to drain ~1/2 liter of oil into a clean container. (This oil will be added to the fluid drive at step 5.3.1.B. 3)

☐ 2) Remove the pipe plugs from the two priming holes on top of the input end bell (motor side of the fluid drive).

☐ 3) Pour oil (collected in step 5.3.1.B. 1) a) and/or from the approved lubrication list) into either hole until oil rises to the bottom of the other hole and remains there.

☐ 4) Replace and tighten the pipe plugs.

5.3.1

NOTE: R Steps C and D may be considered NA if in a MODE other than MODE 1 or 2. MODE 3

☑ NA C. IF the RCS boron concentration has changed significantly since the PDP was last operated, THEN determine the impact of water in the PDP piping on reactivity as follows:

NOTE: This formula was developed using data from Eval 2004-000944-04-00. It assumes 84 gallons for the PDP piping. All the factors that would not change were calculated to give a constant (0.00128) to simplify the formula(updated in EVAL-2009-000420-02). This formula does not take into account the diffusion effect. So, the boron concentration could be less than the PDP plaque indicates. The temperature change calculated below represents worst case. Operating experience has shown actual temperature change was less than results of the calculation below.

NA

ΔB = RCS Boron Concentration Difference

ΔB = (NA ppm PDP - NA ppm RCS) x 0.00128

ΔB = NA ppm

On the Reactivity Briefing Sheet get the following information:

ITC NA pcm/°F HFP Differential Boron Worth NA pcm/ppm

$$\frac{\text{ITC}}{\text{HFP Differential Boron Worth}} = \frac{\text{NA pcm/°F}}{\text{pcm/ppm}} = \text{NA ppm/°F}$$

$$\Delta\text{Tave} = \frac{\Delta\text{B}}{\text{ppm/°F}} = \frac{\text{NA ppm}}{\text{NA ppm/°F}} = \text{NA °F}$$

☑ NA D. IF ΔTave calculated above is >1°F, THEN notify Shift Operations Manager to discuss contingency actions.

NOTE: If the Stuffing Box Coolant Tank is overfilled, the PDP Charging Pump Room will become contaminated.

E. IF Stuffing Box Coolant Tank is low, THEN fill per the following steps:

☑ NA 1) Slowly crack OPEN 1CS-0119, PD PMP 1-01 STUFFING BOX COOL TK MU ISOL VLV, until desired fill rate is achieved.

☑ 2) When the desired tank level has been established, CLOSE 1CS-0119.

☐ F. Ensure 1-8388-RO, PD CHRG PMP 1-01 DISCH VLV RMT OPER, is OPEN.

G. OPEN the following valves:

☐ ● 1/1-8202A, VENT VLV (MCB)

☐ ● 1/1-8202B, VENT VLV (MCB)

5.3.1

☐ H. Ensure 1APPD, POSITIVE DISPLACEMENT CHARGING PUMP 1-01 MOTOR BREAKER 1EB1/2B/BKR is racked to the CONNECT position.

☐ I. Place 1-SK-459A, PDP SPD CTRL, in MANUAL with demand at 55%.

NOTE: The PDP will not start until 1-8109, PD CHRG PMP 1-01 RECIRC VLV, is open and handswitch 1/1-APPD, PDP, is in the START position. Two minutes after the PDP breaker is closed, 1-8109 will automatically close.

☐ J. OPEN 1/1-8109, PDP RECIRC VLV.

NOTE: PDP speed may have to be raised rapidly when a CCP is also in operation to prevent the PDP from stalling on low oil pressure.

☐ K. WHEN 1/1-8109, PDP RECIRC VLV is open, THEN start the PDP by placing handswitch 1/1-APPD PDP, to the START position.

☐ L. Ensure 1/1-8109, PDP RECIRC VLV, is CLOSED.

NOTE: During PDP operation the following step may be performed to lower PDP suction stabilizer level.

M. IF 1/1-8204, H2/N2 SPLY VLV indicates OPEN (red light on), THEN perform the following to lower suction stabilizer level:

[C] ☐ ● OPEN 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV for no more than 10 seconds to clear the high level, then close.

N. IF a CCP is in operation AND it is to be placed in standby, THEN perform the following:

☐ 1) Ensure only ONE letdown orifice is in service per Section 5.2.3.

☐ 2 Alternately raise PDP speed using 1-SK-459A, PDP SPD CTRL, and lower CCP flow using 1-FK-121, CCP CHRG FLO CTRL, until 1-FK-121 is at minimum.

☐ 3) Shut down the running CCP per Section 5.3.4.

[IV] ☐ O. IF desired, THEN gradually adjust 1-SK-459A, PDP SPD CTRL, to achieve the required flow rate AND place in AUTO.

☐ P. Adjust 1-LK-459, PRZR LVL CTRL, as necessary to maintain stable Pressurizer level.

COMMENTS ______________________________

____

____

5.3.4 Centrifugal Charging Pump Shutdown

This section describes the steps to remove a CCP from service.

☐ A. If only one CCP is in operation, place 1-FK-121, CCP CHRG FLO CTRL in MANUAL AND slowly reduce to 0% demand.

B. STOP the selected CCP.

☐ ● 1/1-APCH1, CCP 1

☐ ● 1/1-APCH2, CCP 2

C. After the CCP has stopped rotating, place the local handswitch for the Aux Lube Oil Pump for selected pump in STOP.

☐ ● 1/1 APCH1-LP, CCP 1-01 AUX LUBE OIL PUMP

☐ ● 1/1 APCH2-LP, CCP 1-02 AUX LUBE OIL PUMP

NOTE: 1-FK-121 is returned to AUTO in the next step to ensure RCP seal injection is maintained should an SI signal occur.

[IV] ☐ D. Ensure 1-FK-121, CCP CHRG FLO CTRL is placed in AUTO.

CAUTION: A maximum of two charging pumps shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F (TS 3.4.12)

[IV] E. IF the unit is in Mode 4 or in Mode 3 with any RCS cold leg temperature less than 350°F, THEN verify at least one CCP handswitch is in AUTO.

☐ ● 1/1-APCH1, CCP 1

☐ ● 1/1-APCH2, CCP 2

[IV] F. IF all RCS cold leg temperatures are > 350°F, THEN ensure both CCP handswitches are in AUTO.

☐ ● 1/1-APCH1, CCP 1

☐ ● 1/1-APCH2, CCP 2

COMMENTS ____

Facility: CPNPP JPM # NRC S-3 Task #RO5024 K/A #010.A4.03 4.0 / 3.8 SF-3

Title: PORV Block Valve Operability Test

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	______

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:

- Unit 1 is in MODE 1 at 100% power.
- Surveillance on the PORV Block Valves is required.
- All Prerequisites have been met.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- PERFORM the PORV Block Valve Operability Test per OPT-109A, PORV Block Valve Test for both Block Valves.
- RECORD data on OPT-109A-1, PORV Block Valve Data Sheet

Task Standard: Perform the PORV Block Valve Operability Test per OPT-109A and take action to isolate an open PORV per ALM-0053, 1-ALB-5C, Window 1.4.

Required Materials: OPT-109A, PORV Block Valve Test, Rev. 10.
OPT-109A-1, PORV Block Valve Data Sheet, Rev. 12.
ALM-0053A, 1-ALB-5C, Window 1.4 - PORV 455A/456 NOT CLOSE, Rev. 6-15.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-18 or any at power Initial Condition and PERFORM the following:

- **VERIFY both PRZR Block Valves are OPEN.**
- **EXECUTE malfunction RX16B, PRZR PORV 456 fails 30% open when 1/1-8000B, PRZR PORV Block Valve is reopened at Step 8.2.3.**

EXAMINER:

PROVIDE the examinee with a copy of Form 1 and Procedure 1:

- **OPT-109A, PORV Block Valve Test.**
- **OPT-109A-1, PORV Block Valve Data Sheet.**

When referenced, PROVIDE the examinee with a copy of Procedure 2:

- **ALM-0053A, 1-ALB-5C, Window 1.4 - PORV 455A/456 NOT CLOSE.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from OPT-109A, Step 8.0.**
Perform Step: 1 8.1 & 8.1.1	Stroke test of 1-8000A, PRZR PORV BLK VLV: • ENSURE 1/1-8000A, PRZR PORV BLK VLV is OPEN.
Standard:	DETERMINED 1/1-8000A, PRZR PORV BLK VLV is OPEN.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 8.1 & 8.1.2	Stroke test of 1-8000A, PRZR PORV BLK VLV: • CLOSE 1/1-8000A, PRZR PORV BLK VLV (RECORD).
Standard:	PLACED 1/1-8000A, PRZR PORV BLK VLV in CLOSE and OBSERVED green CLOSE light lit and RECORDED on Form OPT-109A-1 at Step 8.1.2.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 8.1 & 8.1.3	Stroke test of 1-8000A, PRZR PORV BLK VLV: • OPEN 1/1-8000A, PRZR PORV BLK VLV (RECORD).
Standard:	PLACED 1/1-8000A, PRZR PORV BLK VLV in OPEN and OBSERVED red OPEN light lit and RECORDED on Form OPT-109A-1 at Step 8.1.3.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 8.2 & 8.2.1	Stroke test of 1-8000B, PRZR PORV BLK VLV: • ENSURE 1/1-8000B, PRZR PORV BLK VLV is OPEN.
Standard:	DETERMINED 1/1-8000B, PRZR PORV BLK VLV is OPEN.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 8.2 & 8.2.2	Stroke test of 1-8000B, PRZR PORV BLK VLV: • CLOSE 1/1-8000B, PRZR PORV BLK VLV (RECORD).
Standard:	PLACED 1/1-8000B, PRZR PORV BLK VLV in CLOSE and OBSERVED green CLOSE light lit and RECORDED on Form OPT-109A-1 at Step 8.2.2.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6 8.2 & 8.2.3	Stroke test of 1-8000B, PRZR PORV BLK VLV: • OPEN 1/1-8000B, PRZR PORV BLK VLV (RECORD).
Standard:	PLACED 1/1-8000B, PRZR PORV BLK VLV in OPEN and OBSERVED red OPEN light lit and RECORDED on Form OPT-109A-1 at Step 8.2.3.
Comment:	**SAT ☐ UNSAT ☐**

Booth Operator:	**When 1/1-8000B is reopened, EXECUTE malfunction RX16B at 30%.**
Perform Step: 7	Acknowledge annunciator 1-ALB-5C, Window 1.4 - PORV 455A/456 NOT CLOSE.
Standard:	ACKNOWLEDGED annunciator 5C, Window 1.4 - PORV 455A/456 NOT CLOSE and RECOGNIZED PORV 456 is OPEN.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Examiner Note:	**When referenced, PROVIDE examinee with a copy of Procedure 2.**
Examiner Note:	**The following steps are from ALM-0053A, 1-ALB-5C, Window 1.4.**
NOTE: 1/1-PCV-455A, PRZR PORV and 1/1-PCV-456, PRZR PORV will relieve at approximately 2335 psig. 1/1-PCV-455A, PRZR PORV is interlocked with 1-PI-458 to close at 2185 psig. 1/1-PCV-456, PRZR PORV is interlocked with 1-PI-457 to close at 2185 psig.	
Perform Step: 8 1	Determine affected PORV.
Standard:	DETERMINED affected PORV is 1/1-PCV-456.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9 2 & 2.A	Monitor pressurizer pressure. • If one channel is indicating > 60 psig difference between the remaining operable channels, go to ABN-705.
Standard:	DETERMINED all Pressurizer pressure indications are reading approximately the same value.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**1/1-PCV-456, PRZR PORV will be stuck in mid-position.**
Perform Step: 10 2, 2.B, & 2nd bullet	Monitor pressurizer pressure. • If reactor is in Mode 1, 2 or 3 with pressurizer pressure < 2335 psig, close affected PORV. • 1/1-PCV-456, PRZR PORV
Standard:	PLACED 1/1-PCV-456, PRZR PORV in CLOSE and OBSERVED red OPEN and green CLOSE lights lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 11 4, 4.A, & 2nd bullet	Verify pressurizer or RCS wide range pressure stabilizes. • If pressure continues to decrease due to PORV leakage, close both PORV block valves and determine affected PORV. • 1/1-8000B, PRZR PORV BLK VLV
Standard:	PLACED 1/1-8000B, PRZR PORV BLK VLV in CLOSE and OBSERVED green CLOSE light lit.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is in MODE 1 at 100% power.**
- **Surveillance on the PORV Block Valves is required.**
- **All Prerequisites have been met.**

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

- **PERFORM the PORV Block Valve Operability Test per OPT-109A, PORV Block Valve Test for both Block Valves.**
- **RECORD data on OPT-109A-1, PORV Block Valve Data Sheet**

COMANCHE PEAK NUCLEAR POWER PLANT
UNIT 1
OPERATIONS TESTING MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE______________________

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

LEVEL OF USE:
CONTINUOUS USE

QUALITY RELATED

PORV BLOCK VALVE TEST

PROCEDURE NO. OPT-109A

REVISION NO. 10

EFFECTIVE DATE: 7/15/10 1200

SURVEILLANCE TEST

PREPARED BY: (Print) J.D. STONE EXT. 0564

TECHNICAL REVIEW BY: (Print) EDITORIAL REVISION EXT. NA

APPROVED BY: Bart Smith for Steven Sewell DATE: 7/1/10
DIRECTOR, OPERATIONS

1.0 PURPOSE

This procedure satisfies SR 3.4.11.1 and a portion of 5.5.8 requirements for PORV block valves by the performance of a stroke test.

Section 8.1: 1-8000A
1-8000B

2.0 ACCEPTANCE AND REVIEW CRITERIA

2.1 Acceptance Criteria

2.1.1 The acceptance criteria are listed on the data sheet.

3.0 DEFINITIONS/ACRONYMS

None

4.0 REFERENCES

4.1 Performance

4.1.1 Technical Specification 3.4.11 "Pressure Operated Relief Valves"

4.1.2 Technical Specification 3.4.12 "Low Temperature Overpressure Protection (LTOP) System"

4.1.3 Technical Specification 5.5.8 "Inservice Testing Program"

4.2 Development

4.2.1 FSAR Section 5.4.13 "Safety & Relief Valves"

4.2.2 FSAR Section 15.6.1 "Inadvertent Opening of a Pressurizer Safety or Relief Valve"

4.2.3 M1-0250 "Flow Diagram Reactor Coolant System"

4.2.4 M1-0251 "Flow Diagram Reactor Coolant System"

5.0 PRECAUTIONS, LIMITATIONS AND NOTES

5.1 Precautions

5.1.1 Prior to positioning a valve for stroke testing, the potential consequences on the System or component OPERABILITY shall be evaluated with the valve in the abnormal position. IF the valve stroke impacts OPERABILITY of the System or component, THEN a LCOAR entry shall be made to track the condition.

5.2 Limitations

5.2.1 Each PORV and associated block valve shall be OPERABLE in MODES 1,2 and 3, per TS 3.4.11.

5.2.2 An LTOP system shall be OPERABLE per the requirements of TS 3.4.12 in MODES 4,5 and 6.

5.2.3 IF any acceptance criteria are not met, THEN immediately notify the Shift Manager to refer to the applicable TS.

5.3 Notes

None

6.0 PREREQUISITES

☐ 6.1 This test may be performed in any MODE.

☐ 6.2 The PORV block valve to be tested is NOT closed per Required Actions for Conditions A, B, or E in TS 3.4.11.

7.0 TEST EQUIPMENT

None

8.0 INSTRUCTIONS

> NOTE: Record all data on Form OPT-109A-1.

8.1 Stroke test of 1-8000A, PRZR PORV BLK VLV:

☐ 8.1.1 ENSURE 1/1-8000A, PRZR PORV BLK VLV is OPEN.

☐ 8.1.2 CLOSE 1/1-8000A, PRZR PORV BLK VLV (RECORD).

☐ 8.1.3 OPEN 1/1-8000A, PRZR PORV BLK VLV (RECORD).

8.2 Stroke test of 1-8000B, PRZR PORV BLK VLV:

☐ 8.2.1 ENSURE 1/1-8000B, PRZR PORV BLK VLV is OPEN.

☐ 8.2.2 CLOSE 1/1-8000B, PRZR PORV BLK VLV (RECORD).

☐ 8.2.3 OPEN 1/1-8000B, PRZR PORV BLK VLV (RECORD).

8.3 Independently VERIFY pressurizer PORV block valves are OPEN:

☐ ● 1/1-8000A, PRZR PORV BLK VLV

☐ ● 1/1-8000B, PRZR PORV BLK VLV

9.0 RESTORATION

None

10.0 ATTACHMENTS/FORMS

10.1 Attachments

None

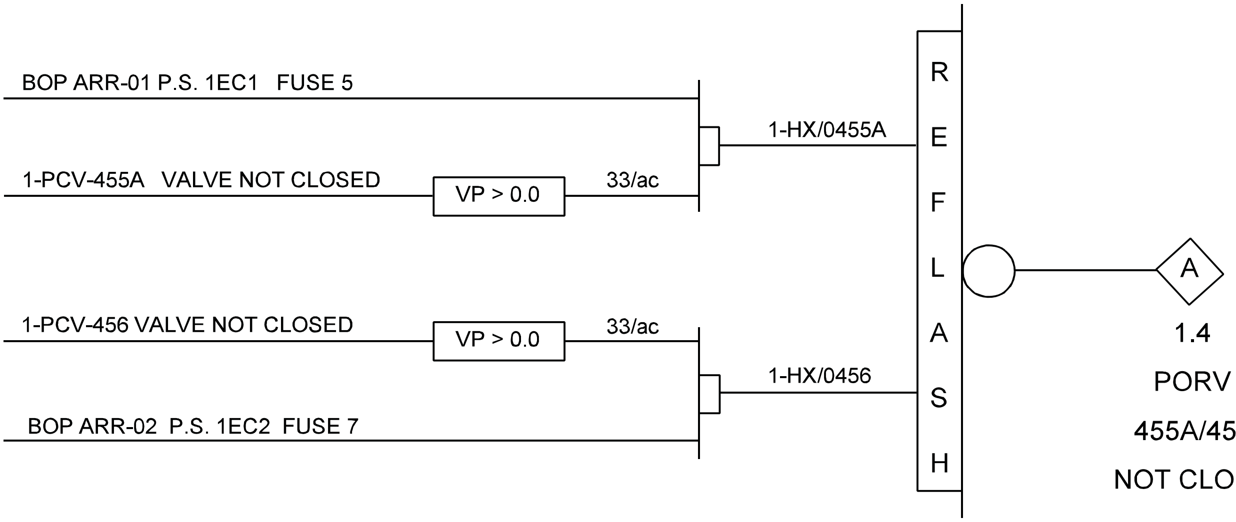
10.2 Forms

10.2.1 OPT-109A-1 "PORV Block Valve Data Sheet"

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0053A
ALARM PROCEDURE 1-ALB-5C	REVISION NO. 6	PAGE 12 OF 61

ANNUNCIATOR NO.: **1.4**

LOGIC:



PLANT COMPUTER:

P6480A	PRZR PRESS CHAN I	Y6469D	PRZR PORV (1-PCV-455A)
P0481A	PRZR PRESS CHAN II	Y6780D	PRZR PORV (1-PCV-456)
P0482A	PRZR PRESS CHAN III	P6498A	HL 1 PRESS (WR)
P6483A	PRZR PRESS CHAN IV	P6499A	HL 4 PRESS (WR)

LOCAL INSTRUMENTS:

None

REFERENCES:

M1-0251
E1-0018 Sh.D, F
E1-0064 Sh.11,12

ANNUNCIATOR NOM./NO.: **PORV 455A/456 NOT CLOSE** **1.4**

PROBABLE CAUSE:

High pressurizer pressure
Instrument malfunction
PORV malfunction
Blown control power fuse

AUTOMATIC ACTIONS: None

NOTE: 1/1-PCV-455A, PRZR PORV and 1/1-PCV-456, PRZR PORV will relieve at approximately 2335 psig. 1/1-PCV-455A, PRZR PORV is interlocked with 1-PI-458 to close at 2185 psig. 1/1-PCV-456, PRZR PORV is interlocked with 1-PI-457 to close at 2185 psig.

OPERATOR ACTIONS:

CAUTION: When a safety valve actuation has resulted in plant shutdown, subsequent Mode 4 operation shall not be commenced until affected safety valve has been inspected.

1. Determine affected PORV.

2. Monitor pressurizer pressure.
   A. If one channel is indicating >60 psig difference between the remaining operable channels, go to ABN-705.
   B. If reactor is in Mode 1, 2 or 3 with pressurizer pressure <2335 psig, close affected PORV.
      - 1/1-PCV-455A, PRZR PORV
      - 1/1-PCV-456, PRZR PORV

3. With reactor in Mode 4, 5 or 6, refer to TDM-301A to determine RCS pressure and temperature limits.
   - 1-TR-413A/23A, HL 1 & 2 WR TEMP
   - 1-TR-413B/23B, HL 1 & 2 WR TEMP
   - 1-TR-433B/23B, HL 3 & 4 WR TEMP
   - 1-TR-433A/23A, HL 3 & 4 WR TEMP
   - 1-PI-403A, HL 4 PRESS (NR)
   - 1-PI-403, HL 4 PRESS (NR)
   - 1-PI-405, HL 1 PRESS (WR)
   - 1-PR-437, HL 1 WR PRESS

   A. If RCS pressure is within the limits based on current RCS temperature, close affected PORV.
      - 1/1-PCV-455A, PRZR PORV
      - 1/1-PCV-456, PRZR PORV

4. Verify pressurizer or RCS wide range pressure stabilizes.
   A. If pressure continues to decrease due to PORV leakage, close both PORV block valves and determine affected PORV.
      - 1/1-8000A, PRZR PORV BLK VLV
      - 1/1-8000B, PRZR PORV BLK VLV

CONTINUED...

ANNUNCIATOR NOM./NO.: **PORV 455A/456 NOT CLOSE** **1.4**

OPERATOR ACTIONS: (Continued)

5. Monitor PRT pressure, temperature and level.
   A. If PRT parameters do not stabilize, perform OPT-303 to determine leakage rate.

   B. If excessive PORV seat leakage is indicated:

      1) Ensure affected PORV block valve is closed.
         - 1/1-8000A, PRZR PORV BLK VLV
         - 1/1-8000B, PRZR PORV BLK VLV

      2) Cycle affected PORV open and closed at least two times.
         - 1/1-PCV-455A, PRZR PORV
         - 1/1-PCV-456, PRZR PORV

NOTE: Operational experience has shown that the time required to re-establish a loop seal varies depending on the leak size. Experience indicates that times of 48 to 96 hours are typical. Consult System Engineering if additional guidance is desired.

      3) Close affected PORV block valve to allow block valve loop seal to be reestablished and then reopen affected block valve.
         - 1/1-8000A PRZR, PORV BLK VLV
         - 1/1-8000B PRZR, PORV BLK VLV

      4) Perform OPT-303 to verify PORV seat leakage has been terminated.

NOTE: A PRZR PORV is considered OPERABLE with known seat leakage when:
1. The PORV is capable of being manually cycled, and
2. With the PORV block valve open:
   - The automatic control system can maintain PRZR pressure and level within assumed accident analysis limits (± 30 psig of pressure setpoint and ± 5% of level setpoint), and
   - RCS identified leakage is less than LCO 3.4.13 limits.

6. Refer to TS 3.4.11, 3.4.13 and 3.4.12.

7. Correct the condition or initiate a work request per STA-606.

PORV BLOCK VALVE DATA SHEET

> NOTE: PORV Block valve operated through one complete cycle of full valve travel satisfies SR 3.4.11.1 requirements.

STEP		OBSERVED	ACCEPTANCE CRITERIA	INITIALS
6.0	PREREQUISITES MET	N/A	N/A	_______
8.1.2	1/1-8000A CLOSED	CLOSED/OPEN	CLOSED	_______
8.1.3	1/1-8000A OPEN	OPEN/CLOSED	OPEN	_______
8.2.2	1/1-8000B CLOSED	CLOSED/OPEN	CLOSED	_______
8.2.3	1/1-8000B OPEN	OPEN/CLOSED	OPEN	_______
8.3	INDEPENDENT VERIFICATION			
	● 1/1-8000A OPEN	N/A	N/A	_______
	● 1/1-8000B OPEN	N/A	N/A	_______

COMMENTS/DISCREPANCIES:____

____

____

CORRECTIVE ACTIONS: ____

____

____

PERFORMED BY: ______________________________ DATE:______________
SIGNATURE

REVIEWED BY: ______________________________ DATE: ______________
OPERATIONS MANAGEMENT

CONTINUOUS USE

OPT-109A-1
Page 1 of 1
R-12

Facility: CPNPP JPM # NRC S-4 Task #RO1507 K/A #011.EA1.11 4.2 / 4.2 SF-4P

Title: Transfer Residual Heat Removal Pumps and Safety Injection Pumps to Hot Leg Recirculation

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	______

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 Large Break Loss of Coolant Accident occurred on Unit 1 three (3) hours ago.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- TRANSFER Residual Heat Removal Pumps and Safety Injection Pumps from Cold Leg Recirculation to Hot Leg Recirculation per EOS-1.4A, Transfer to Hot Leg Recirculation.

Task Standard: Transfer Residual Heat Removal Pumps and Safety Injection Pumps from Cold Leg Recirculation to Hot Leg Recirculation per EOS-1.4A.

Required Materials: EOS-1.4A, Transfer to Hot Leg Recirculation, Rev. 8-1.

Validation Time: 12 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ________________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-36 or any post-LOCA Initial Condition and PERFORM the following:

- **EXECUTE malfunction RC08C2, Hot Leg Loop 3 Large Break LOCA.**
- **EXECUTE override D1S18802A, CB-02 1/1-8802A, SI Pumps to RCS Hot Leg Valve OPEN Switch - CLS**

BOOTH OPERATOR NOTE:

- **After each JPM, REMOVE key T-112, RHR System from the following:**
  - **1/1-8840, RHR to Hot Leg 2 and 3 Injection Isolation Valve.**
  - **1/1-8809A, RHR to Cold Leg 1 & 2 Injection Isolation Valve.**
  - **1/1-8809B, RHR to Cold Leg 3 & 4 Injection Isolation Valve.**
  - **1/1-8802A, SI to Hot Leg 2 & 3 Injection Isolation Valve.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **EOS-1.4A, Transfer to Hot Leg Recirculation.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from EOS-1.4A.**
Perform Step: 1 1.a & 1.a.1)	Perform the following to align Train A RHR to Hot Leg Recirculation: • Check RHR Train A Available.
Standard:	DETERMINED RHR Train A available.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√ 1.a, 1.a.2), & bullet	Perform the following to align Train A RHR to Hot Leg Recirculation: • Close RHR TO CL 1 & 2 INJ ISOL VLV: 1/1-8809A
Standard:	INSERTED key T-112, RHR System into 69/1-8809A POWER switch and TURNED to ON position then PLACED 1/1-8809A, RHR TO CL 1 & 2 INJ ISOL VLV in CLOSE and OBSERVED green CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√ 1.a, 1.a.3), & bullet	Perform the following to align Train A RHR to Hot Leg Recirculation: • Open RHRP 1 XTIE VLV: 1/1-8716A
Standard:	PLACED 1/1-8716A, RHRP 1 XTIE VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 1.a, 1.a.4), & bullet	Perform the following to align Train A RHR to Hot Leg Recirculation: • Ensure RHR TO HL 2 & 3 INJ ISOL VLV is open: 1/1-8840
Standard:	INSERTED key T-112, RHR System into 69/1-8840 POWER switch and TURNED to ON position then PLACED 1/1-8840, RHR TO HL 2 & 3 INJ ISOL VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 1.a & 1.a.5)	Perform the following to align Train A RHR to Hot Leg Recirculation: • Verify RHR TO HL 2 & 3 INJ FLO, 1-FI-988.
Standard:	OBSERVED 1-FI-988, RHR TO HL 2 & 3 INJ FLO at ~ 3000 GPM.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6 1.b & 1.b.1)	Perform the following to align Train B RHR to Hot Leg Recirculation: • Check RHR Train B available.
Standard:	DETERMINED RHR Train B available.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 7√ 1.b, 1.b.2), & bullet	Perform the following to align Train B RHR to Hot Leg Recirculation: • Close RHR TO CL 3 & 4 INJ ISOL VLV: 1/1-8809B
Standard:	INSERTED key T-112, RHR System into 69/1-8809B POWER switch and TURNED to ON position then PLACED 1/1-8809B, RHR TO CL 3 & 4 INJ ISOL VLV in CLOSE and OBSERVED green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 8√ 1.b, 1.b.3), & bullet	Perform the following to align Train B RHR to Hot Leg Recirculation: • Open RHRP 2 XTIE VLV: 1/1-8716B
Standard:	PLACED 1/1-8716B, RHRP 2 XTIE VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 9 1.b, 1.b.4), & bullet	Perform the following to align Train B RHR to Hot Leg Recirculation: • Ensure RHR TO HL 2 & 3 INJ ISOL VLV is open: 1/1-8840
Standard:	DETERMINED 1/1-8840, RHR TO HL 2 & 3 INJ ISOL VLV is OPEN.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 10 1.b & 1.b.5)	Perform the following to align Train A RHR to Hot Leg Recirculation: • Verify RHR TO HL 2 & 3 INJ FLO, 1-FI-988.
Standard:	OBSERVED 1-FI-988, RHR TO HL 2 & 3 INJ FLO at ~ 4000 GPM.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 11 2 & 2.a	Align SI Pumps Flow Path For Hot Leg Recirculation: • Check SI Train A available.
Standard:	OBSERVED SIP 1 red FAN and PUMP lights lit with 1-PI- 919, SIP 1 DISCH PRESS and 1-FI-918, DISCH FLO indications.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 12√ 2 & 2.b	Align SI Pumps Flow Path For Hot Leg Recirculation: • Stop SI pump 1.
Standard:	PLACED 1/1-APSI1, SIP 1 handswitch in STOP and OBSERVED green PUMP and red FAN lights lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 13√ 2, 2.c, & bullet	Align SI Pumps Flow Path For Hot Leg Recirculation: • Close SIP 1 XTIE VLV: 1/1-8821A
Standard:	PLACED 1/1-8821A, SIP 1 XTIE VLV in CLOSE and OBSERVED green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 14√ 2, 2.d, & bullet	Align SI Pumps Flow Path For Hot Leg Recirculation: • Open SI TO HL 2 & 3 INJ ISOL VLV: 1/1-8802A
Standard:	INSERTED key T-112, RHR System into 69/1-8802A POWER switch and TURNED to ON position then PLACED 1/1-8802A, SI TO HL 2 & 3 INJ ISOL VLV in OPEN and OBSERVED green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 15 2.d RNO	Go to Step 2f.
Standard:	DETERMINED 1/1-8802A, SI TO HL 2 & 3 INJ ISOL VLV will NOT OPEN and TRANSITIONED to Step 2f per RNO column.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 16 2.f	Verify SI pump 1 discharge flow.
Standard:	OBSERVED 1-FI-918, SIP 1 DISCH FLO at 0 GPM and REFERRED to RNO column.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 17√ 2.f & 2.f.1) RNO	Perform the following: • Stop SI pump 1
Standard:	DETERMINED SIP 1 STOPPED.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 18√ 2.f, 2.f.2), & bullet RNO	Perform the following: • Close SI TO HL 2 & 3 INJ ISOL VLV: 1/1-8802A
Standard:	OBSERVED 1/1-8802A, SI TO HL 2 & 3 INJ ISOL VLV green CLOSE light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 19√ 2.f, 2.f.3), & bullet RNO	Perform the following: • Open SIP 1 XTIE VLV: 1/1-8821A
Standard:	PLACED 1/1-8821A, SIP 1 XTIE VLV in OPEN and OBSERVED red OPEN light lit.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 20√ 2.f & 2.f.4) RNO	Perform the following: • Start SI pump 1 to re-establish Cold Leg Recirculation. Consult Plant Staff to evaluate long term core cooling.
Standard:	PLACED 1/1-APSI1, SIP 1 handswitch in START and OBSERVED red PUMP and FAN lights lit with 1-PI- 919, SIP 1 DISCH PRESS and 1-FI-918, DISCH FLO indications.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **A Large Break Loss of Coolant Accident occurred on Unit 1 three (3) hours ago.**

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

- **TRANSFER Residual Heat Removal Pumps and Safety Injection Pumps from Cold Leg Recirculation to Hot Leg Recirculation per EOS-1.4A, Transfer to Hot Leg Recirculation.**

COMANCHE PEAK STEAM ELECTRIC STATION
UNIT 1
EMERGENCY RESPONSE GUIDELINES

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN 1 10/27/09 1200

__________/__________DMS CROSS VERIFICATION PERFORMED - WORKING COPY
INITIAL & DATE

TRANSFER TO HOT LEG RECIRCULATION

PROCEDURE NO. EOS-1.4A

REVISION NO. 8

EFFECTIVE DATE: 08-03-06 1200

PREPARED BY (Print): BART SMITH **EXT:** 8837

TECHNICAL REVIEW BY (Print): MIKE MANIS **EXT:** 5536

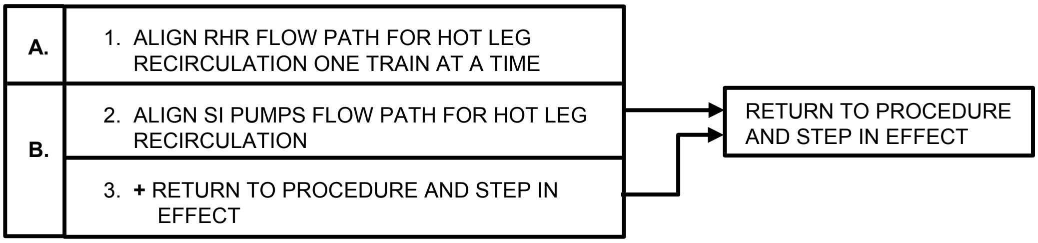
APPROVED BY: A. Hall for R. Smith **DATE:** 7/5/06
DIRECTOR, OPERATIONS

EOS-1.4A TRANSFER TO HOT LEG RECIRCULATION
REV. 8

MAJOR ACTION CATEGORIES

A. ALIGN RHR FLOW PATH FOR HOT LEG RECIRCULATION

B. ALIGN SI PUMPS FLOW PATH FOR HOT LEG RECIRCULATION



+ THE SWITCH BETWEEN HOT LEG AND COLD LEG RECIRCULATION SHOULD BE PERFORMED <u>EVERY 24 HOURS</u> AFTER THE INITIATION OF HOT LEG RECIRCULATION

A. PURPOSE

This procedure provides the necessary instructions for transferring the ECCS system to the hot leg recirculation mode.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2 and 3. This procedure assumes RHR is not in service in the shutdown cooling mode of operation. Using this procedure when not in these modes requires a step by step evaluation to determine if the required action is still applicable to current plant conditions.

C. SYMPTOMS OR ENTRY CONDITIONS

This procedure is entered:

1) From EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT when 3 hours have elapsed.

2) When a decision is made, based upon recommendation of the Plant Staff, that transfer to hot leg recirculation is required. Transfer to hot-leg recirculation might be required eventually, after transferring to cold leg recirculation during the implementation of:
   - EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION
   - ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, or
   - ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Align RHR Flow Path For Hot Leg Recirculation One Train at a Time:	
	a. Perform the following to align Train A RHR to Hot Leg Recirculation:	
	1) Check RHR Train A Available.	1) Go to Step 1b.
	2) Close RHR TO CL 1 & 2 INJ ISOL VLV: • 1/1-8809A	
	3) Open RHRP 1 XTIE VLV: • 1/1-8716A	
	4) Ensure RHR TO HL 2 & 3 INJ ISOL VLV is open: • 1/1-8840	

-CONT 1-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	5) Verify RHR TO HL 2 & 3 INJ FLO, 1-FI-988.	5) Perform the following: A) Close RHRP 1 XTIE VLV: • 1/1-8716A B) Open RHR TO CL 1 & 2 INJ ISOL VLV: • 1/1-8809A C) Verify RHR TO CL 1 & 2 INJ FLO, 1-FI-618. D) Consult with Plant Staff to evaluate long term core cooling. E) IF RHR Train B available, THEN perform Step 1b and attempt to establish Hot Leg Recirculation via that train. F) IF Hot Leg Recirculation from RHR can NOT be established, THEN close RHR TO HL 2 & 3 INJ ISOL VLV, 1/1-8840. Go to Step 2.

-CONT 1-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	b. Perform the following to align Train B RHR to Hot Leg Recirculation:	
	1) Check RHR Train B available.	1) Go to Step 2.
	2) Close RHR TO CL 3 & 4 INJ ISOL VLV: • 1/1-8809B	
	3) Open RHRP 2 XTIE VLV: • 1/1-8716B	
	4) Ensure RHR TO HL 2 & 3 INJ ISOL VLV is open: • 1/1-8840	
	5) Verify RHR TO HL 2 & 3 INJ FLO, 1-FI-988.	5) Perform the following: A) Close RHRP 2 XTIE VLV: • 1/1-8716B B) Open RHR TO CL 3 & 4 INJ ISOL VLV: • 1/1-8809B C) Verify RHR TO CL 3 & 4 INJ FLO, 1-FI-619. D) Consult with Plant Staff to evaluate long term core cooling. E) IF RHR Train A available, THEN perform Step 1a and attempt to establish Hot Leg Recirculation via that train. F) IF Hot Leg Recirculation from RHR can NOT be established, THEN close RHR TO HL 2 & 3 INJ ISOL VLV, 1/1-8840. Go to Step 2.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Align SI Pumps Flow Path For Hot Leg Recirculation:	
	a. Check SI Train A available.	a. Go to Step 2g.
	b. Stop SI pump 1.	

-CONT 2-

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Close SIP 1 XTIE VLV • 1/1-8821A	c. <u>IF</u> 1/1-8821B is available, <u>THEN</u> perform 1). <u>IF</u> 1/1-8821B is <u>NOT</u> available, <u>THEN</u> perform 2). 1) <u>IF</u> 1/1-8821B is available, <u>THEN</u> perform the following: A) Stop SI pump 2. B) Close SIP 2 XTIE VLV: • 1/1-8821B C) Open SI TO HL 1 & 4 INJ ISOL VLV: • 1/1-8802B D) Start SI pump 2. E) Verify SI pump 2 discharge flow. <u>IF</u> <u>NOT</u>, <u>THEN</u> stop SI pump 2 <u>AND</u> close SI TO HL 1 & 4 INJ ISOL VLV, 1/1-8802B. F) Close SI TO CL 1•4 INJ ISOL VLV: • 1/1-8835 G) Open SI TO HL 2 & 3 INJ ISOL VLV: • 1/1-8802A H) Start SI pump 1. I) Verify SI pump 1 discharge flow. <u>IF</u> <u>NOT</u>, <u>THEN</u> stop SI pump 1 <u>AND</u> close SI TO HL 2 & 3 INJ ISOL VLV, 1/1-8802A.

-CONT 2-

J) IF no SI pump running, THEN perform the following:

1. Open SI TO CL 1•4 INJ ISOL VLV, 1/1-8835.
2. Start SI pump 1 to re-establish Cold Leg Recirculation.
3. Open SIP 2 XTIE VLV, 1/1-8821B.
4. Start SI pump 2 to re-establish Cold Leg Recirculation
5. Consult Plant Staff to evaluate long term core cooling.

K) Return to procedure and step in effect.

2) IF 1/1-8821B is NOT available, THEN perform the following:

A) Start SI pump 1 to re-establish Cold Leg Recirculation.

B) Consult Plant Staff to evaluate long term core cooling.

C) Return to procedure and step in effect.

d. Open SI TO HL 2 & 3 INJ ISOL VLV:

- 1/1-8802A

d. Go to Step 2f.

e. Start SI pump 1.

-CONT 2-

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.4A
TRANSFER TO HOT LEG RECIRCULATION	REVISION NO. 8	PAGE 9 OF 23

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	f. Verify SI pump 1 discharge flow.	f. Perform the following: 1) Stop SI pump 1. 2) Close SI TO HL 2 & 3 INJ ISOL VLV: • 1/1-8802A 3) Open SIP 1 XTIE VLV: • 1/1-8821A 4) Start SI pump 1 to re-establish Cold Leg Recirculation. Consult Plant Staff to evaluate long term core cooling.
	g. Check SI Train B available.	g. Go to Step 2m.
	h. Stop SI pump 2.	

-CONT 2-

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

i. Close SIP 2 XTIE VLV:

- 1/1-8821B

i. <u>IF</u> 1/1-8821A is available <u>AND</u> open, <u>THEN</u> perform 1).
<u>IF</u> 1/1-8821A is closed, <u>THEN</u> perform 2).
<u>IF</u> 1/1-8821A is <u>NOT</u> available, <u>THEN</u> perform 3).

1) <u>IF</u> 1/1-8821A is available <u>AND</u> open, <u>THEN</u> perform the following:

A) Stop SI pump 1.

B) Close SIP 1 XTIE VLV:

- 1/1-8821A

C) Open SI TO HL 2 & 3 INJ ISOL VLV:

- 1/1-8802A

D) Start SI pump 1.

E) Verify SI pump 1 discharge flow. <u>IF</u> <u>NOT</u>, <u>THEN</u> stop SI pump 1 <u>AND</u> close SI TO HL 2 & 3 INJ ISOL VLV, 1/1-8802A.

F) Close SI TO CL 1•4 INJ ISOL VLV:

- 1/1-8835

G) Open SI TO HL 1 & 4 INJ ISOL VLV

- 1/1-8802B

H) Start SI pump 2.

-CONT 2-

I) Verify SI pump 2 discharge flow. IF NOT, THEN stop SI pump 2 AND close SI TO HL 1 & 4 INJ ISOL VLV, 1/1-8802B.

J) IF no SI pump running, THEN perform the following:

1. Open SI TO CL 1•4 INJ ISOL VLV, 1/1-8835.

2. Start SI pump 2 to re-establish Cold Leg Recirculation.

3. Open SIP 1 XTIE VLV, 1/1-8821A.

4. Start SI pump 1 to re-establish Cold Leg Recirculation.

5. Consult Plant Staff to evaluate long term core cooling.

K) Return to procedure and step in effect.

-CONT 2-

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

2) IF 1/1-8821A is closed, THEN perform the following:

   A) Ensure SI pump 2 stopped.

   B) Close SI TO CL 1•4 INJ ISOL VLV:

      - 1/1-8835

   C) Open SI TO HL 1 & 4 INJ ISOL VLV:

      - 1/1-8802B

   D) Start SI pump 2.

   E) Verify SI pump 2 discharge flow. IF NOT, THEN stop SI pump 2 AND close SI TO HL 1 & 4 INJ ISOL VLV, 1/1-8802B.

   F) IF SI pump 2 NOT running, THEN perform the following:

      1. Open SI TO CL 1•4 INJ ISOL VLV, 1/1-8835.

      2. Start SI pump 2 to re-establish Cold Leg Recirculation.

      3. Consult Plant Staff to evaluate long term core cooling.

   G) Return to procedure and step in effect.

-CONT 2-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		3) <u>IF</u> 1/1-8821A is <u>NOT</u> available, <u>THEN</u> perform the following: A) Start SI pump 2 to re-establish Cold Leg Recirculation. B) Consult Plant Staff to evaluate long term core cooling. C) Return to procedure and step in effect.
	j. Open SI TO HL 1 & 4 INJ ISOL VLV: • 1/1-8802B	j. Go to Step 21.
	k. Start SI pump 2.	
	l. Verify SI pump 2 discharge flow.	l. Perform the following: 1) Stop SI pump 2. 2) Close SI TO HL 1 & 4 INJ ISOL VLV: • 1/1-8802B 3) Open SIP 2 XTIE VLV: • 1/1-8821B 4) Start SI pump 2 to re-establish Cold Leg Recirculation. Consult Plant Staff to evaluate long term core cooling.

-CONT 2-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	m. Check if SI TO CL 1•4 INJ ISOL VLV, 1/1-8835 should be closed:	
	1) Check that NO SI pump is injecting into cold legs.	1) DO NOT close 1/1-8835. Go to Step 3. OBSERVE NOTE PRIOR TO STEP 3.
	2) Close SI to CL 1•4 INJ ISOL VLV, 1/1-8835.	2) IF 1/1-8835 can NOT be closed, THEN consult Plant Staff to evaluate SI alignment. Go to Step 3. OBSERVE NOTE PRIOR TO STEP 3.
	3) Open one SIP XTIE VLV: • 1/1-8821A -OR- • 1/1-8821B	

NOTE: After the initiation of hot leg recirculation, the action to switch between hot leg and cold leg recirculation should be performed every 24 hours per Attachment 2. Alternating recirculation paths is performed to prevent excessive boron concentration in the reactor during long term operation following a LOCA.

3	Return To Procedure And Step In Effect.	

-END-

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-1.4A, TRANSFER TO HOT LEG RECIRCULATION

1. SECONDARY INTEGRITY CRITERIA

   IF any SG pressure is decreasing in an uncontrolled manner or has completely depressurized and has not been isolated, THEN go to EOP-2.0A, FAULTED STEAM GENERATOR ISOLATION, Step 1.

2. EOP-3.0A TRANSITION CRITERIA

   Manually start SI pumps as necessary and go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1, if any SG level increases in an uncontrolled manner or any SG has abnormal radiation.

3. AFW SUPPLY SWITCHOVER CRITERION

   IF CST level decreases to less than 10%, THEN switch to alternate AFW water supply per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION.

ATTACHMENT 2
PAGE 1 OF 5

TRANSFER TO COLD LEG RECIRCULATION FROM HOT LEG RECIRCULATION

NOTE: The transfer back to Hot Leg Recirculation is normally performed 24 hours after this attachment is complete. The Plant Staff may direct transfer to Hot Leg Recirculation at an interval less than 24 hours based on the possibility of core recriticality due to boron plateout.

1. Align RHR Flow Path For Cold Leg Recirculation One Train At A Time

   a. Perform the following to align Train A RHR to Cold Leg Recirculation:

1) Check RHR Train A in hot leg recirculation.	1) Go to Step 1b.
2) Close RHRP 1 XTIE VLV: • 1/1-8176A	
3) Open RHR TO CL 1 & 2 INJ ISOL VLV: • 1/1-8809A	
4) Verify RHR TO CL 1 & 2 INJ FLO, 1-FI-618.	4) Perform the following: A) Close RHR TO CL 1 & 2 INJ ISOL VLV: • 1/1-8809A B) Open RHRP 1 XTIE VLV: • 1/1-8716A C) Verify RHR TO HL 2 & 3 INJ FLO, 1-FI-988. D) Consult with Plant Staff to evaluate long term core cooling.

-CONT 1-

<u>ATTACHMENT 2</u>
PAGE 2 OF 5

<u>TRANSFER TO COLD LEG RECIRCULATION FROM HOT LEG RECIRCULATION</u>

	E) <u>IF</u> RHR Train B available, <u>THEN</u> perform Step 1b and attempt to establish Cold Leg Recirculation via that train. F) <u>IF</u> Hot Leg Recirculation from RHR can <u>NOT</u> be established, <u>THEN</u> close RHR TO HL 2 & 3 INJ ISOL VLV, 1/1-8840. Go to Step 2.
b. Perform the following to align Train B RHR to Cold Leg Recirculation:	
1) Check RHR Train B in hot leg recirculation.	1) Go to Step 1c.
2) Close RHRP 2 XTIE VLV: • 1/1-8716B	
3) Open RHR TO CL 3 & 4 INJ ISOL VLV: • 1/1-8809B	
4) Verify RHR TO CL 3 & 4 INJ FLO, 1-FI-619.	4) Perform the following: A) Close RHR TO CL 3 & 4 INJ ISOL VLV: • 1/1-8809B B) Open RHRP 2 XTIE VLV: • 1/1-8716B C) Verify RHR TO CL 3 & 4 INJ FLO, 1-FI-619. D) Consult with Plant Staff to evaluate long term core cooling.

-CONT 1-

ATTACHMENT 2
PAGE 3 OF 5

TRANSFER TO COLD LEG RECIRCULATION FROM HOT LEG RECIRCULATION

c. Check if RHR TO HL 2 & 3 INJ ISOL VLV, 1/1-8840 should be closed:	
1) Check that NO RHR pump is injecting into hot legs.	1) DO NOT close 1/1-8840. Go to Step 2.
2) Close RHR TO HL 2 & 3 INJ ISOL VLV: • 1/1-8840	2) IF 1/1-8840 can NOT be closed, THEN consult Plant Staff to evaluate RHR alignment. Go to Step 2.
2. Align SI Pumps Flow Path For Cold Leg Recirculation:	
a. Check SI Train A - ALIGNED IN HOT LEG RECIRCULATION	a. Go to Step 2i.
b. Check SIP 1 XTIE VLV - OPEN: • 1/1-8821A	b. IF SIP 2 XTIE VLV, 1/1-8821B is open AND SI pump 2 NOT aligned in Cold Leg Recirculation, THEN close 1/1-8821B.
c. Stop SI pump 1.	
d. Close SI TO HL 2 & 3 INJ ISOL VLV. • 1/1-8802A	
e. Ensure SIP 1 XTIE VLV open: • 1/1-8821A	
f. Ensure SI TO CL 1•4 INJ ISOL VLV open: • 1/1-8835	
g. Start SI pump 1.	

-CONT 2-

ATTACHMENT 2
PAGE 4 OF 5

TRANSFER TO COLD LEG RECIRCULATION FROM HOT LEG RECIRCULATION

h. Verify SI pump 1 discharge flow.	h. Perform the following: 1) Stop SI pump 1. 2) Close SIP 1 XIE VLV: • 1/1-8821A 3) Open SI TO HL2 &3 INJ ISOL VLV: • 1/1-8802A 4) Start SI pump 1 to re-establish Hot Leg Recirculation flow. Consult Plant Staff to evaluate long term core cooling.
i. Check SI Train B - ALIGNED IN HOT LEG RECIRCULATION	i. Go to Step 3.
j. Stop SI pump 2.	
k. Close SI TO HL 1 & 4 INJ ISOL VLV: • 1/1-8802B	
l. Ensure SIP 2 XTIE VLV open: • 1/1-8821B	
m. Ensure SI TO CL 1•4 INJ ISOL VLV open: • 1/1-8835	

-CONT 2-

ATTACHMENT 2
PAGE 5 OF 5

TRANSFER TO COLD LEG RECIRCULATION FROM HOT LEG RECIRCULATION

n. Start SI pump 2.	
o. Verify SI pump 2 discharge flow.	o. Perform the following: 1) Stop SI pump 2. 2) Close SIP 2 XTIE VLV; • 1/1-8821B 3) Open SI TO HL 1 & 4 INJ ISOL VLV: • 1/1-8802B 4) Start SI pump 2 to re-establish Hot Leg Recirculation flow. Consult Plant Staff to evaluate long term core cooling.

3. Notify Plant Staff to evaluate possibility of core recriticality prior to transfer back to Hot Leg Recirculation.

4. Return To Procedure And Step In Effect.

ATTACHMENT 3
PAGE 1 OF 3

BASES

STEP 1: This step aligns the RHR flow path for the hot leg recirculation mode.

Following the initiation of a LOCA, switchover to hot leg recirculation mode is performed due to boron precipitation concerns. Establishing hot leg recirculation terminates boiling in the core and precludes boron precipitation from the boric acid solution which could potentially hinder core cooling.

Contingent actions are provided to realign RHR to cold leg recirculation if hot leg recirculation can not be established. RHR provides significant core cooling during the recirculation phase of ECCS operation; therefore, the operator is directed to re-establish RHR flow via cold leg recirculation in the event hot leg recirculation can not be established. Plant Staff is informed of the condition to allow consideration for increased possibility of boron plate out in the upper vessel regions and to investigate why hot leg recirculation can not be established. When the condition preventing Hot Leg Recirculation is corrected, the operator should attempt to establish Hot Leg Recirculation at that time.

STEP 2: This step aligns the SI pumps for hot leg recirculation mode.

Following the initiation of a LOCA, switchover to hot leg recirculation mode is performed due to boron precipitation concerns. Establishing hot leg recirculation terminates boiling in the core and precludes boron precipitation from the boric acid solution which could potentially hinder core cooling.

Contingent actions are provided to realign SI to cold leg recirculation if hot leg recirculation can not be established. Cold leg recirculation is re-established to maintain maximum core cooling capability. Plant Staff is informed of the condition to allow consideration for increased possibility of boron plate out in the upper vessel regions and to investigate why hot leg recirculation can not be established. When the condition preventing Hot Leg Recirculation is corrected, the operator should attempt to establish Hot Leg Recirculation at that time.

One SIP Cross-Tie Valve (8821A or 8821B) is reopened following alignment of the SI System to the hot leg recirculation mode to ensure the Cold Leg penetration isolation valve (8835) remains pressurized. During hot leg recirculation, the cold leg penetration is not in service (valves closed) but remains pressurized by the safety injection pumps to a pressure in excess of containment design pressure, which ensures that a leakage path for containment atmosphere does not exist during a LOCA.

ATTACHMENT 3
PAGE 2 OF 3

BASES

NOTE: Alternating recirculation paths is performed to limit excessive boron concentration from being localized.

STEP 3: For larger breaks in the RCS the operator would be transferred from EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, into this procedure. After transfer to hot leg recirculation has been accomplished, the operator should return to EOP-1.0A for further plant recovery.

For some smaller breaks in the RCS or a SGTR plus a LOCA, the operator could enter this procedure after transferring to cold leg recirculation while performing EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION; ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT -SUBCOOLED RECOVERY DESIRED; or ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED. In this case, after the transfer to hot leg recirculation has been completed, the operator should return to the step in effect in the appropriate procedure.

ATTACHMENT 1.A

SECONDARY INTEGRITY CRITERIA - The criteria selected (i.e., steam generator pressure decreasing in an uncontrolled manner or steam generator completely depressurized) are indicative of a break in the secondary pressure boundary. If the operator has not already isolated the faulted steam generator, the operator is instructed to transition to EOP-2.0A to perform the isolation.

EOP-3.0A TRANSITION CRITERIA - Abnormal steam generator radiation or steam generator level increasing in an uncontrolled manner are indicative of primary-to-secondary leakage. The operator is instructed to transition to EOP-3.0A since its associated network deals with minimizing primary-to-secondary leakage whether it be due to a single event or multiple events, such as a steam generator tube rupture in coincidence with a loss of reactor coolant accident.

AFW SUPPLY SWITCHOVER CRITERION - This criterion is on the FOLDOUT PAGE to remind the operator that the supply of water from the CST to the suction of the AFW pumps is limited and, if it is depleted, an alternate suction supply of water to the AFW pumps is necessary.

ATTACHMENT 3
PAGE 3 OF 3

BASES

ATTACHMENT 2

Following the transfer to Hot Leg Recirculation performed by this procedure, the transfer to Cold Leg Recirculation will be performed again at approximately 24 hour intervals. Alternating between hot and cold leg injection is utilized to prevent excessive concentration in the reactor vessel during long-term operation following a LOCA. This attachment provides the actions to accomplish transfer back to Cold Leg Recirculation for Hot Leg Recirculation.

ATTACHMENT 3

The Bases attachment provides a discussion for the steps and attachments of this procedure. The information that forms the basis steps and attachments has been taken from the WOG ERG Background Information or from specific CPSES operating experience or information.

Facility: CPNPP JPM # NRC S-5 Task #RO3007 K/A #041.A4.08 3.0 / 3.1 SF-4S
Title: Transfer Steam Dump System to Steam Pressure Mode

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	______	Plant:	______

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:

- Unit 1 is in HOT STANDBY.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- TRANSFER the Steam Dump System to the Steam Pressure Mode per IPO-009A, Plant Equipment Shutdown Following a Trip, Step 5.45, Transfer Steam Dump System to Steam Pressure Mode.
- PLACE 1-PK-507, Steam Dump Controller in AUTO.

Task Standard: Transfer the Steam Dump System to the Steam Pressure Mode per IPO-009A.

Required Materials: IPO-009A, Plant Equipment Shutdown Following a Trip, Rev. 14-11.

Validation Time: 5 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-38 or any shutdown Initial Condition and PERFORM the following:

- **EXECUTE the following remote functions:**
  - **FWR047, Main Feedwater Pump trip fuses [16, 17, 20, 21] to Auxiliary Feedwater Pumps, Condensate Storage Tank, and Steam Generator Blowdown.**
  - **FWR106, PV-2242, Feedwater Pump Suction Header Pressure override.**

BOOTH OPERATOR NOTE:

- **After each JPM, SET 1-PK-507 potentiometer between 7.5 and 8.0.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **IPO-009A, Plant Equipment Shutdown Following a Trip.**
  - **Step 5.45, Transfer Steam Dump System to Steam Pressure Mode.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from IPO-009A, Step 5.45.**
Perform Step: 1 5.45.A	Ensure 1-PK-507, STM DMP PRESS CTRL is in MANUAL.
Standard:	OBSERVED 1-PK-507, STM DMP PRESS CTRL amber MAN light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2 5.45.B	Match 1-PK-507, STM DMP PRESS CTRL demand to 1-UI-500, STM DMP demand.
Standard:	ADJUSTED 1-PK-507, STM DMP PRESS CTRL RAISE (▲) / LOWER (▼) pushbuttons to MATCH 1-UI-500, STM DMP DEMAND.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 5.45.C	Verify 1-PCIP, 1.4, CNDSR AVAIL STM DMP ARMED C-9 is ON.
Standard:	OBSERVED 1-PCIP-1.4, CNDSR AVAIL STM DMP ARMED C-9 window lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4 5.45.D	Ensure BOTH STM DMP INTLK SELECT switches are ON.
Standard:	OBSERVED 43/1-SDA, STM DMP INTLK SELECT **and** 43/1-SDB, STM DMP INTLK SELECT in ON position.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ 5.45.E	Place 43/1-SD, STM DMP MODE SELECT in STM PRESS and verify proper response of steam dump valves.
Standard:	PLACED 43/1-SD, STM DMP MODE SELECT in STM PRESS position and VERIFIED Steam Dump Valves maintain position.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 5.45.F	Ensure 1-PK-507, STM DMP PRESS CTRL set to 6.86.	
Standard:	ADJUSTED 1-PK-507, STM DMP PRESS CTRL potentiometer to 6.86.	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 7√ 5.45.G	Place 1-PK-507, STM DMP PRESS CTRL in AUTO.	
Standard:	DEPRESSED 1-PK-507, STM DMP PRESS CTRL white AUTO pushbutton and OBSERVED white AUTO light lit.	
Terminating Cue:	**This JPM is complete.**	
Comment:		**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 is in HOT STANDBY.**

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

- **TRANSFER the Steam Dump System to the Steam Pressure Mode per IPO-009A, Plant Equipment Shutdown Following a Trip, Step 5.45, Transfer Steam Dump System to Steam Pressure Mode.**
- **PLACE 1-PK-507, Steam Dump Controller in AUTO.**

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-009A
PLANT EQUIPMENT SHUTDOWN FOLLOWING A TRIP	REVISION NO. 14	PAGE 24 OF 36

[C] 5.44 B. IF RX ≥10% PWR bistable lights are ON, THEN refer to ABN-703.

/ 6/10/11
Initials Date

C. Verify the following:

- ☑ 1-PCIP, 4.5, RX ≤48% PWR 3-LOOP FLO PERM P-8 is ON.
- ☑ 1-PCIP, 4.6, TURB ≤10% PWR P-13 is ON.
- ☑ 1-PCIP, 1.6, RX ≥10% PWR P-10 is OFF.
- ☑ 1-PCIP, 1.2, IR TRN A RX TRIP BLK is OFF.
- ☑ 1-PCIP, 2.2, IR TRN B RX TRIP BLK is OFF.
- ☑ 1-PCIP, 3.2, PR TRN A LO SETPT RX TRIP BLK is OFF.
- ☑ 1-PCIP, 4.2, PR TRN B LO SETPT RX TRIP BLK is OFF.
- ☑ 1-PCIP, 3.5, RX & TURB ≤10% PWR P-7 is ON.
- ☑ 1-PCIP, 2.4, LO TURB PWR ROD WITHDRWL BLK C-5 is ON.
- ☑ 1-ALB-6D, 1.1, SR HI VOLT FAIL is OFF.

/ 6/10/11
Initials Date

D. Verify the following annunciators are OFF:

- ☑ 1-ALB-6C, 3.4, PR FLUX RATE HI
- ☑ 1-ALB-6D, 3.3, 1 OF 4 PR FLUX RATE HI

/ 6/10/11
Initials Date

NOTE: The following step may be marked N/A if performed during the implementation of EOS-0.1A.

N/A

5.45 IF not previously aligned to Steam Pressure Mode, THEN Transfer the Steam Dump System to the Steam Pressure mode as follows:

☐ A. Ensure 1-PK-507, STM DMP PRESS CTRL is in MANUAL.

☐ B. Match 1-PK-507, STM DUMP PRESS CTRL demand to 1-UI-500, STM DMP DEMAND.

☐ C. Verify 1-PCIP, 1.4, CNDSR AVAIL STM DMP ARMED C-9 is ON.

☐ D. Ensure BOTH STM DMP INTLK SELECT switches are ON.

☐ E. Place 43/1-SD, STM DMP MODE SELECT in STM PRESS and verify proper response of steam dump valves.

☐ F. Ensure 1-PK-507, STM DMP PRESS CTRL set to 6.86.

☐ G. Place 1-PK-507, STM DMP PRESS CTRL in AUTO.

☐ H. Verify 1-PI-507, MS HDR PRESS is approximately 1092 psig.

______ / ______
Initials Date

Facility: CPNPP JPM # NRC S-6 Task #RO1702 K/A #103.A2.03 3.5 / 3.8 SF-5
Title: Verify Containment Spray Not Required

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	______

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:

- Unit 1 has just tripped from 100% power.
- A Loss of Coolant Accident is in progress.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- VERIFY Containment Spray Not Required per EOP-0.0A, Reactor Trip or Safety Injection, Step 7.

Task Standard: Determine Containment Spray is required, initiate Containment Spray Phase B actuation, and manually align Containment Spray Phase B Valves per EOP-0.0 A, Attachment 6.

Required Materials: EOP-0.0A, Reactor Trip or Safety Injection, Rev. 8-5.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-32 or any 100% power Initial Condition and PERFORM the following:

- **INSERT Malfunction RP10A, Train A failure.**
- **INSERT Malfunction RP10B, Train B failure.**
- **INSERT Malfunction RP19C, Train A failure.**
- **INSERT Malfunction RC09A2, Loss of Coolant Accident.**
- **FREEZE the Simulator when Containment Spray has actuated and the Containment Spray Heat Exchanger Outlet Valves are full open.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **EOP-0.0A, Reactor Trip or Safety Injection.**
  - **Step 7, Verify Containment Spray Not Required.**
  - **Attachment 6, Containment Spray / Phase B Isolation.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from EOP-0.0A.**
Perform Step: 1 7 & 7.a	Verify Containment Spray Not Required: • Containment pressure - HAS REMAINED LESS THAN 18.0 PSIG • 1-ALB-2B window 1-8, CS ACT - NOT ILLUMINATED **-AND-** • 1-ALB-2B window 4-11, CNTMT ISOL PHASE B ACT - NOT ILLUMINATED
Standard:	DETERMINED Containment pressure greater than 18 PSIG with some windows illuminated.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 2 7.a & 7.a.1) RNO	Perform the following: • Verify Containment Spray and Phase B Actuation initiated. IF NOT, THEN manually actuate.
Standard:	DETERMINED Containment Spray is actuated with 1-ALB-2B, Window 1-8, CS ACT - ILLUMINATED.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 3 7.a & 7.a.1) RNO	Perform the following: • Verify Containment Spray and Phase B Actuation initiated. IF NOT, THEN manually actuate.
Standard:	DETERMINED Containment Phase B is NOT actuated with 1-ALB-2B, Window 4-11, CNTMT ISOL PHASE B ACT - NOT ILLUMINATED.
Comment:	**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Examinee will attempt actuation of Phase B from C-02 and CB-07.**
Perform Step: 4√ 7.a, 7.a.1), & 7.a.2) RNO	Perform the following: • Verify Containment Spray and Phase B Actuation initiated. IF NOT, THEN manually actuate. • Verify appropriate MLB indication for CNTMT SPRAY (BLUE WINDOWS) AND PHASE B (ORANGE WINDOWS).
Standard:	PERFORMED the following to manually actuate Containment Phase B: • PLACED 1/1-CIPBA1A **and** 1/1-CIPBA2A, CS/CNTMT ISOL - PHASE B MAN ACT switches at CB-02 to ACT position. • DETERMINED Containment Phase B did NOT actuate at CB-02.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5√ 7.a, 7.a.1), & 7.a.2) RNO	Perform the following: • Verify Containment Spray and Phase B Actuation initiated. IF NOT, THEN manually actuate. • Verify appropriate MLB indication for CNTMT SPRAY (BLUE WINDOWS) AND PHASE B (ORANGE WINDOWS).
Standard:	PERFORMED the following to manually actuate Containment Phase B: • PLACED 1/1-CIPBA1B **and** 1/1-CIPBA2B, CS/CNTMT ISOL - PHASE B MAN ACT switches at CB-07 to ACT position. • DETERMINED Containment Phase B did NOT actuate at CB-07.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Examiner Note:	**Examinee may verify valve position using either the indicating light at the valve or windows on 1-MLB-4A3 or 1-MLB-4B3.**
Examiner Note:	**The following steps are from EOP-0.0A, Attachment 6.**
Examiner Note:	**Close either or both Train A Valve 4526 or Train B Valve 4527.**
Perform Step: 6√ Item #15 (Page 1 of 2)	CB-03 1-HS-4526, NON-SFGD LOOP CCW SPLY VLV CLOSED
Standard:	PLACED 1-HS-4526, NON-SFGD LOOP CCW SPLY VLV in CLOSE position and VERIFIED green CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7√ Item #13 (Page 1 of 2)	CB-03 1-HS-4527, NON-SFGD LOOP CCW SPLY VLV CLOSED	
Standard:	PLACED 1-HS-4527, NON-SFGD LOOP CCW SPLY VLV in CLOSE position and VERIFIED green CLOSE light lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Close either <u>or</u> both Train A Valve 4524 or Train B Valve 4525.**	
Perform Step: 8√ Item #16 (Page 1 of 2)	CB-03 1-HS-4524, NON-SFGD LOOP CCW RET VLV CLOSED	
Standard:	PLACED 1-HS-4524, NON-SFGD LOOP CCW RET VLV in CLOSE position and VERIFIED green CLOSE light lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Perform Step: 9√ Item #14 (Page 1 of 2)	CB-03 1-HS-4525, NON-SFGD LOOP CCW RET VLV CLOSED	
Standard:	PLACED 1-HS-4525, NON-SFGD LOOP CCW RET VLV in CLOSE position and VERIFIED green CLOSE light lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Close either <u>or</u> both Train A Valve 4701 or Train B Valve 4708.**	
Perform Step: 10√ Item #5 (Page 2 of 2)	CB-03 1-HS-4701, RCP CLR CCW RET ISOL VLV CLOSED	
Standard:	PLACED 1-HS-4701, RCP CLR CCW RET ISOL VLV in CLOSE position and VERIFIED green CLOSE light lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Perform Step: 11√ Item #6 (Page 2 of 2)	CB-03 1-HS-4708, RCP CLR CCW RET ISOL VLV CLOSED	
Standard:	PLACED 1-HS-4708, RCP CLR CCW RET ISOL VLV in CLOSE position and VERIFIED green CLOSE light lit.	
Comment:		**SAT** ☐ **UNSAT** ☐

Examiner Note:	**Close either <u>or</u> both Train A Valve 4699 or Train B Valve 4700.**
Perform Step: 12√ Item #9 (Page 2 of 2)	CB-03 1-HS-4699, RCP/THBR CLR CCW SPLY ISOL VLV CLOSED
Standard:	PLACED 1-HS-4699, RCP/THBR CLR CCW SPLY ISOL VLV in CLOSE position and VERIFIED green CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 13√ Item #7 (Page 2 of 2)	CB-03 1-HS-4700, RCP/THBR CLR CCW SPLY ISOL VLV CLOSED
Standard:	PLACED 1-HS-4700, RCP/THBR CLR CCW SPLY ISOL VLV in CLOSE position and VERIFIED green CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Close either <u>or</u> both Train A Valve 4696 or Train B Valve 4709.**
Perform Step: 14√ Item #10 (Page 2 of 2)	CB-03 1-HS-4696, THBR CLR CCW RET ISO VLV CLOSED
Standard:	PLACED 1-HS-4696, THBR CLR CCW RET ISO VLV in CLOSE position and VERIFIED green CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 15√ Item #8 (Page 2 of 2)	CB-03 1-HS-4709, THBR CLR CCW RET ISO VLV CLOSED
Standard:	PLACED 1-HS-4709, THBR CLR CCW RET ISO VLV in CLOSE position and VERIFIED green CLOSE light lit.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 has just tripped from 100% power.**
- **A Loss of Coolant Accident is in progress.**

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

- **VERIFY Containment Spray Not Required per EOP-0.0A, Reactor Trip or Safety Injection, Step 7.**

COMANCHE PEAK STEAM ELECTRIC STATION
UNIT 1
EMERGENCY RESPONSE GUIDELINES

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN 5 4-29-10 1200

____________/____________ **DMS CROSS VERIFICATION PERFORMED - WORKING COPY**
INITIAL & DATE

REACTOR TRIP OR SAFETY INJECTION

PROCEDURE NO. EOP-0.0A

REVISION NO. 8

EFFECTIVE DATE: 08/03/06 1200

PREPARED BY (Print): BART SMITH **EXT:** 8837

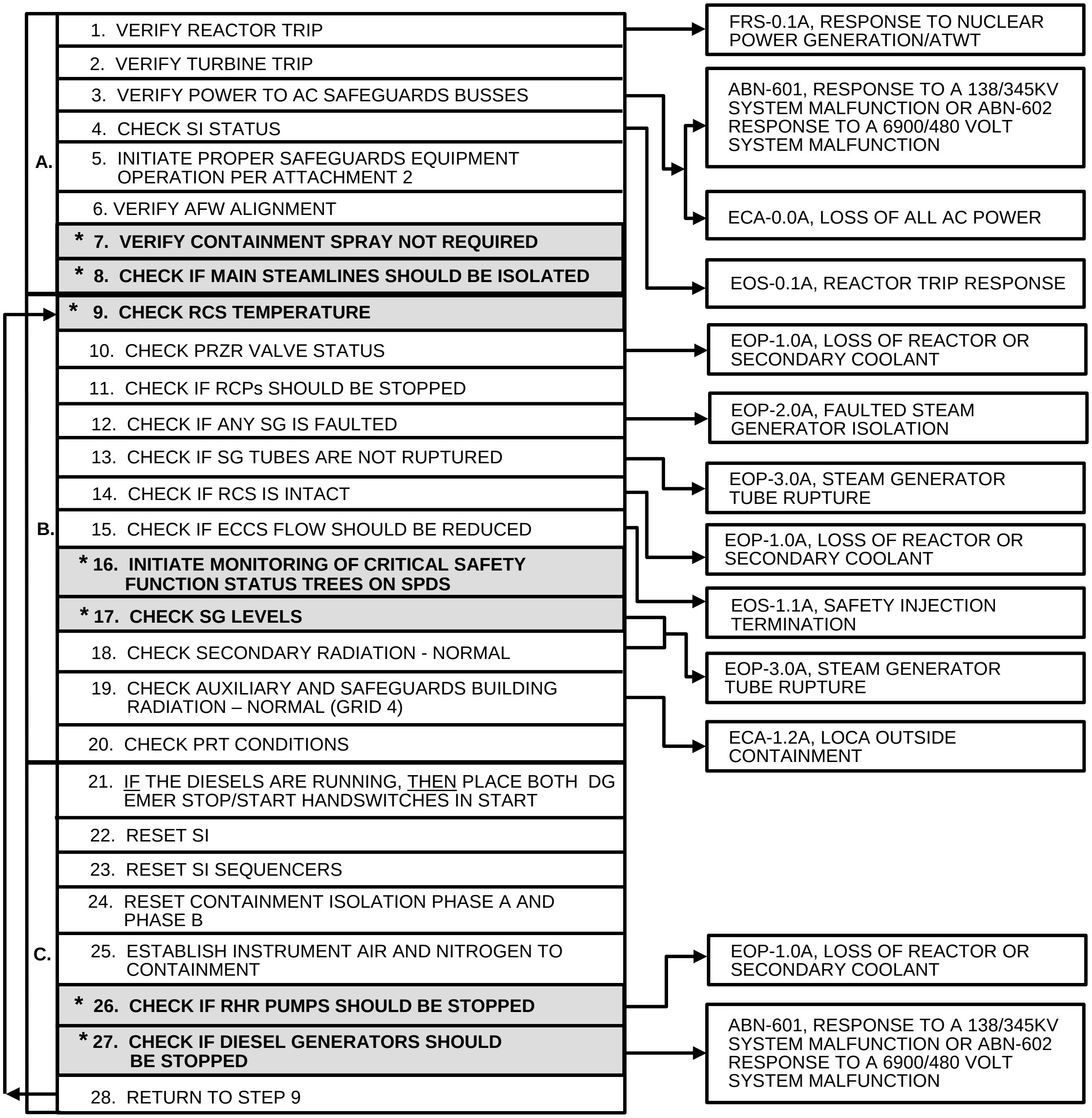
TECHNICAL REVIEW BY (Print): MIKE MANIS **EXT:** 5536

APPROVED BY: ____________________ **DATE:** 07/03/06
DIRECTOR, OPERATIONS

MAJOR ACTION CATEGORIES

- A. VERIFY AUTO ACTIONS
- B. IDENTIFY RECOVERY PROCEDURE
- C. SHUTDOWN UNNECESSARY EQUIPMENT AND CONTINUE TRYING TO IDENTIFY APPROPRIATE RECOVERY PROCEDURE

EOP-0.0A REV. 8 REACTOR TRIP OR SAFETY INJECTION



*** CONTINUOUS ACTION STEP**

A. PURPOSE

This procedure provides actions to verify proper response of automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery procedure.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2 and 3 GREATER THAN OR EQUAL TO 1000 PSIG. Using this procedure when not in these modes requires a step by step evaluation to determine if the required action is still applicable to current plant conditions.

C. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip:

- 2/4 Neutron Flux power ranges greater than 109%
- 2/4 Neutron Flux power ranges greater than 25% (Below P-10 permissive)
- 2/4 Neutron Flux rate trip lights as indicated on NIS cabinets (POSITIVE RATE TRIP)
- 1/2 Neutron Flux source ranges greater than 10^5 CPS (Below P-6 permissive)
- 1/2 Neutron Flux intermediate ranges greater than Amps approximately 25% (Above P-6 permissive and below P-10 permissive)
- 2/4 N-16 power exceed indicated Overtemperature N-16 setpoint
- 2/4 N-16 power greater than 112%
- Pressurizer pressures less than 1880 psig (Above P-7 permissive)
- 2/4 Pressurizer pressures greater than 2385 psig
- 2/3 Pressurizer water levels greater than 92% (Above P-7 permissive)
- 2/3 Reactor coolant loop flows less than 90% on 1/4 loops (Above P-8 permissive)
- 2/3 Reactor coolant loop flows less than 90% on 2/4 loops (Above P-7 and less than P-8 permissives)
- 2/4 Steam Generator levels less than 38% of Narrow Range on 1/4 steam generators
- 2/3 Turbine trip oil pressures less than 60 psig or 4/4 stop valves closed (Above P-9 permissives)
- Any Safety Injection signal
- Any First Out Annunciator lit
- Reactor trip logic met as indicated on the trip status logic bistable (TSLB) lights
- Both SSPS General Warning alarms in (1-ALB-6D, 1-5 and 2-5)

2) The following are symptoms of a reactor trip:

   a. Any reactor trip first out annunciator lit.
   b. Rapid decrease in neutron flux level.
   c. Shutdown and control rods inserted.

3) The following are symptoms that require a safety injection:

- 2/3 containment pressures greater than 3.0 psig.
- 2/4 pressurizer pressures less than 1820 psig.
- 2/3 steam line pressures less than 610 psig in any steam line.

4) The following are symptoms of a safety injection:
   a. SI annunciator lit (PCIP or First Out).
   b. ECCS pumps running.
   c. Diesel Generators running.
   d. Non-essential electrical power load shedding.

0700

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
①	Verify Reactor Trip:	
	a. Verify the following: • Reactor trip breakers - AT LEAST ONE OPEN -AND- • Neutron flux - DECREASING	a. Manually trip reactor from both trip switches. IF reactor will not trip, THEN momentarily de-energize 480V normal switchgear 1B3 AND 1B4. IF reactor NOT tripped, THEN go to FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT, Step 1.
	b. All control rod position rod bottom lights - ON	
②	Verify Turbine Trip: • All HP turbine stop valves - CLOSED	Manually trip turbine. IF the turbine will NOT trip, THEN pull-out all EHC fluid pumps. IF turbine still NOT tripped, THEN close or verify closed main steamline isolation valves.
③	Verify Power To AC Safeguards Busses:	
	a. AC safeguards busses - AT LEAST ONE ENERGIZED • AC safeguards bus voltage- 6900 Volts(6500-7100 Volts)	a. Go to ECA-0.0A, LOSS OF ALL AC POWER, Step 1.
	b. AC safeguards busses - BOTH ENERGIZED	b. Restore power to de-energized AC safeguards bus per ABN-601, RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION or ABN-602, RESPONSE TO A 6900/480 VOLT SYSTEM MALFUNCTION when time permits.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Check SI Status:	
	a. Check If SI Is Actuated: • SI actuation as indicated on the First Out Annunciator 1-ALB-6C • SI Actuated blue status light - ON	a. Check if SI is required: • Steam Line Pressure less than 610 psig. • Pressurizer Pressure less than 1820 psig. • Containment Pressure greater than 3.0 psig. <u>IF</u> SI is required, <u>THEN</u> manually actuate SI from either handswitch. <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> go to EOS-0.1A, REACTOR TRIP RESPONSE, Step 1.
	b. Verify Both Trains SI Actuated: • SI Actuated blue status light - ON <u>NOT</u> FLASHING	b. Manually Actuate SI.

CAUTION: A Safety Injection actuation will affect normal egress from the Containment Building. Attachment 9 of this procedure provides instructions to evacuate personnel from the Containment during a Safety Injection actuation.

NA

~~NOTE~~: Attachment 2 is required to be completed before FRGs are implemented.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Initiate Proper Safeguards Equipment Operation Per Attachment 2 0707	
~~6~~	Verify AFW Alignment:	
	a. MDAFW Pumps – RUNNING	a. Manually start pump(s).
	b. Turbine Driven AFW Pump – RUNNING IF NECESSARY	b. Manually open steam supply valve(s).
	c. AFW total flow – GREATER THAN 460 GPM	c. Check narrow range levels and perform the following: IF narrow range level greater than 43%(50% FOR ADVERSE CONTAINMENT) in any SG, THEN control feed flow to maintain narrow range level AND go to Step 6d. IF narrow range level less than 43%(50% FOR ADVERSE CONTAINMENT) in all SGs. THEN manually start pumps and align valves as necessary.
	d. AFW valve alignment – PROPER ALIGNMENT	d. Manually align valve(s) as necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 7	Verify Containment Spray Not Required:	
	a. Containment pressure – HAS REMAINED LESS THAN 18.0 PSIG • 1-ALB-2B window 1-8, CS ACT – NOT ILLUMINATED -AND- • 1-ALB-2B window 4-11, CNTMT ISOL PHASE B ACT – NOT ILLUMINATED -AND- • Containment Pressure – LESS THAN 18.0 PSIG	a. Perform the following: 1) Verify Containment Spray AND Phase B Actuation initiated. IF NOT, THEN manually actuate. 2) Verify appropriate MLB indication for CNTMT SPRAY (BLUE WINDOWS) AND PHASE B (ORANGE WINDOWS). IF valves NOT aligned, THEN manually align valve(s) as appropriate. (Refer to Attachment 6 as necessary). 3) Verify containment spray flow. 4) Ensure CHEM ADD TK DISCH VLVs – OPEN • 1-HS-4752 • 1-HS-4753 5) Stop all RCPs. 6) Go to Step 8.
	b. Verify containment spray heat exchanger out valves – CLOSED	b. Manually close valve(s).
	c. Verify containment spray pumps – RUNNING	c. Manually start pump(s).

ATTACHMENT 6
PAGE 1 OF 2

CONTAINMENT SPRAY/PHASE B ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	POSITION	ESFAS TRAIN	MLB LOCATION
☐ CB-02	1-HS-4754	CHEM ADD TK DISCH VLV	OPEN	A	1-MLB-4A3/1.9
☐ CB-02	1-HS-4776	CS HX 1 OUT VLV	OPEN	A	1-MLB-4A3/2.7
☐ CB-02	1-HS-4772-1	CSP 1 RECIRC VLV	CLOSED	A	1-MLB-4A3/1.6
☐ CB-02	1-HS-4772-2	CSP 3 RECIRC VLV	CLOSED	A	1-MLB-4A3/2.6
☐ CB-02	1-HS-4755	CHEM ADD TK DISCH VLV	OPEN	B	1-MLB-4B3/1.9
☐ CB-02	1-HS-4777	CS HX 2 OUT VLV	OPEN	B	1-MLB-4B3/2.7
☐ CB-02	1-HS-4773-1	CSP 2 RECIRC VLV	CLOSED	B	1-MLB-4B3/1.6
☐ CB-02	1-HS-4773-2	CSP 4 RECIRC VLV	CLOSED	B	1-MLB-4B3/2.6
☐ CB-03	1-HS-4514	SFGD LOOP CCW SPLY VLV	CLOSED	A	1-MLB-4A3/1.8
☐ CB-03	1-HS-4574	CS HX 1 CCW RET VLV	PARTIALLY OPEN, MLB LIT	A	1-MLB-4A3/3.7
☐ CB-03	1-HS-4572	RHR HX 1 CCW RET VLV	PARTIALLY OPEN, MLB LIT	A	1-MLB-4A3/1.7
☐ CB-03	1-HS-4512	SFGD LOOP CCW RET VLV	CLOSED	A	1-MLB-4A3/2.8
☐ CB-03	1-HS-4527	NON-SFGD LOOP CCW SPLY VLV	CLOSED	B	1-MLB-4B3/3.8
☐ CB-03	1-HS-4525	NON-SFGD LOOP CCW RET VLV	CLOSED	B	1-MLB-4B3/4.8
☐ CB-03	1-HS-4526	NON-SFGD LOOP CCW SPLY VLV	CLOSED	A	1-MLB-4A3/3.8
☐ CB-03	1-HS-4524	NON-SFGD LOOP CCW RET VLV	CLOSED	A	1-MLB-4A3/4.8

ATTACHMENT 6
PAGE 2 OF 2

CONTAINMENT SPRAY/PHASE B ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	POSITION	ESFAS TRAIN	MLB LOCATION
☐ CB-03	1-HS-4515	SFGD LOOP CCW SPLY VLV	CLOSED	B	1-MLB-4B3/1.8
☐ CB-03	1-HS-4575	CS HX 2 CCW RET VLV	PARTIALLY OPEN, MLB LIT	B	1-MLB-4B3/3.7
☐ CB-03	1-HS-4573	RHR HX 2 CCW RET VLV	PARTIALLY OPEN, MLB LIT	B	1-MLB-4B3/1.7
☐ CB-03	1-HS-4513	SFGD LOOP CCW RET VLV	CLOSED	B	1-MLB-4B3/2.8
☐ CB-03	1-HS-4701	RCP CLR CCW RET ISOL VLV	CLOSED	A	1-MLB-4A3/4.9
☐ CB-03	1-HS-4708	RCP CLR CCW RET ISOL VLV	CLOSED	B	1-MLB-4B3/4.9
☐ CB-03	1-HS-4700	RCP/THBR CLR CCW SPLY ISOL VLV	CLOSED	B	1-MLB-4B3/3.9
☐ CB-03	1-HS-4709	THBR CLR CCW RET ISOL VLV	CLOSED	B	1-MLB-4B3/2.9
☐ CB-03	1-HS-4699	RCP/THBR CLR CCW SPLY ISOL VLV	CLOSED	A	1-MLB-4A3/3.9
☐ CB-03	1-HS-4696	THBR CLR CCW RET ISOL VLV	CLOSED	A	1-MLB-4A3/2.9

☐ Notify Unit Supervisor attachment instructions complete <u>AND</u> identify Containment Spray/Phase B Isolation alignment status.

Facility: CPNPP JPM # NRC S-7 Task #RO4201 K/A #062.A4.07 3.1 / 3.1 SF-6

Title: Shift Normal Bus 1A4 Between Unit Auxiliary Transformer and Startup Transformer

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	X	Plant:	______

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:

- Unit 1 is at 100% power.
- Prerequisites for Normal Bus 1A4 have been met.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- SHIFT Normal Bus 1A4 between the Unit Auxiliary Transformer and the Startup Transformer per SOP-603A, 6900 V Switchgear, Step 5.3.2, Transferring a 6.9 KV Normal Bus from Unit 1 Auxiliary Transformer 1UT to Station Service Transformer 1ST.

Task Standard: Shift Normal Bus 1A4 between the Unit Auxiliary Transformer and the Startup Transformer and perform actions when Incoming Breaker 1A4-1 fails to trip during Bus transfer per SOP-603A.

Required Materials: SOP-603A, 6900 V Switchgear, Rev. 14-9.

ALM-0102A, 1-ALB-10B, Window 3.4 - 6.9 KV ANY NON-1E BUS PARALLELED, Rev. 11-4.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-18 or any at power Initial Condition and PERFORM the following:

- **When directed, EXECUTE malfunction ED20L, Breaker 1A4-1 interlock failure.**

BOOTH OPERATOR NOTE:

- **After each JPM, PERFORM the following:**
  - **MOVE the Synchroscope Switch to an alternate 6900 V Bus position.**
  - **PLACE VS-1A, 6.9 KV BUS VOLT / FREQ SELECT switch in the 1A1 position.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **SOP-603A, 6900 V Switchgear.**
  - **Step 5.3.2, Transferring a 6.9 KV Normal Bus from Unit 1 Auxiliary Transformer 1UT to Station Service Transformer 1ST.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from SOP-603A, Step 5.3.2.**
Perform Step: 1 5.3.2.A & 4th bullet	Ensure the prerequisites in Section 2.3 are met for the selected bus. • 6.9 KV SWITCHGEAR 1A4
Standard:	DETERMINED Prerequisites have been met for 6900 V Switchgear 1A4 per the Initial Conditions.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√ 5.3.2.B & 4th bullet	Turn synchroscope ON for the selected Bus Feeder Breaker AND ensure proper phasing and frequency. • SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE
Standard:	PLACED synchroscope switch into SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE position and TURNED to ON and OBSERVED proper phasing and frequency.
Comment:	**SAT ☐ UNSAT ☐**

NOTE: Closing an incoming feeder breaker will cause the other incoming feeder breaker for the bus to automatically trip open.

Perform Step: 3√ 5.3.2.C & 4th bullet	Close the incoming breaker from Station Service Transformer 1ST to the desired bus. • CS-1A4-2, INCOMING BKR 1A4-2
Standard:	PLACED CS-1A4-2, INCOMING BKR 1A4-2 in CLOSE and OBSERVED red CLOSE light lit.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps represent the Alternate Path of this JPM.**
Perform Step: 4 5.3.2.D & 4th bullet	Ensure the feeder breaker from Unit Auxiliary Transformer 1UT to the bus being transferred trips open. • CS-1A4-1, INCOMING BKR 1A4-1
Standard:	OBSERVED CS-1A4-1, INCOMING BKR 1A4-1 red CLOSE light lit and DETERMINED breaker did NOT OPEN.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following NOTE is from 1-ALB-10B, Window 3.4.**
NOTE:	This alarm occurs when both preferred and alternate power supply breakers are closed for more than one-half second. If anti-paralleling interlock relay is disabled, offgoing feeder breaker will fail to trip within one second. Manual operator action will be required to open appropriate feeder breaker.
Perform Step: 5	Acknowledge annunciator alarm.
Standard:	ACKNOWLEDGED annunciator alarm 1-ALB-10B, Window 3.4 - 6.9 KV ANY NON-1E BUS PARALLELED.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**This action is considered skill of the craft and, if referenced, is found in the Alarm Response Procedure.**
Perform Step: 6√ 5.3.2.D & 4th bullet	Ensure the feeder breaker from Unit Auxiliary Transformer 1UT to the bus being transferred trips open. • CS-1A4-1, INCOMING BKR 1A4-1
Standard:	PLACED CS-1A4-1, INCOMING BKR 1A4-1 in TRIP and OBSERVED green TRIP light lit.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 7 5.3.2.E & 5.3.2.E.1)	Check transferred bus voltage normal by performing the following: • Position VS-1A, 6.9 KV BUS VOLT/FREQ SELECT to the desired bus.
Standard:	PLACED VS-1A, 6.9 KV BUS VOLT/FREQ SELECT to BUS 1A4 position.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 8 5.3.2.E & 5.3.2.E.2)	Check transferred bus voltage normal by performing the following: • Verify V-1A, 6.9 KV NON-SFGD BUS VOLT approximately 6900 VOLTS, for the selected bus (6450-7150 volts required).
Standard:	OBSERVED V-1A, 6.9 KV NON-SFGD BUS VOLT approximately 6900 volts for BUS 1A4.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 9 5.3.2.F & 4th bullet	Match handswitch target by placing the selected breaker, for the bus transferred, in NEUTRAL-AFTER-TRIP. • CS-1A4-1, INCOMING BKR 1A4-1
Standard:	DETERMINED CS-1A4-1, INCOMING BKR 1A4-1 is in NEUTRAL-AFTER-TRIP position with green FLAG indicating.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 10 5.3.2.G & 4th bullet	Turn synchroscope OFF for the selected breaker. • SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE
Standard:	TURNED SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE to OFF.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is at 100% power.**
- **Prerequisites for Normal Bus 1A4 have been met.**

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

- **SHIFT Normal Bus 1A4 between the Unit Auxiliary Transformer and the Startup Transformer per SOP-603A, 6900 V Switchgear, Step 5.3.2, Transferring a 6.9 KV Normal Bus from Unit 1 Auxiliary Transformer 1UT to Station Service Transformer 1ST.**

COMANCHE PEAK STEAM ELECTRIC STATION

UNIT 1

SYSTEM OPERATING PROCEDURE MANUAL

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__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

6900 V SWITCHGEAR

PROCEDURE NO. SOP-603A

REVISION NO. 14

EFFECTIVE DATE: 3/22/05 1200

PREPARED BY (Print): Steven Lewis Ext: 6524

TECHNICAL REVIEW BY (Print): Allan Glass Ext: 5145

APPROVED BY: Alan Hall for R A Smith Date: 3/15/05
DIRECTOR, OPERATIONS

1.0 APPLICABILITY

This procedure provides instructions for energizing, de-energizing and transferring the electrical line up of the 6.9 KV electrical switchgear.

2.0 PREREQUISITES

2.1 Energizing a 6.9 KV Bus

N/A ☐ A. As applicable, Transformer XST1, XST2, XST1/2 (XST2A), 1UT, 1ST or 2ST energized.

B. Ensure the applicable control power lineup for the bus being energized has been completed:

N/A ☐ ● SOP-603A-EPA-E01, 6.9 KV Safeguards Bus Control Power Lineup

☐ ● SOP-603A-EPA-E02, 6.9 KV Normal and Common Bus Control Power Lineup

☐ C. Ensure that the white supervisory light on the bus being energized is illuminated.

2.2 Deenergizing a 6.9 KV Bus

☐ ● Prior to deenergizing the bus, ensure all loads necessary for plant operation have been transferred to an alternate source or the function is supplied by alternate equipment.

2.3 Normal Operation of a 6.9 KV Bus

☐ ● Prior to transferring the bus to an alternate source, ensure the alternate source is energized and capable of handling the load of the associated bus.

3.0 PRECAUTIONS

- All electrical switching will be done under the direction of the Shift Manager or his designated representative.
- Observe all normal precautions pertaining to energized electrical gear.
- Open all load feeder breakers prior to energizing or deenergizing electrical buses.
- Check the bus voltage before and after a bus is energized.
- When energizing a dead 6.9 KV Normal Bus, the alternate feeder breaker (the -2, Incoming Breaker) will close automatically when the breaker handswitch is placed in the NEUTRAL position from the PULLOUT position, if the unit auxiliary/preferred power supply breaker is RACKED IN.
- The 6.9 KV safeguards bus tie breakers must be closed at all times for proper operation.
- All 6.9 KV breaker racking operations shall be performed in accordance with Attachment 1, “Guidelines on Proper Operation of 6.9 KV Breakers”.

4.0 LIMITATIONS/NOTES

4.1 Limitations

- The Loss of Power Diesel Generator Start Instrumentation for each Function in TS Table 3.3.5-1, including the 6.9 KV Class 1E bus undervoltage and degraded voltage instrumentation, shall be OPERABLE in MODEs 1, 2, 3, & 4 per TS 3.3.5. The response times for this instrumentation shall be OPERABLE per TR 13.3.5.

- Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE in MODEs 1, 2, 3, and 4 per TS 3.8.9. (also reference Table B 3.8.9-1)

- The necessary portion of the Train A or Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support one train of equipment required to be OPERABLE in MODEs 5 and 6 per TS 3.8.10. (also reference Table B 3.8.9-1)

4.2 Notes

- None

5.3.2 Transferring a 6.9 KV Normal Bus from Unit 1 Auxiliary Transformer 1UT to Station Service Transformer 1ST

This section describes the steps required to transfer a 6.9 KV normal bus from Unit 1 Auxiliary Transformer 1UT to Station Service Transformer 1ST.

A. Ensure the prerequisites in Section 2.3 are met for the selected bus.

N/A
- ☒ 6.9 KV SWITCHGEAR 1A1
- ☐ 6.9 KV SWITCHGEAR 1A2
- ☒ 6.9 KV SWITCHGEAR 1A3
- ☐ 6.9 KV SWITCHGEAR 1A4

B. Turn synchroscope ON for the selected Bus Feeder Breaker AND ensure proper phasing and frequency.

N/A
- ☒ SS-1A1-2, BKR 1A1-2 SYNCHROSCOPE
- ☐ SS-1A2-2, BKR 1A2-2 SYNCHROSCOPE
- ☒ SS-1A3-2, BKR 1A3-2 SYNCHROSCOPE
- ☐ SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE

> NOTE: Closing an incoming feeder breaker will cause the other incoming breaker for the bus to automatically trip open.

C. Close the incoming breaker from Station Service Transformer 1ST to the desired bus.

N/A
- ☒ CS-1A1-2, INCOMING BKR 1A1-2
- ☐ CS-1A2-2, INCOMING BKR 1A2-2
- ☒ CS-1A3-2, INCOMING BKR 1A3-2
- ☐ CS-1A4-2, INCOMING BKR 1A4-2

D. Ensure the feeder breaker from Unit Auxiliary Transformer 1UT to the bus being transferred trips open.

N/A
- ☐ CS-1A1-1, INCOMING BKR 1A1-1
- ☐ CS-1A2-1, INCOMING BKR 1A2-1
- ☒ CS-1A3-1, INCOMING BKR 1A3-1
- ☐ CS-1A4-1, INCOMING BKR 1A4-1

CPSES SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-603A
6900 V SWITCHGEAR	REVISION NO. 14	PAGE 31 OF 51

5.3.2 E. Check transferred bus voltage normal by performing the following:

☐ 1) Position VS-1A, 6.9 KV BUS VOLT/FREQ SELECT to the desired bus.

☐ 2) Verify V-1A, 6.9 KV NON-SFGD BUS VOLT approximately 6900 VOLTS, for the selected bus (6450-7150 volts required).

F. Match handswitch target by placing the selected breaker, for the bus transferred, in NEUTRAL-AFTER-TRIP.

N/A ☐ ● CS-1A1-1, INCOMING BKR 1A1-1

☐ ● CS-1A2-1, INCOMING BKR 1A2-1

☑ ● CS-1A3-1, INCOMING BKR 1A3-1

☐ ● CS-1A4-1, INCOMING BKR 1A4-1

G. Turn synchroscope OFF for the selected breaker.

N/A ☐ ● SS-1A1-2, BKR 1A1-2 SYNCHROSCOPE

☐ ● SS-1A2-2, BKR 1A2-2 SYNCHROSCOPE

☑ ● SS-1A3-2, BKR 1A3-2 SYNCHROSCOPE

☐ ● SS-1A4-2, BKR 1A4-2 SYNCHROSCOPE

COMMENTS: ____

____

Facility: CPNPP JPM # NRC S-8 Task #RO3603 K/A #008.A4.10 3.1 / 3.1 SF-8

Title: Remove Train A Component Cooling Water Safeguards Loop from Service

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	______	Classroom:	______
Actual Performance:	X	Simulator:	X
Alternate Path:	______	Plant:	______

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:

- Unit 1 is at 100% power with all controls in AUTOMATIC.
- Train A Component Cooling Water Pump (CCW) is in service and Train B Component Cooling Water Loop is in Standby.
- An operator is standing by at the Train B Component Cooling Water Pump.
- Chemistry has been notified that CCW alignment changes are underway.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- REMOVE the Train A Component Cooling Water Safeguards Loop from service, LEAVE Train A Component Cooling Water Pump in operation, and PLACE Train B CCW Safeguards Loop in service per SOP-502A, Component Cooling Water System, Section 5.3.2, Removal/Restoration of Train A Safeguards Loop from Service, START at Step 5.3.2.1.C.

Task Standard: Start the Train B CCW Pump and remove the Train A CCW Safeguards Loop from service per SOP-502A.

Required Materials: SOP-502A, Component Cooling Water System, Rev. 18-11.

Validation Time: 15 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ______________

SIMULATOR SETUP

BOOTH OPERATOR:

INITIALIZE to IC-18 or any at power Initial Condition and PERFORM the following:

- **ENSURE Train A Component Cooling Water Loop is in service.**

EXAMINER:

PROVIDE the examinee with a copy of Procedure 1:

- **SOP-502A, Component Cooling Water System.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**The following steps are from SOP-502A, Step 5.3.2.**
Perform Step: 1 5.3.2.1.C	IF Train B is to be placed in service, THEN Start Train B CCW Pump per Section 5.2.1.
Standard:	STARTED Train B CCW Pump per Section 5.2.1.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The following steps are from SOP-502A, Step 5.2.1.**
Perform Step: 2 5.2.1.1.A & 2nd bullet	Ensure the Station Service Water Pump, associated with the CCW Pump to be started is operating. • SSWP 2
Standard:	DETERMINED Station Service Water Pump 1-02 is RUNNING.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3 5.2.1.1.B & 2nd bullet	Ensure the oil level in the bearing housings are normal.
Standard:	CONTACTED NEO to ENSURE the oil level in the bearing housings is normal.
Booth Operator:	**When contacted, REPORT oil level in bearing housing is normal.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Step may be performed on Train A; however, Train B is desired.**
Perform Step: 4 5.2.1.1.C Train B & 1st bullet	IF CCW heat load is low, THEN additional CCW flow should be established through the CS HX or RHR HX prior to starting the second pump. Train B • 1-HS-4575, CS HX 2 CCW RET VLV
Standard:	OPENED 1-HS-4575, CS HX 2 CCW RET VLV to establish flow through the Containment Spray Heat Exchanger.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Step may be performed on Train A; however, Train B is desired.**
Perform Step: 5 5.2.1.1.C Train B & 2nd bullet	IF CCW heat load is low, THEN additional CCW flow should be established through the CS HX or RHR HX prior to starting the second pump. Train B • 1-HS-4573, RHR HX 2 CCW RET VLV
Standard:	OPENED 1-HS-4573, RHR HX 2 CCW RET VLV to establish flow through the RHR Heat Exchanger.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 6√ 5.2.1.1.D & 2nd bullet	Start the idle CCW Pump. • 1-HS-4519A, CCWP 2
Standard:	PLACED 1-HS-4519A, CCWP 2 in START and OBSERVED: • Red FAN and PUMP lights lit. • 1-PI-4521, CCWP 2 DISCH PRESS rising. • 1-FI-4537A, CCW HX 2 OUT FLO rising. • 1-FI-4537B, CCW HX 2 RECIRC FLO rising.
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The examinee may perform either or both of the following 2 steps.**
Examiner Note:	**The following steps are from SOP-502A, Step 5.3.2.**
CAUTION:	For continued operation with the safeguards loop isolated a flowpath through the RHR or CS Heat Exchangers should be established to provide adequate flow to the CCW heat exchanger for heat transfer and to provide for CCW Pump protection.
Perform Step: 7√ 5.3.2.1.E.1) & 1st bullet	IF Train A CCW Pump is to continue operation with the loops isolated, THEN perform the following: • Throttle Open the return valve for the heat exchanger to establish a safeguards loop flowpath through the RHR or CS HX. • 1-HS-4572, RHR HX 1 CCW RET VLV
Standard:	THROTTLED OPEN 1-HS-4572, RHR HX 1 CCW RET VLV and OBSERVED flow on 1-FI-4556, RHR HX 1 CCW RET FLO rising.
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 8√ 5.3.2.1. E.1) & 2nd bullet	IF Train A CCW Pump is to continue operation with the loops isolated, THEN perform the following: • Throttle Open the return valve for the heat exchanger to establish a safeguards loop flowpath through the RHR or CS HX. • 1-HS-4574, CS HX 1 CCW RET VLV
Standard:	THROTTLED OPEN 1-HS-4574, CS HX 1 CCW RET VLV and OBSERVED flow on 1-FI-4560, CS HX 1 CCW RET FLO rising.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 9√ 5.3.2.1.E.2) & 1st bullet	Close the following to isolate Train A Safeguards Loop: • 1-HS-4514, SFGD LOOP CCW SPLY VLV
Standard:	PLACED 1-HS-4514, SFGD LOOP CCW SPLY VLV in CLOSE and OBSERVED green CLOSE light illuminated.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 10√ 5.3.2.1.E.2) & 2nd bullet	Close the following to isolate Train A Safeguards Loop: • 1-HS-4512, SFGD LOOP CCW RET VLV
Standard:	PLACED 1-HS-4512, SFGD LOOP CCW RET VLV in CLOSE and OBSERVED green CLOSE light illuminated.
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 11 5.3.2.1.E.3)	Verify proper operation of 1-HS-4536, CCWP 1 RECIRC VLV.
Standard:	VERIFIED 1-HS-4536, CCWP 1 RECIRC VLV is OPEN if CCW Heat Exchanger outlet flow drops below 8200 gpm.
Terminating Cue:	**This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is at 100% power with all controls in AUTOMATIC.**
- **Train A Component Cooling Water Pump (CCW) is in service and Train B Component Cooling Water Loop is in Standby.**
- **An operator is standing by at the Train B Component Cooling Water Pump.**
- **Chemistry has been notified that CCW alignment changes are underway.**

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

- **REMOVE the Train A Component Cooling Water Safeguards Loop from service, LEAVE Train A Component Cooling Water Pump in operation, and PLACE Train B CCW Safeguards Loop in service per SOP-502A, Component Cooling Water System, Section 5.3.2, Removal/Restoration of Train A Safeguards Loop from Service, START at Step 5.3.2.1.C.**

COMANCHE PEAK STEAM ELECTRIC STATION

UNIT 1 and COMMON

SYSTEM OPERATING PROCEDURE MANUAL

ELECTRONIC CONTROLLED COPY

CHANGES ARE NOT INDICATED

LATEST CHANGE NOTICE EFFECTIVE DATE PCN 11 9/29/10 1200

__________/________ Verify current status in the Document Control Database prior to use.
INITIAL & DATE

QUALITY RELATED

COMPONENT COOLING WATER SYSTEM

PROCEDURE NO. SOP-502A

REVISION NO. 18

EFFECTIVE DATE: 11/18/05 1200

PREPARED BY (Print): Juannelle Miller Ext: 5835

TECHNICAL REVIEW BY (Print): Steven Lewis Ext: 6524

APPROVED BY: Jim Brau for R.A. Smith Date: 11/14/05
DIRECTOR, OPERATIONS

2.3 Filling and Venting

2.3.1 Safeguards Loop Train A

N/A ☐ ● The Demineralized Water System is available for CCW Surge Tank Makeup.

● CCW Drain Tank valves have been aligned per the following:

- N/A ☐ ● SOP-502A-CC-V05, “Valve Lineup - CCW Drains (part 1)”
- ☐ ● SOP-502A-CC-V06, “Valve Lineup - CCW Drains (part 2)”

2.3.2 Safeguards Loop Train B

☐ ● The Demineralized Water System is available for CCW Surge Tank Makeup.

● CCW Drain Tank valves have been aligned per the following:

- ☐ ● SOP-502A-CC-V05, “Valve Lineup - CCW Drains (part 1)”
- ☐ ● SOP-502A-CC-V06, “Valve Lineup - CCW Drains (part 2)”

2.3.3 Non-Safeguards Loop Outside the Containment Building

☐ ● Sections 5.4.1, 5.4.2, or 5.4.5 have been completed.

2.3.4 Non-Safeguards Loop Inside the Containment Building

☐ ● Section 5.4.3 has been completed OR is in progress.

3.0 PRECAUTIONS

- The CCW System contains hydrazine. Protective clothing for the face and hands should be worn when working with this corrosion inhibitor.
- All releases from the CCW System should be coordinated with Chemistry and Environmental to ensure compliance with State and Federal regulatory requirements.
- The symbol [IV] has been located throughout this procedure to identify those steps requiring Verification. Initial performance and Verification of these steps shall be documented on the Verification Log Sheet (STA-694-1).
- The following Forms may be utilized to verify the valve, control switch and electrical lineups with the system in operation.
  - SOP-502A-CC-V01
  - SOP-502A-CC-V02
  - SOP-502A-CC-V03
  - SOP-502A-CC-V04
  - SOP-502A-CC-V05
  - SOP-502A-CC-V06
  - SOP-502A-CC-C01
  - SOP-502A-CC-E01
  - SOP-502A-CC-E02

3.0 PRECAUTIONS (continued)

- Demineralized water should be used as the source of makeup to the CCW Surge Tank when filling and venting the CCW System.

- All drainage from the CCW System should be directed to the CCW Drain System or to a sump which pumps directly to LVW.

- The CCW pumps will automatically start from the following signals, if the pump control switches are in AUTO:

  Safety Injection sequence signal

  Blackout sequence signal

  Low CCW pressure at the opposite train CCW heat exchanger outlet

  An AUTO start of the associated train SSW pump on low pressure in the alternate SSW train.

- Starting a CCW pump will automatically start the following equipment, if their control switches are in AUTO:

  Associated CCW pump room fan cooler

  Associated SSW pump

  Associated Safety Chilled Water recirc pump

- Air pockets can form in isolated portions during fill and vent operation. Caution should be exercised when filling the surge tank due to potential for CCW pump surge tank overflow when the CCW pump is stopped and the compressed air pockets expand.

- To prevent Chloride infusion if a tube leak exists, the CCW HX Shell side should be filled, vented and pressurized prior to operating SSW OR the CCW HX shell side should be isolated and drained with the drain valves open.

- Starting a second CCW pump or isolation of a large load may increase flow to the vent chillers causing 1-FV-4650A and 1-FV-4650B isolation, if flow remains high for 30 seconds.

- Isolating a large load or several smaller loads concurrently, while running both CCW pumps, may spike system pressure enough to cause lifting of relief valves in the affected loop.

- The Unit 1 and Unit 2 CCW supply and return isolation valves to all common equipment are required to be LOCKED CLOSED per OWI-103 and the valve lineups for this procedure. To provide cooling to a common piece of equipment the applicable Unit (1 OR 2) CCW isolation valves may be deviated with the Shift Manager's permission.

- Both units CCW Surge Tanks should be monitored during system draining.

3.0 PRECAUTIONS (continued)

- The Reactor Coolant Pump Upper Bearing Lube Oil Coolers should be drained and dried when non-flowing water is expected to remain in the coolers for periods longer than two months, as this can be detrimental to the tubing material. Instructions to drain and dry the Reactor Coolant Pump Upper Bearing Lube Oil Coolers are contained in a subsection of “Section 5.5, Draining”.
- Containment penetrations MV-0003 and MV-0004 are used for CCW flow to the RCDT and Excess Letdown heat exchangers. Thermal relief protection of these penetrations is provided through locked open valves 1CC-0610, 1CC-0613, 1CC-0616, and 1CC-0619.

4.0 LIMITATIONS AND NOTES

4.1 Limitations

- CCW Pump bearing temperature exceeding 200°F requires the CCW Pump be stopped.
- Two CCW trains shall be OPERABLE in MODES 1, 2, 3 and 4 (TS 3.7.7)
- Flow rates on the CCW System of 17,500 gpm per CCW pump should not be exceeded.
- A CCW Pump Motor may have two (2) starting attempts from ambient temperature. At least a 45 minute standing period should be observed between any additional attempts.
- A CCW Pump motor may have one (1) immediate restart attempt from operating temperature. At least a 15 minute running period should be observed between any additional restart attempts.
- Normal flow rates for CCW supplied loads are listed in Attachment 1, “Normal CCW Flows”.
- CCW System relief valves are listed in Attachment 2, “CCW Relief Valves”.
- The maximum flow rate for CCW flow through the CCW Filter Demineralizer Skid is 80 gpm. CCW flow through the skid is controlled at approximately 50 gpm in accordance with COP-502A, "Component Cooling Water."
- CCW supply should not be isolated to operating equipment.
- The RHR and Containment Spray Heat Exchanger supply manual butterfly valves (flow restricting orifices) are normally locked closed to provided acceptable CCW flow balancing for Design Basis Accidents to limit heat addition to CCW. These valves are not required to be opened to mitigate a Design Basis Accident (e.g. LOCA); however, they may be opened to accelerate cooldown after accident heat loads have sufficiently decayed if desired and if the valve locations are accessible. Therefore, 1CC-0109 and 1CC-0157, which are closed to provide a flow limiting function in MODES 1, 2 and 3, may be opened in MODE 3, at or below 400 °F, as needed to support RHR cooldown in MODE 4, 5 and 6. Manual Valves 1CC-0107 and 1CC-0158, which provide a flow limiting function in MODES 1, 2 and 3 may be open in MODE 4, 5, and 6.

4.2 Notes

None

CPSES SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-502A
COMPONENT COOLING WATER SYSTEM	REVISION NO. 18	PAGE 14 OF 176

5.2 Normal Operation

5.2.1 Operating CCW Pumps

This section describes the steps to start up a standby CCW Pump for alternating or dual pump operation and place a running CCW pump in standby as required.

5.2.1.1 Starting a Standby CCW Pump During Normal Operation.

NOTE: Starting a CCW Pump will automatically start the following equipment, if their control switches are in AUTO:

- Associated CCW Pump room fan cooler
- Associated SSW Pump
- Associated Safety Chilled Water Recirc Pump

A. Ensure the Station Service Water Pump, associated with the CCW Pump to be started is operating.

☐ ● SSWP 1

☐ ● SSWP 2

B. Ensure the oil level in the bearing housings are normal.

☐ ● CCWP 1

☐ ● CCWP 2

<u>NOTE</u>:

- The following step may be required to limit CCW System pressure and prevent relief valve operation when two CCW Pumps are running.
- Low flow alarms are provided for both CT & RHR Heat Exchanger flow. These alarms may or may not occur as flow is started and stopped, dependent on time spent at or near the flow setpoint. This is a normal occurrence.

5.2.1.1 C. <u>IF</u> CCW heat load is low, <u>THEN</u> additional CCW flow should be established through the CS HX or RHR HX prior to starting the second pump.

<u>TRAIN A</u>

☐ ● 1-HS-4574, CS HX 1 CCW RET VLV

☐ ● 1-HS-4572, RHR HX 1 CCW RET VLV

<u>TRAIN B</u>

☐ ● 1-HS-4575, CS HX 2 CCW RET VLV

☐ ● 1-HS-4573, RHR HX 2 CCW RET VLV

<u>NOTE</u>: The following indications are available on the Plant computer.

		<u>ALARM</u>
T2740A	CCWP 1 INBD RDL BRG TEMP	185°F
T2741A	CCWP 1 OUTBD RDL BRG TEMP	185°F
T2742A	CCWP 1 ACTIVE FACE THR BRG TEMP	185°F
T2744A	CCWP 1 MOT INBD BRG TEMP	185°F
T2745A	CCWP 1 MOT OUTBD BRG TEMP	185°F
T2746A	CCWP 1 MOT STAT PHASE A TEMP	236°F
T2747A	CCWP 1 MOT STAT PHASE B TEMP	236°F
T2748A	CCWP 1 MOT STAT PHASE C TEMP	236°F
T2760A	CCWP 2 INBD RDL BRG TEMP	185°F
T2761A	CCWP 2 OUTBD RDL BRG TEMP	185°F
T2762A	CCWP 2 ACTIVE FACE THR BRG TEMP	185°F
T2764A	CCWP 2 MOT INBD BRG TEMP	185°F
T2765A	CCWP 2 MOT OUTBD BRG TEMP	185°F
T2766A	CCWP 2 MOT STAT PHASE A TEMP	236°F
T2767A	CCWP 2 MOT STAT PHASE B TEMP	236°F
T2768A	CCWP 2 MOT STAT PHASE C TEMP	236°F

D. Start the idle CCW Pump.

☐ N/A ● 1-HS-4518A, CCWP 1

☐ ● 1-HS-4519A, CCWP 2

> <u>NOTE</u>: Low flow alarms are provided for both CT & RHR Heat Exchanger flow. These alarms may or may not occur as flow is started and stopped, dependent on time spent at or near the flow setpoint. This is a normal occurrence.

5.2.1.1 E. <u>IF</u> the CCW PUMPS are being alternated for their bi-weekly rotation per OWI-409 "EQUIPMENT ROTATION PROGRAM", <u>THEN</u> momentarily initiate flow through each RHR and CS heat exchanger while <u>BOTH</u> pumps are in service.

<u>TRAIN A</u>

NA

- ☐ ● 1-HS-4574, CS HX 1 CCW RET VLV
- ☐ ● 1-HS-4572, RHR HX 1 CCW RET VLV

<u>TRAIN B</u>

- ☐ ● 1-HS-4575, CS HX 2 CCW RET VLV
- ☐ ● 1-HS-4573, RHR HX 2 CCW RET VLV

COMMENTS: ______________________________

____

____

5.3.2 Removal / Restoration of Train A Safeguards Loop from Service

These sections describe the steps to isolate Train A Safeguards Loop of CCW, AND to restore the Loop to service.

5.3.2.1 Removal of Train A Safeguards Loop from Service (Train A CCW Pump OPERATING OR SHUTDOWN)

This section describes the steps to isolate Train A Safeguards Loop of CCW. This section allows the loop to be isolated with the CCW Pump in operation, or with the CCW Pump shutdown.

☑ A. Notify Chemistry that CCW system alignment changes may require adjustment to the CCW hydrazine injection rate per COP-502A.

B. IF Train A CCW Pump is to be stopped, THEN ensure the following equipment has been removed from service OR supplied by Unit 2 where applicable:

N/A
- ☐ ● RHR Pump 1-01
- ☐ ● CS Pump 1-01
- ☐ ● CS Pump 1-03
- ☐ ● Safety Chiller 1-05
- ☐ ● UPS A\C X-01
- ☐ ● Control Room A\C Unit X-01
- ☐ ● Control Room A\C Unit X-02

☐ C. IF Train B is to be placed in service, THEN Start Train B CCW Pump per Section 5.2.1.

> **CAUTION:** To prevent Chloride infusion if a tube leak exists, the CCW HX shell side should be filled, vented and pressurized OR the CCW HX should be isolated and drained within 30 minutes of depressurizing the safeguard loop. To meet the intent of this CAUTION, CCW pressure should be greater than SSW pressure to prevent lake water from leaking into CCW.

D. IF the Train A CCW Pump is to be stopped, THEN perform the following:

N/A
- ☐ 1) Stop Train A CCW Pump 1-HS-4518A, CCWP 1, AND place the handswitch in PULL OUT.
- ☐ 2) Isolate Service Water flow to the Train A CCW Heat Exchanger per SOP-501A, Station Service Water System OR prepare to isolate and drain the CCW shell side of the CCW heat exchanger.

5.3.2.1 D 3) Close the following to isolate Train A Safeguards Loop:

N/A

☐ ● 1-HS-4514, SFGD LOOP CCW SPLY VLV

☐ ● 1-HS-4512, SFGD LOOP CCW RET VLV

☐ ● 1-HS-4536, CCWP 1 RECIRC VLV

4) IF the CCW heat exchanger is depressurized AND SSW flow is aligned to the heat exchanger, THEN perform the following:

☐ ● Isolate the CCW shell side of the heat exchanger, AND Open and Tag the CCW drain valve(s).

☐ ● Tag Train A CCW Pump 1-HS-4518A, CCWP 1, handswitch.

E. IF Train A CCW Pump is to continue operation with the loops isolated, THEN perform the following:

CAUTION: For continued operation with the safeguards loop isolated a flowpath through the RHR or CS Heat Exchangers should be established to provide adequate flow to the CCW heat exchanger for heat transfer and to provide for CCW Pump protection.

1) Throttle Open the return valve for the heat exchanger to establish a safeguards loop flowpath through the RHR or CS HX.

☐ ● 1-HS-4572, RHR HX 1 CCW RET VLV

☐ ● 1-HS-4574, CS HX 1 CCW RET VLV

2) Close the following to isolate Train A Safeguards Loop:

☐ ● 1-HS-4514, SFGD LOOP CCW SPLY VLV

☐ ● 1-HS-4512, SFGD LOOP CCW RET VLV

NOTE: 1-HS-4536, CCWP 1 RECIRC VLV should open on low CCW heat exchanger outlet flow of approximately < 8,200 gpm with the CCW Pump breaker closed.

☐ 3) Verify proper operation of 1-HS-4536, CCWP 1 RECIRC VLV.

> <u>NOTE</u>: Isolating the Train A Safeguards Loop may cause an alarm at the PC-11 console for 1-RE-4509 (CCW167) due to OPERATE FAILURE - LOSS OF SAMPLE FLOW.

5.3.2.1 F. <u>IF</u> desired to remove 1-RE-4509, UNIT 1 COMPONENT COOLING WATER TRAIN A SFGD LOOP RADIATION DETECTOR from service, <u>THEN</u> perform the following:

N/A

☐ 1) <u>IF</u> desired, initiate LCOAR for ODCM 3.3.3.4.

☐ 2) At the PC-11 console, place CCW167 (1-RE-4509) in OFF.

COMMENTS: ____

____

____

Facility: CPNPP JPM # NRC P-1-U1 Task #AO3005 K/A #039.A2.01 3.1 / 3.2 SF-4S

Title: Locally Control Steam Generator Atmospheric Relief Valve

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	X	Classroom:	______
Actual Performance:	______	Simulator:	______
Alternate Path:	______	Plant:	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:

- A Loss of Instrument Air has occurred on Unit 1.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- LOCALLY CONTROL the Unit 1 Atmospheric Relief Valve on Steam Generator 1-03 per ABN-301, Instrument Air Malfunction, Attachment 11, Local Control of SG Atmospheric Relief Valves.

Task Standard: Locally control the Atmospheric Relief Valve on a Steam Generator during Loss of Instrument Air per ABN-301.

Required Materials: ABN-301, Instrument Air Malfunction, Rev. 11-7.

Validation Time: 10 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ______________________________ Date: ____________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **ABN-301, Instrument Air Malfunction, Attachment 11, Local Control of SG Atmospheric Relief Valves, OR**
- **Posted Job Aid PLR 2008-0040 for Local Control of SG Atmospheric Relief Valves.**

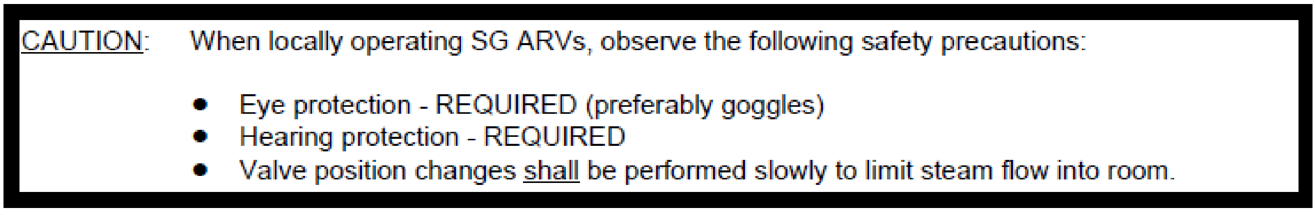
EXAMINER NOTE:

- **This JPM MUST be performed on Unit 1.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**Remind examinee to simulate all actions.**
Examiner Note:	**The Steam Generator Atmospheric Relief Valve is located in the Safeguards Building 881' Room 1-109. Job Aid is posted on wall.**
Examiner Note:	**Unit 1 handwheels are blue.**
Examiner Note:	**The following steps are from ABN-301, Attachment 11.**



CAUTION: When locally operating SG ARVs, observe the following safety precautions:

- Eye protection - REQUIRED (preferably goggles)
- Hearing protection - REQUIRED
- Valve position changes shall be performed slowly to limit steam flow into room.

Perform Step: 1√ 1) & 3rd bullet	Locally close selected Steam Generator Relief Valve upstream isolations: • uMS-0098-R0, SG u-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
Standard:	CLOSED 1MS-0098-R0, SG 1-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER by TURNING handwheel in the CLOCKWISE direction.
Examiner Cue:	**The valve is CLOSED.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√ 2) & 2)a	Perform the following to manually control selected ARV. • Open actuator cylinder bypass valve.
Standard:	OPENED black handled Actuator Cylinder Bypass Valve located on right-hand side of Actuator.
Examiner Cue:	**The valve turned 90°.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√ 2) & 2)b	Perform the following to manually control selected ARV. • Place Bailey positioner by-pass valve in MANUAL (push in to turn).
Standard:	PUSHED IN and TURNED Bailey Positioner Bypass Valve to PLACE in MANUAL.
Examiner Cue:	**The valve is pushed in and turned to position.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 2) & 2)c	Perform the following to manually control selected ARV. • Unscrew coupling (located on top of screw shaft) to access upper stem.
Standard:	UNSCREWED coupling on top of screw shaft to access upper stem.
Examiner Cue:	**The coupling is UNSCREWED.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 5√ 2) & 2)d	Perform the following to manually control selected ARV. • Rotate handwheel in closed direction to lower screw shaft until upper stem exposed sufficient to engage coupling (CCW to close **UNIT 1 ONLY**) (CW to close **UNIT 2 ONLY**).
Standard:	ROTATED the blue handwheel COUNTERCLOCKWISE to lower screw shaft until upper stem exposed sufficient to engage coupling.
Examiner Cue:	**The stem is EXPOSED.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 6√ 2) & 2)e	Perform the following to manually control selected ARV. • Completely insert fork of coupling into groove to secure positive control.
Standard:	INSERTED fork of coupling into groove to secure positive control.
Examiner Cue:	**The fork is INSERTED in the coupling.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 7 2) & 2)f	Perform the following to manually control selected ARV. • Open valve approximately 50% OR as directed by Shift Manager or Unit Supervisor.
Standard:	CONTACTED the Control Room to DETERMINE required valve position.
Examiner Cue:	**The Unit Supervisor directs you to OPEN the valve 50%.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 8√ 2) & 2)f	Perform the following to manually control selected ARV. • Open valve approximately 50% OR as directed by Shift Manager or Unit Supervisor.
Standard:	ROTATED the blue handwheel on top of the ARV until the valve position indicator on the stem reads ~50% OPEN.
Examiner Cue:	**The valve is 50% OPEN. Open the isolation valve 5%.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**The examinee may refer to a posted Job Aid.**
Perform Step: 9√ 3) & 3rd bullet	Locally slowly throttle open selected Steam Generator Relief Valve upstream isolations: • uMS-0098-R0, SG u-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
Standard:	Slowly THROTTLED OPEN 1MS-0098-R0, SG 1-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER in CLOCKWISE direction to 5% OPEN position.
Terminating Cue:	**The valve is 5% OPEN. This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **A Loss of Instrument Air has occurred on Unit 1.**

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

- **LOCALLY CONTROL the Unit 1 Atmospheric Relief Valve on Steam Generator 1-03 per ABN-301, Instrument Air Malfunction, Attachment 11, Local Control of SG Atmospheric Relief Valves.**

[L]

ATTACHMENT 11
PAGE 1 OF 1

LOCAL CONTROL OF SG ATMOSPHERIC RELIEF VALVES

CAUTION: When locally operating SG ARVs, observe the following safety precautions:

- Eye protection - REQUIRED (preferably goggles)
- Hearing protection - REQUIRED
- Valve position changes shall be performed slowly to limit steam flow into room.

1) Locally close selected Steam Generator Relief Valve upstream isolations (SFGD 881' Rm u-109):

- NA uMS-0026-R0, SG u-01 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- ↓ uMS-0063-R0, SG u-02 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- uMS-0098-R0, SG u-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- NA uMS-0134-R0, SG u-04 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER

2) Perform the following to manually control selected ARV.

   a. Open actuator cylinder bypass valve.

   b. Place Bailey positioner by-pass valve in MANUAL (push in to turn).

   c. Unscrew coupling (located on top of screw shaft) to access upper stem.

   d. Rotate handwheel in closed direction to lower screw shaft until upper stem exposed sufficient to engage coupling (CCW to close **UNIT 1 ONLY**) (CW to close **UNIT 2 ONLY**).

   e. Completely insert fork of coupling into groove to secure positive control.

   f. Open valve approximately 50% OR as directed by Shift Manager or Unit Supervisor.

3) Locally slowly throttle open selected Steam Generator Relief Valve upstream isolations:

- NA uMS-0026-R0, SG u-01 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- ↓ uMS-0063-R0, SG u-02 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- uMS-0098-R0, SG u-03 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER
- NA uMS-0134-R0, SG u-04 ATMOS RLF VLV UPSTRM ISOL VLV RMT OPER

Facility: CPNPP JPM # NRC P-2-U1 Task #RO1120 K/A #037.AA1.04 3.6 / 3.9 SF-7

Title: Restore Condenser Off Gas Radiation Detector Dryer

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	X	Classroom:	______
Actual Performance:	______	Simulator:	______
Alternate Path:	______	Plant:	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:

- Unit 1 is in MODE 1.
- 1-HS-2959, Condenser Off-Gas Monitor Vacuum Sample Pump Control handswitch is in OFF.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- RESTORE the Unit 1 Condenser Off Gas Radiation Detector Dryer per SOP-309A, Condenser Vacuum and Water Box Priming System, Section 5.3.4, Bypassing/Restoring Condenser Off Gas Radiation Detector 2959 Dryer.

Task Standard: Restore the Condenser Off Gas Radiation Detector 2959 Dryer to service per SOP-309A.

Required Materials: SOP-309A, Condenser Vacuum and Water Box Priming System, Rev. 20-2.

Validation Time: 6 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: _______________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **SOP-309A, Condenser Vacuum and Water Box Priming System for Unit 1.**
    - **Section 5.3.4, Bypassing/Restoring Condenser Off Gas Radiation Detector 2959 Dryer.**

EXAMINER NOTE:

- **This JPM MUST be performed on Unit 1.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**Remind examinee to simulate all actions.**
Examiner Note:	**The following steps are from SOP-309A, Step 5.3.4.**
Perform Step: 1√ 5.3.4.B.1) & 1st bullet	Restore the Air Dryer by OPENING the following valves: • 1CV-0275, U1 CNDSR OFF GAS RAD DET 2959 DRYER INLET VLV.
Standard:	ROTATED 1CV-0275, U1 CNDSR Off Gas Rad Det 2959 Dryer Inlet VLV 90° COUNTERCLOCKWISE to OPEN position.
Examiner Cue:	**Valve is rotated 90° counterclockwise.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 2√ 5.3.4.B.1) & 2nd bullet	Restore the Air Dryer by OPENING the following valves: • 1CV-0276, U1 CNDSR OFF GAS RAD DET 2959 DRYER OUTLET VLV.
Standard:	ROTATED 1CV-0276, U1 CNDSR Off Gas Rad Det 2959 Dryer Outlet VLV 90° COUNTERCLOCKWISE to OPEN position.
Examiner Cue:	**Valve is rotated 90° counterclockwise.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 3√ 5.3.4.B.2) & 1st bullet	CLOSE the following valves: • 1CV-0277, U1 CNDSR OFF GAS RAD DET 2959 DRYER BY-PASS VLV.
Standard:	ROTATED 1CV-0277, U1 CNDSR Off Gas Rad Det 2959 Dryer By-Pass VLV 90° CLOCKWISE to CLOSE position.
Examiner Cue:	**Valve is rotated 90° clockwise.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 5.3.4.B.2) & 2nd bullet	CLOSE the following valves: • 1CV-0279, U1 CNDSR OFF GAS DRYER BY-PASS FLOAT DRAIN VALVE ISOLATION VALVE
Standard:	ROTATED 1CV-0279, U1 CNDSR Off Gas Dryer By-Pass Float Drain Valve Isolation Valve 90° CLOCKWISE to CLOSE position.
Examiner Cue:	**Valve is rotated 90° clockwise.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 5 5.3.4.B.3)	Ensure 1-HS-2959, CONDENSER OFF-GAS MONITOR VACUUM SAMPLE PUMP CONTROL HAND SWITCH, is in AUTO.
Standard:	ROTATED 1-HS-2959, Condenser Off-Gas Monitor Vacuum Sample Pump Control handswitch to AUTO position.
Examiner Cue:	**Handswitch is in AUTO.**
Comment:	**SAT** ☐ **UNSAT** ☐

Perform Step: 6 5.3.4.B.4)	Verify COG182 at PC-11 is indicating <u>AND</u> green.
Standard:	CONTACTED the Control Room to VERIFY that COG182 at PC-11 is indicating and green.
Terminating Cue:	**COG182 at PC-11 is indicating and green. This JPM is complete.**
Comment:	**SAT** ☐ **UNSAT** ☐

STOP TIME:	

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

- **Unit 1 is in MODE 1.**
- **1-HS-2959, Condenser Off-Gas Monitor Vacuum Sample Pump Control handswitch is in OFF.**

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

- **RESTORE the Unit 1 Condenser Off Gas Radiation Detector Dryer per SOP-309A, Condenser Vacuum and Water Box Priming System, Section 5.3.4, Bypassing/Restoring Condenser Off Gas Radiation Detector 2959 Dryer.**

5.3.4 Bypassing/Restoring Condenser Off Gas Radiation Detector 2959 Dryer

This section describes the steps to by-pass and restore CP1-CVDYRE-01, Condenser Off Gas Radiation Detector 2959 Dryer. The sample may be routed through the by-pass until the dryer is back in service. The by-pass line and associated valves, coalescing filter and drain system will provide an alternate route in the case where the dryer is unavailable.

NOTE:

- ✓ This bypass system will provide some moisture removal but is not expected to serve the radiation monitor with the driest air possible; therefore, when this system is placed in service, Operations and Chemistry personnel must be aware of the limitations of the monitor since it may not be operating at its highest efficiency.
- ✓ 1-RE-2959, UNIT 1 CONDENSER OFF GAS RADIATION DETECTOR is designed to detect primary-to-secondary leakage continuously at rates of 30 gpd or less and is identified in ODA-308 as a component subject to the Systems Important to Safety Log (SISL).
- ✓ STA-732 requires Chemistry to collect grab samples to support leakage detection when ANY Steam Generator Leak Rate Monitor AND the Condenser Off-Gas Monitor are determined to be unavailable for detecting primary-to-secondary leak rates less than or equal to 30 gpd.
- ✓ When CP1-CVDYRE-01, CONDENSER OFF GAS RADIATION DETECTOR 2959 VACUUM PUMP AIR DRYER 1-01 is bypassed, moisture condensation may accumulate and affect the ability of 1-RE-2959 to accurately detect primary-to-secondary leak rates less than or equal to 30 gpd.
- ✓ Several of the following steps require an Equipment Operator to be dispatched to CP1-CVDYRE-01, U1 TB 803 MEZZANINE FLOOR.

A. Bypassing CP1-CVDYRE-01

WHEN it is desired to bypass CP1-CVDYRE-01, THEN perform the following:

1) OPEN the following valves:

☒ • 1CV-0277, U1 CNDSR OFF GAS RAD DET 2959 DRYER BY-PASS VLV

☒ • 1CV-0279, U1 CNDSR OFF GAS DRYER BY-PASS FLOAT DRAIN VALVE ISOLATION VALVE

2) Isolate the Air Dryer by CLOSING the following valves:

☒ • 1CV-0275, U1 CNDSR OFF GAS RAD DET 2959 DRYER INLET VLV

☒ • 1CV-0276, U1 CNDSR OFF GAS RAD DET 2959 DRYER OUTLET VLV

5.3.4 A.

> ~~NOTE~~: In the next step, placing 1-HS-2959 in OFF will make 1-RE-2959 incapable of detecting primary-to-secondary leak rates less than or equal to 30 gpd . Refer to ODA-308-31.

☒ 3) IF desired, THEN place 1-HS-2959, CONDENSER OFF-GAS MONITOR VACUUM SAMPLE PUMP CONTROL HAND SWITCH in OFF.

B. Restoring CP1-CVDYRE-01 to Service

> NOTE: Moisture accumulation may impede monitor sample flow. IF necessary, contact PROMPT Team to drain the system including the suction and discharge side of the Monitor Vacuum Pump, and the coalescing filter bypass rack.

WHEN it is desired to restore CP1-CVDYRE-01 from bypass, THEN perform the following:

1) Restore the Air Dryer by OPENING the following valves:

- ☐ 1CV-0275, U1 CNDSR OFF GAS RAD DET 2959 DRYER INLET VLV.
- ☐ 1CV-0276, U1 CNDSR OFF GAS RAD DET 2959 DRYER OUTLET VLV.

2) CLOSE the following valves:

- ☐ 1CV-0277, U1 CNDSR OFF GAS RAD DET 2959 DRYER BY-PASS VLV.
- ☐ 1CV-0279, U1 CNDSR OFF GAS DRYER BY-PASS FLOAT DRAIN VALVE ISOLATION VALVE.

☐ 3) Ensure 1-HS-2959, CONDENSER OFF-GAS MONITOR VACUUM SAMPLE PUMP CONTROL HAND SWITCH, is in AUTO.

☐ 4) Verify COG182 at PC-11 is indicating AND green.

COMMENTS: ______________________________

____

Facility: CPNPP JPM # NRC P-3-U1 Task #RO4405 K/A #086.A1.05 2.9 / 3.1 SF-8

Title: Respond to Fire in Service Water Intake Structure

Examinee (Print): ______________________________

Testing Method:

Simulated Performance:	X	Classroom:	______
Actual Performance:	______	Simulator:	______
Alternate Path:	______	Plant:	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:

- Unit 1 is in MODE 5.
- A fire in the Unit 1 Service Water Intake Structure is in progress.

Initiating Cue: The Unit Supervisor directs you to PERFORM the following:

- ALIGN the alternate power supply to the Unit 1 Train B Residual Heat Removal Pump Hot Leg Recirculation Isolation Valve per ABN-808A, Response to Fire in Service Water Intake Structure, Attachment 2, Alternate Power Supply Hookup for 1-8701B.

Task Standard: Align the alternate power supply to the Train B Residual Heat Removal Pump Hot Leg Recirculation Isolation Valve per ABN-808A.

Required Materials: ABN-808A, Response to Fire in Service Water Intake Structure, Rev. 5.

Validation Time: 14 minutes Time Critical: N/A Completion Time: ________ minutes

Comments:

Result: SAT ☐ UNSAT ☐

Examiner (Print / Sign): ____ Date: ______________

PLANT SETUP

EXAMINER:

PROVIDE the examinee with a copy of:

- **ABN-808A, Response to Fire in Service Water Intake Structure for Unit 1.**
    - **Attachment 2, Alternate Power Supply Hookup for 1-8701B.**

EXAMINER NOTE:

- **This JPM MUST be performed on Unit 1.**

√ - Check Mark Denotes Critical Step

START TIME:	

Examiner Note:	**Remind examinee to simulate all actions.**
Examiner Note:	**The following steps are from ABN-808A, Attachment 2.**
Examiner Note:	**Breakers are located in Safeguards Building 852', Room 1-103 on Motor Control Center 1-EB4.**
Perform Step: 1√ 1 & 1st bullet	Open the following breakers: • 1EB4-2/1M/BKR-1 & 2, RHR PUMP 1-02 HOT LEG 1-04 RECIRC PMP ISOL VLV 8701B ALT MOT BKR-1 & 2 (SFGD 852 Rm 1-103)
Standard:	TURNED both breakers 1EB4-2/1M/BKR-1 **and** 1EB4-2/1M/BKR-2 to OFF position.
Examiner Cue:	**Breakers are in OFF.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Breakers are located in Safeguards Building 810', Room 1-083 on Motor Control Center 1-EB3.**
Perform Step: 2√ 1 & 2nd bullet	Open the following breakers: • 1EB3-2/9M/BKR-1 &2, RHR PMP 1-02 HL 1-04 RECIRC OMB ISOL VLV 1-8701B PREF MOTOR BREAKER-1 & 2 (SFGD 810 Rm 1-083)
Standard:	TURNED both breakers 1EB3-2/9M/BKR-1 **and** 1EB3-2/9M/BKR-2 to OFF position.
Examiner Cue:	**Breakers are in OFF.**
Comment:	**SAT ☐ UNSAT ☐**

Examiner Note:	**Cable box is located in Safeguards Building 810', Room 1-083 on the outer Containment wall.**
Perform Step: 3√ 2 & 1st bullet	Disconnect the following control cables from their connectors in JB1S-12290 (SFGD 810 Rm 1-083 on the outer containment wall): • Cable EO122920A from connector C-1-8701B-CN
Standard:	UNSCREWED Cable EO122920A from connector C-1-8701B-CN.
Examiner Cue:	**Cable is unscrewed.**
Comment:	**SAT ☐ UNSAT ☐**

Perform Step: 4√ 2 & 2nd bullet	Disconnect the following control cables from their connectors in JB1S-12290 (SFGD 810 Rm 1-083 on the outer containment wall): • Cable EO100806A from connector C-1-8701B-PN	
Standard:	UNSCREWED Cable EO100806A from connector C-1-8701B-PN.	
Examiner Cue:	**Cable is unscrewed.**	
Comment:		**SAT ☐ UNSAT ☐**

Examiner Note:	**Cables are routed through the connector box then plugged in.**	
Perform Step: 5√ 3	Route cables to JB1S-1230G via JB1S-12280.	
Standard:	ROUTED cables to JB1S-1230G via JB1S-12280.	
Examiner Cue:	**Cables are routed through the connector box.**	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 6√ 4 & 1st bullet	Plug cables into their respective connectors at JB1S-1230G: • EO122920A into connector C-1-8701B-CA	
Standard:	PLUGGED cable EO122920A into connector C-1-8701B-CA.	
Examiner Cue:	**Cable is plugged in.**	
Comment:		**SAT ☐ UNSAT ☐**

Perform Step: 7√ 4 & 2nd bullet	Plug cables into their respective connectors at JB1S-1230G: • EO100806A into connector C-1-8701B-PA	
Standard:	PLUGGED cable EO100806A into connector C-1-8701B-PA.	
Examiner Cue:	**Cable is plugged in.**	
Comment:		**SAT ☐ UNSAT ☐**

CAUTION: All RHR suction valve interlocks are bypassed when the valve is operated from the MCC or locally. Proper operating sequence of RHR suction valves is critical to prevent aligning RHR to the RCS without isolating the RWST.

Perform Step: 8√ 5	Close 1EB4-2/1M/BKR-1 & 2.
Standard:	TURNED both breakers 1EB4-2/1M/BKR-1 **and** 1EB4-2/1M/BKR-2 to ON position and OBSERVED green CLOSE light lit.
Terminating Cue:	**Breakers are ON and green CLOSE light is lit. This JPM is complete.**
Comment:	**SAT ☐ UNSAT ☐**

STOP TIME:	

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

- **Unit 1 is in MODE 5.**
- **A fire in the Unit 1 Service Water Intake Structure is in progress.**

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

- **ALIGN the alternate power supply to the Unit 1 Train B Residual Heat Removal Pump Hot Leg Recirculation Isolation Valve per ABN-808A, Response to Fire in Service Water Intake Structure, Attachment 2, Alternate Power Supply Hookup for 1-8701B.**

ATTACHMENT 2
PAGE 1 OF 1

ALTERNATE POWER SUPPLY HOOKUP FOR 1-8701B

1. Open the following breakers:

   ☐ ● 1EB4-2/1M/BKR-1 & 2, RHR PUMP 1-02 HOT LEG 1-04 RECIRC PMP ISOL VLV 8701B ALT MOT BKR-1 & 2 (SFGD 852 Rm 1-103)

   ☐ ● 1EB3-2/9M/BKR-1 &2, RHR PMP 1-02 HL 1-04 RECIRC OMB ISOL VLV 1-8701B PREF MOTOR BREAKER-1 & 2 (SFGD 810 Rm 1-083)

☐ 2. Disconnect the following control cables from their connectors in JB1S-12290 (SFGD 810 Rm 1-083 on the outer containment wall):

   - Cable EO122920A from connector C-1-8701B-CN
   - Cable EO100806A from connector C-1-8701B-PN

☐ 3. Route cables to JB1S-1230G via JB1S-1228 0

☐ 4. Plug cables into their respective connectors at JB1S-1230G:

   - EO122920A into connector C-1-8701B-CA
   - EO100806A into connector C-1-8701B-PA

<u>CAUTION</u>: All RHR suction valve interlocks are bypassed when the valve is operated from the MCC or locally. Proper operating sequence of RHR suction valves is critical to prevent aligning RHR to the RCS without isolating the RWST.

☐ 5. Close 1EB4-2/1M/BKR-1 & 2.

☐ 6. Operate 1-8701B as required from 1EB4-2/1M/BKR. Valve position indication and handswitch are located on Breaker.

Facility:	CPNPP 1 & 2	Scenario No.: 1	Op Test No.: June 2011 NRC
Examiners:		Operators:	

Initial Conditions:
- 100% power MOL - RCS Boron is 908 ppm (by sample).

Turnover: Maintain steady-state power conditions.

Critical Tasks:
- Manually Initiate Safety Injection due to Failure to Automatically Actuate Prior to Exiting EOP-0.0A.
- Perform Actions to Identify and Isolate Faulted Steam Generator Prior to exiting EOP-2.0A.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	RP05B	I (RO, SRO) TS (SRO)	Reactor Coolant System Loop 2 T_{COLD} Instrument (TI-421A) Fails High.
2 +30 min	ED07B	C (RO, BOP, SRO) TS (SRO)	Loss of Protection Bus IV1PC2.
3 +50 min	TU04	R (RO) N (BOP, SRO)	Main Turbine Bearing Vibration at 10.5 mils (180 second ramp). Power Reduction to Lower Main Turbine Vibration.
4 +55 min	MS02	M (RO, BOP, SRO)	Main Steam Header Leak Outside Containment (300 second ramp).
5 +55 min	RP07A RP07B	I (RO)	Safety Injection Trains A and B Fail to Automatically Actuate.
6 +55 min	RP08B	I (BOP)	Manual Safety Injection Train B Failure at CB-07.
7 +55 min	MS08C	C (RO)	Steam Generator (1-03) Main Steam Isolation Valve (HV-2335A) Fails to Close.
8 +60 min	RH01C	C (BOP)	Residual Heat Removal Pump (1-01) Auto Start Failure on Safety Injection Signal.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	**Target Quantitative Attributes**
8	Total malfunctions (5-8)
4	Malfunctions after EOP entry (1-2)
3	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO SUMMARY NRC #1

The crew will assume the watch at 100% power with no scheduled activities per IPO-003A, Power Operations.

The first event it is a high failure of T_{COLD} Temperature Instrument, TI-421A. Operator actions are per ABN-704, TC/N-16 Instrumentation Malfunction, and require stopping Control Rod motion and stabilizing Reactor Coolant System (RCS) temperature and Pressurizer level. The SRO will refer to Technical Specifications.

The next event is a Loss of Protection Bus IV1PC2. Crew actions are per ABN-603, Loss of a Protection or Instrument Bus, and include stabilizing the plant, restoring an alternate power source, and verification of instrument restoration. The SRO will refer to Technical Specifications.

The next event is initiated with Main Turbine high vibration. The crew enters ABN-401, Main Turbine Malfunction, which will require reducing load to 900 MWe. When the crew commences reducing load, Main Turbine vibration will improve over a 10 minute period.

When Main Turbine vibration is restored to normal, a Main Steam header leak will ramp in over 300 seconds. The crew should recognize the requirement to manually trip the Reactor. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection. While performing the actions of EOP-0.0A, the RO will attempt to manually initiate both Trains of Safety Injection at CB-07; however, this task will be completed by the BOP at CB-02.

While performing actions in EOP-0.0A, the crew should recognize lowering Main Steam pressure with an associated Main Steam Isolation Signal. Steam Generator 1-03 Main Steam Isolation Valve HV-2335A will fail to automatically or manually close. The crew will transition from EOP-0.0A to EOP-2.0A, Faulted Steam Generator. When the Faulted Steam Generator (1-03) has been isolated, entry into EOS-1.1A, Terminate Safety Injection, is performed.

The scenario is includes a Residual Heat Removal Pump that fails to start upon initiation of the Safety Injection Sequencer. This scenario is terminated when the Faulted Steam Generator is isolated and the crew secures High Head Safety Injection.

Risk Significance:

• Failure of risk important system prior to trip:	Loss of Inverter IV1PC2
• Risk significant core damage sequence:	Main Steam Header Failure
	Main Turbine Vibration
• Risk significant operator actions:	Restore Power to Protection Bus 1PC2
	Manually Initiate Safety Injection
	Manually Start RHR Pump
	Isolate Faulted Steam Generator

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC #18 and Event File for NRC Scenario #1.

EVENT	TYPE	MALF #	DESCRIPTION	DEMAND VALUE	INITIATING PARAMETER
SETUP		RP07A/B	Safety Injection Train A/B actuation failure	OFF	K0
		RP08B	Manual SI Train B actuation failure at CB-07	OFF	K0
		MS08C	SG (1-03) MSIV (HV-2335A) fails to close	OPEN	K0
		RH01C	RHR Pump (1-01) auto start failure on SI signal	-	K0
		CS02E	CS Pump (1-01) auto start failure on SI signal	-	K0
1		RP05B	Loop 2 Tcold NR Instrument (TE-421A) failure	630ºF	K1
2		ED07B	Loss of Inverter IV1PC2	TRIP	K2
2	EDR02		Restore Inverter IV1PC2 power	ALT	K10
3		TU04	Main Turbine bearing vibration at 10.5 mils Power reduction **(NOTE 1)**	10.5 mils total	K3 (180 sec. ramp)
NOTE 1: When load reduction is initiated, RAMP TU04 to 4 mils over 600 seconds.					
4		MS02	Main Steam header leak	4E^7 lbm/hr	K4 (300 sec. ramp)
5		RP07A/B	Safety Injection Train A/B actuation failure	OFF	K0
6		RP08B	Manual SI Train B actuation failure at CB-07	OFF	K0
7		MS08C	SG (1-03) MSIV (HV-2335A) fails to close **(NOTE 2)**	OPEN	K0
NOTE 2: When directed to locally close valve, DELETE malfunction MS08C.					
8		RH01C	RHR Pump (1-01) auto start failure on SI signal	-	K0

Booth Operator: **INITIALIZE to IC #18 and NRC Scenario #1 SETUP file.**
ENSURE all Simulator Annunciator Alarms are ACTIVE.
ENSURE all Control Board Tags are removed.
ENSURE Operator Aid Tags reflect current boron conditions.
ENSURE Rod Bank Update (RBU) is performed.
ENSURE Turbine Load Rate set at 10 MWe/minute.
ENSURE 60/90 buttons DEPRESSED on ASD.
ENSURE Reactivity Briefing Sheet printout provided with Turnover.
ENSURE procedures in progress are on SRO desk:
- **COPY of IPO-003A, Power Operations, Section 5.5, Operating at Constant Turbine Load.**

ENSURE Control Rods are in AUTO with Bank D at 215 steps.

Control Room Annunciators in Alarm:

PCIP-1.1 – SR TRN A RX TRIP BLK
PCIP-1.2 – IR TRN A RX TRIP BLK
PCIP-1.4 – CNDSR AVAIL STM DMP ARMED C-9
PCIP-1.6 – RX ≥ 10% PWR P-10
PCIP-2.1 – SR TRN B RX TRIP BLK
PCIP-2.2 – IR TRN B RX TRIP BLK
PCIP-2.5 – SR RX TRIP BLK PERM P-6
PCIP-3.2 – PR TRN A LO SETPT RX TRIP BLK
PCIP-4.2 – PR TRN B LO SETPT RX TRIP BLK

Operating Test :	NRC	Scenario #	1	Event #	1	Page	5 of 22

Event Description: Loop 2 T_{COLD} Temperature Instrument Failure

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When directed, EXECUTE Event 1. - RP05B, Loop #2 Tcold NR temperature instrument (TI-421A) fails high.		
<u>Indications Available</u>: 5C-1.5 – ANY N16 DEV HI / LO 5C-2.5 – 1 OF 4 OT N16 HI 5C-2.6 – 1 OF 4 OP N16 HI (comes in then clears) 5C-3.5 – ANY T_{AVE} DEV HI / LO 6D-1.10 – AVE T_{AVE} T_{REF} DEV 6D-2.10 – AVE T_{AVE} HI 6D-2.13 – 1 OF 4 OP N16 ROD STOP & TURB RUNBACK 6D-3.14 – 1 OF 4 OT N16 ROD STOP & TURB RUNBACK 1-TI-421A, CL 2 TEMP (NR) CHAN II indication fails high		
+1 min	RO/BOP	RESPOND to Annunciator Alarm Procedures.
	RO	RECOGNIZE Control Rods inserting due to T_{COLD} failed high.
	US	DIRECT performance of ABN-704, Tc / N-16 Instrumentation Malfunction, Section 2.0.
	RO	PLACE 1/1-RBSS Control Rod Bank Select Switch in MANUAL.
	RO	SELECT LOOP 2 on 1-TS-412T, T_{AVE} Channel Defeat.
	RO/BOP	VERIFY Steam Dump System is NOT actuated and NOT armed.
<u>Examiner Note</u>: Crew will withdraw rods to 215 steps in 5 step increments to restore T_{AVE}.		
	RO	RESTORE T_{AVE} to within 1ºF of T_{REF}.
	RO	SELECT LOOP 2 on 1/1-JS-411E, N16 Power Channel Defeat.
	RO	ENSURE a valid N16 channel supplying recorder on 1/1-TS-411E, 1-TR-411 CHAN SELECT.

Operating Test :	NRC	Scenario #	1	Event #	1	Page	6	of 22
Event Description:	Loop 2 T_{COLD} Temperature Instrument Failure							

Time	Position	Applicant's Actions or Behavior
	RO/BOP	VERIFY Steam Dump System is NOT armed by OBSERVING PCIP-3.4 alarm – DARK.
	US	EVALUATE Technical Specifications.
		• LCO 3.3.1.E, Reactor Trip System Instrumentation.
		• CONDITION E - One channel inoperable. • ACTION E.1 - Place channel in trip within 72 hours.
+10 min	US	INITIATE a work request per STA-606.
When Technical Specifications are addressed, or at Lead Examiner discretion, PROCEED to Event 2.		

Operating Test : NRC	Scenario # 1	Event # 2		Page 7	of 22		

Event Description: Loss of Protection Bus IV1PC2

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 2. - ED07B, Loss of Protection Bus IV1PC2.		
Indications Available: **10B-2.16 – 118V CHAN II INV TRBL** **5A-1.3 – RC LOOP 1 1 OF 3 FLO LO** **5A-2.3 – RC LOOP 2 1 OF 3 FLO LO** **5A-3.3 – RC LOOP 3 1 OF 3 FLO LO** **5A-4.3 – RC LOOP 4 1 OF 3 FLO LO** **Channel 2 Windows on TSLB 1 through 7 and 9** **Numerous Other Loss of Protection Bus 1PC2 Alarms**		
+30 sec	RO/BOP	RESPOND to Annunciator Alarm Procedures.
	RO/BOP	RECOGNIZE loss of Protection Bus 1PC2.
	US	DIRECT performance of ABN-603, Loss of Protection or Instrument Bus, Section 2.0.
Booth Operator: If contacted, REPORT Inverter failure on 1PC2 with acrid odor in room.		
	RO/BOP	VERIFY Reactor did NOT trip.
	US	DETERMINE Unit in MODE 1.
	RO	ENSURE 1/1-RBSS, Control Rod Bank Select Switch in MANUAL.
	BOP	Manually CONTROL Steam Generator levels and Main Feed Pumps as necessary to maintain level.
	RO	Manually CONTROL Charging to maintain Pressurizer level as required.
		• DETERMINE Letdown is isolated.
		• ADJUST 1-HCV-182, Seal Injection Flow as necessary.
	RO	VERIFY RCP seal injection within normal operating range.

Operating Test :	NRC	Scenario #	1	Event #	2	Page	8	of 22
Event Description:	Loss of Protection Bus IV1PC2							

Time	Position	Applicant's Actions or Behavior
	RO	VERIFY Pressurizer level control between 25% and 70%.
	RO	DETERMINE Pressurizer pressure within normal operating range.
	BOP	DETERMINE Steam Generator levels NOT being controlled between 60% and 70%.
	BOP	As necessary, PLACE 1-SK-509A, MFW Pump Master Controller in MANUAL.
	BOP	PLACE 1-FK-520 and 1-FK-530, SG 1-02 and SG 1-03 Feedwater Regulating Valves in MANUAL and CONTROL Steam Generator level.
	RO	DETERMINE Loop 2 selected on 1-TS-412T, T_{AVE} CHAN DEFEAT Switch.
	RO/BOP	DISPATCH an operator to REENERGIZE Protection Bus 1PC2.
<u>Booth Operator</u>: When contacted to reenergize 1PC2, WAIT 2 minutes then PERFORM remote function EDR02 to transfer to the alternate power supply.		
	BOP	RESET C-7, Steam Dump Arming Signal Interlock.
		• PLACE 43/1-SD, Steam Dump Mode Select switch to RESET.
	US/RO	DETERMINE 1-TS-412T, Tave CHAN DEFEAT Switch should remain in Loop 2 position.
	US/RO	DETERMINE Control Rod Bank Select Switch should remain in MANUAL.
	BOP	PLACE 1-FK-520 and 1-FK-530, SG 1-02 and SG 1-03 Feedwater Regulating Valves in AUTO and MONITOR Steam Generator level.
	BOP	As necessary, PLACE 1-SK-509A, MFW Pump Master Controller in AUTO.
	RO	RESET Power Range Flux Rate Mode Selector on Drawer N-42A and VERIFY Positive Rate Mode alarm light off.

Operating Test :	NRC	Scenario #	1	Event #	2	Page	9 of 22
Event Description:	Loss of Protection Bus IV1PC2						

Time	Position	Applicant's Actions or Behavior
	RO	ADJUST 1-HC-182, Seal Flow Control Valve to CONTROL RCP seal flow.
	RO	ENSURE 1/1-8105 and 1/1-8106, Charging Isolation Valves are OPEN.
	RO	RESTORE Letdown flow per Control Board Job Aid.
		• OPEN or VERIFY OPEN both Letdown Isolation Valves.
		• ENSURE 1-PK-131, LTDN HX OUT PRESS CTRL in MANUAL and 30% (75 gpm) or 50% (120 gpm) DEMAND.
		• ENSURE 1-TK-130, LTDN HX OUT TEMP CTRL in MANUAL and 50% DEMAND.
		• ADJUST Charging to desired flow and MAINTAIN Seal Injection flow between 6 and 13 gpm.
		• OPEN the desired Orifice Isolation Valves.
		• ADJUST 1-PK-131, LTDN HX OUT PRESS CTRL to ~310 psig on 1-PI-131, LTDN HX OUT PRESS then PLACE in AUTO.
		• ADJUST 1-TK-130, LTDN HX OUT TEMP CTRL to obtain ~95ºF on 1-TI-130, LTDN HX OUT TEMP, then place in AUTOMATIC.
+20 min	US	EVALUATE Technical Specifications.
		• LCO 3.8.7.A, Inverters - Operating.
		• CONDITION A - One required inverter inoperable. • ACTION A.1 - Restore inverter to OPERABLE status within 24 hours.
Examiner Note: **LCO 3.8.9B would be entered if power was NOT restored to Bus IV1PC2. It MAY or MAY NOT be reported due to its short duration.**		
		• LCO 3.8.9.B, Distribution Systems - Operating.
		• CONDITION B - One AC vital bus subsystem inoperable. • ACTION B.1 - Restore AC vital bus subsystem to OPERABLE status within 2 hours.
When Technical Specifications are addressed, or at Lead Examiner discretion, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	1	Event #	3	Page	10 of 22
Event Description:	Main Turbine Bearing Vibration and Power Reduction						

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 3. - TU04, Main Turbine bearing vibration at 10.5 mils on 180 second ramp.		
Indications Available: **Main Turbine Digital Alarm Summary Display in alarm**		
+3 min	BOP	RESPOND to Main Turbine Digital Alarm Summary.
	US	DIRECT performance of ABN-401, Main Turbine Malfunction, Section 2.0.
	BOP	OBSERVE Turbine Vibration and Generator Vibration Displays to determine alarm validity.
	BOP	DETERMINE Turbine Vibration and Generator Vibration Displays all readings either yellow or green.
	BOP	DETERMINE Turbine Vibration and Generator Vibration Displays has yellow readings (LP2 Turbine rear shaft bearing).
Examiner Note: The crew may initially execute a 50 MWe power reduction (Runback) in an attempt to reduce Turbine vibration. Rods may or may not be placed in AUTO.		
	US	NOTIFY Generation Controller of imminent load reduction.
	BOP/US	DETERMINE Turbine shaft vibration greater than 10 mils.
	US	NOTIFY Plant Management of the need to reduce load and CONTACT System Engineering.
Booth Operator: When Plant Management is notified, REPORT as Shift Manager to ramp the Unit to 900 MWe in 30 minutes per the Reactivity Briefing Sheet.		
	US	DIRECT load reduction to 900 MWe per IPO-003A, Power Operations, Section 5.6, Reducing Turbine Power from 100% to MODE 3.

Time	Position	Applicant's Actions or Behavior
	RO	INITIATE RCS boration per SOP-104A, Reactor Make-up and Chemical Control System.
	BOP	SET Turbine Load Rate Setpoint Controller as desired and Load Target to 900 MWe.
Examiner Note:		**Crew may or may not borate for the power change. If boration is desired, the amount would be determined per the Reactivity Briefing Sheet.**
	RO	If desired, PERFORM the following to COMMENCE RCS boration:
		• ENSURE 1/1-MU, RCS Makeup Manual Actuation is in STOP.
		• PLACE 43/1-MU, RCS Makeup Mode Select in BORATE.
		• SET 1-FK-110, BA Blender Flow Control to desired flowrate.
		• SET 1-FY-110B, BA Batch Flow counter for the desired number of gallons.
		• ENSURE 1/1-FCV-110A, Boric Acid Blender Flow Control Valve is in AUTO.
		• PLACE 1/1-MU, RCS Makeup Manual Actuation in START.
		• VERIFY 1/1-APBA1, Boric Acid Transfer Pump starts.
		• VERIFY 1/1-FCV-110A, Boric Acid Blender Flow Control Valve throttles to the preset flow rate.
		• VERIFY 1/1-FCV-110B, RCS Makeup to Charging Pump Suction Isolation Valve OPEN.
		• VERIFY 1-FR-110, Boric Acid Flow to Blender RED pen operating properly.
		• VERIFY 1-FY-110B, Batch Flow counter operating properly.
		• When desired amount of boric acid is added, PLACE 1/1-MU, RCS Makeup Manual Actuation in STOP.
		• FLUSH the blender with approximately 50 gallons makeup water when boration is complete.
Booth Operator:		**When Turbine load reduction is commenced, RAMP malfunction TU04 to 4 mils over 10 minutes.**

Operating Test :	NRC	Scenario #	1	Event #	3	Page	12	of 22
Event Description:	Main Turbine Bearing Vibration and Power Reduction							

Time	Position	Applicant's Actions or Behavior
	BOP	PERFORM the following to LOWER Turbine Load:
		• CHANGE Turbine Load Rate to ~12 MWe/min.
		• OPEN "Load Target" OSD.
		• SELECT blue bar and ENTER 900 MWe.
		• DEPRESS "Accept" then VERIFY value in blue bar is desired "Load Target" (magnitude and direction).
		• DEPRESS "Execute" then VERIFY "Load Target" changes to desired load.
		• CLOSE "Load Target" OSD.
+20 min	CREW	MONITOR load change.
When power is reduced 3% to 5% and Turbine vibration is lowering, or at Lead Examiner discretion, PROCEED to Events 4, 5, 6, 7, 8, and 9.		

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	13	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Events 4, 5, 6, 7, 8, and 9. - MS02, Main Steam header leak @ 4E^7 lbm/hr. - RP07A/B, Safety Injection Trains A and B fail to auto actuate. - RP08B, Manual Safety Injection Train B failure at CB-07. - MS08C, HV-2335A, SG 1-03 Main Steam Isolation Valve fails to close. - RH01C, Residual Heat Removal Pumps start failure on SI Sequencer.		
Indications Available: **6A-3.4 – CHRG FLO HI / LO** **5C-3.3 – PRZR PRESS LO BACKUP HTRS ON**		
+30 sec	RO/BOP	RECOGNIZE lowering RCS temperature and pressure.
Booth Operator: If asked, REPORT steam in the Turbine Building.		
	RO/BOP	DETERMINE Reactor Trip required and manually TRIP Reactor.
	US	DIRECT performance of EOP-0.0A, Reactor Trip or Safety Injection.
	RO	VERIFY Reactor Trip:
		• DETERMINE Reactor Trip Breakers – OPEN.
		• DETERMINE Neutron flux – DECREASING.
	RO	DETERMINE all Control Rod Position Rod Bottom Lights – ON.
	BOP	VERIFY Turbine Trip:
		• DETERMINE all HP Turbine Stop Valves – CLOSED.
	BOP	VERIFY Power to AC Safeguards Buses:
		• DETERMINE both AC Safeguards Buses – ENERGIZED.
	RO	DETERMINE SI required but NOT actuated.
CRITICAL TASK STATEMENT		**Manually Initiate Safety Injection due to Failure to Automatically Actuate Prior to Exiting EOP-0.0A.**

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	14	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior
CRITICAL TASK	RO	Manually INITIATE both Trains of Safety Injection.
	RO	• PLACE 1/1-SIA2, SI MAN ACT Switch to ACT position at CB-07.
	BOP	• PLACE 1/1-SIA1, SI MAN ACT Switch to ACT position at CB-02.
Examiner Note: **EOP-0.0A, Attachment 2 steps performed by BOP are identified later in the scenario. The RCPs may be tripped if subcooling is observed to be < 25ºF. This condition is only temporary and subcooling will recover by the time EOP-0.0A, Step 11, Check If RCPs Should Be Stopped, is performed.**		
	US/BOP	INITIATE Proper Safeguards Equipment Operation Per Attachment 2.
	RO	VERIFY AFW Alignment:
		• DETERMINE both MDAFW Pumps – RUNNING.
		• PLACE Turbine Driven AFW Pump in PULLOUT per Foldout Page.
		• CONTROL AFW Flow as follows:
		• CONTROL AFW flow as necessary to maintain narrow range level > 43% in any SG or total AFW flow > 460 gpm per Foldout Page.
		• STOP AFW flow to Faulted SG 1-03 per Foldout Page.
		• MAINTAIN proper AFW valve alignment.
	RO	VERIFY Containment Spray Not Required:
		• VERIFY 1-ALB-2B Window 1-8, CS ACT NOT illuminated.
		• VERIFY 1-ALB-2B Window 4-11, CNTMT ISOL PHASE B ACT NOT illuminated.
		• VERIFY Containment pressure < 18.0 PSIG.
		• VERIFY Containment Spray Heat Exchanger Outlet Valves – CLOSED.
		• VERIFY Containment Spray Pumps – RUNNING.
Booth Operator: **When contacted, WAIT 2 minutes then CLOSE MSIV 1-03.**		

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	15	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior
	RO	DETERMINE Main Steam lines should be ISOLATED:
		• VERIFY Main Steam Isolation complete:
		• DETERMINE Main Steam Isolation Valves – NOT CLOSED.
	RO	• [RNO] MANUALLY or LOCALLY CLOSE MSIV 1-03.
		• VERIFY Main Steam Isolation Bypass Valves – CLOSED.
		• VERIFY Before MSIV Drip Pot Isolation Valves – CLOSED.
	RO	CHECK RCS Temperature:
		• DETERMINE RCS Average Temperature less than 557ºF.
	RO	VERIFY NOT dumping steam.
	RO	REDUCE total AFW flow to minimize the cooldown:
		• MAINTAIN a minimum of 460 gpm UNTIL narrow range level greater than 50% in at least one SG.
		• STOP Turbine Driven AFW Pump.
	RO	CHECK PRZR Valve Status:
		• VERIFY PRZR Safeties – CLOSED.
		• VERIFY Normal PRZR Spray Valves – CLOSED.
		• VERIFY PORVs – CLOSED.
		• VERIFY Power to at least one Block Valve – AVAILABLE.
		• VERIFY Block Valves – AT LEAST ONE OPEN.
	RO	CHECK if RCPs Should Be Stopped:
		• DETERMINE all ECCS Pumps – RUNNING.
		• DETERMINE RCS subcooling – GREATER THAN 25ºF.
		• Continue RUNNING Reactor Coolant Pumps.
	RO/BOP	CHECK if Any Steam Generator Is Faulted:
		• DETERMINE SG 1-03 completely DEPRESSURIZED.

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	16	of	22
Event Description:	Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
	US	TRANSITION to EOP 2.0A, Faulted Steam Generator Isolation, Step 1.
Examiner Note: **EOP-2.0A, Faulted Steam Generator Isolation, steps begin here.**		
+15 min	US/RO	CHECK Main Steam line Isolation Valves – CLOSED.
		• VERIFY MSIV 1-03 – LOCALLY CLOSED.
	US/RO	CHECK at Least One Steam Generator Pressure – STABLE OR INCREASING.
	US/RO	IDENTIFY Faulted Steam Generator 1-03.
CRITICAL TASK STATEMENT		**Perform Actions to Identify and Isolate Faulted Steam Generator Prior to exiting EOP-2.0A.**
CRITICAL TASK	RO/BOP	ISOLATE Faulted Steam Generator 1-03.
		• ISOLATE Main Feed Line to Steam Generator 1-03.
		• ISOLATE AFW flow to Steam Generator 1-03.
		• ISOLATE Blowdown and Sample Lines to Steam Generator 1-03.
		• ENSURE Steam Generator 1-03 Atmospheric Relief Valve – CLOSED.
		• ENSURE Main Steam Line Drip Pot Isolation Valve – CLOSED.
	RO	CHECK CST Level – GREATER THAN 10%.
Examiner Note: **EOP-2.0A, Attachment 2 actions are performed outside of the Control Room.**		
	US/BOP	VERIFY Faulted Steam Generator 1-03 Break Outside Containment.
		• DIRECT performance of EOP-2.0A, Attachment 2.
	US/RO	CHECK Secondary Radiation:
		• REQUEST periodic activity samples of all Steam Generators.

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	17	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior
		• CHECK available Secondary Radiation Monitors – NORMAL.
	US/RO	CHECK if ECCS Flow to Should Be Reduced:
		• VERIFY Secondary heat sink:
		• DETERMINE Total AFW Flow to intact SGs > 460 GPM.
		• DETERMINE Narrow Range Level in SGs 1-01, 1-02, & 1-04 > 43%.
		• VERIFY RCS subcooling > 25ºF.
		• VERIFY RCS pressure – STABLE <u>OR</u> INCREASING.
		• VERIFY PRZR level > 13%.
+20 min	US	DETERMINE ECCS flow should be reduced and TRANSITION to EOS-1.1A, Safety Injection Termination, Step 1.
<u>Examiner Note</u>: EOS-1.1A, Safety Injection Termination, steps begin here.		
<u>Examiner Note</u>: The following six (6) steps are performed per EOS-1.1A, Attachment 1.D.		
	BOP	[1.D] PLACE both Diesel EMER START/STOP Handswitches in START.
	BOP	[1.D] RESET SI.
	BOP	[1.D] RESET SI Sequencers.
	BOP	[1.D] RESET Containment Isolation Phase A and B.
	BOP	[1.D] RESET Containment Spray Signal.
	BOP/RO	[1.D] ESTABLISH Instrument Air and Nitrogen to Containment.
		• OPEN 1-HS-3487, Containment Instrument Air Isolation Valve.
	RO	STOP Train B CCP and PLACE in Standby.

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	18	of	22
Event Description:	Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
	US/RO	CHECK RCS Pressure – STABLE OR INCREASING.
Examiner Note: **The following two (2) steps are performed per EOS-1.1A, Attachment 1.J.**		
	RO	[1.J] ISOLATE CCP Injection Line Flow Path:
		• VERIFY CCP – SUCTION ALIGNED TO RWST.
		• ALIGN CCP Miniflow Valves:
		• OPEN 1/1-8110 and 1/1-8111, CCP Miniflow Valves.
		• CLOSE 1/1-8511A and 1/1-8511B, CCP Alternate Miniflow Isolation Valves.
		• PLACE Charging Flow Control Valve in MANUAL and 35% demand.
		• CLOSE 1/1-8801A and 1/1-8801B, CCP Injection Line Isolation Valves.
+30 min	RO	[1.J] ESTABLISH Charging Flow Path:
		• OPEN 1/1-8105 and 1/1-8106, Charging Line Isolation Valves.
		• ADJUST Charging Flow Control Valve to establish Charging flow.
		• ADJUST RCP seal flow to maintain between 6 gpm and 13 gpm.
When EOS-1.1A, Safety Injection Termination, Attachment 1.J is complete, TERMINATE the scenario.		

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	19	of	22
Event Description:	Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure								

Time	Position	Applicant's Actions or Behavior
Examiner Note: These steps are performed by the BOP per EOP-0.0A, Attachment 2.		
	BOP	VERIFY SSW Alignment:
		• VERIFY SSW Pumps – RUNNING.
		• VERIFY Diesel Generator Cooler SSW return flow.
	BOP	VERIFY Safety Injection Pumps – RUNNING.
	BOP	VERIFY Containment Isolation Phase A.
	BOP	VERIFY Phase A Actuation.
	BOP	VERIFY Containment Ventilation Isolation.
	BOP	VERIFY CCW Pumps – RUNNING.
	BOP	VERIFY RHR Pumps – RUNNING.
	BOP	• DETERMINE RHR Pump 1-01 failed to start and MANUALLY START RHR Pump 1-01.
	BOP	VERIFY Proper CVCS Alignment:
		• VERIFY both CCPs – RUNNING.
		• VERIFY Letdown Relief Valve isolation:
		• DETERMINE Letdown Orifice Isolation Valves – CLOSED.
		• DETERMINE Letdown Isolation Valves1/1-LCV-459 & 1/1-LCV-460 – CLOSED.
	BOP	VERIFY ECCS flow:
		• VERIFY CCP SI flow indicator.
		• VERIFY RCS pressure < 1800 PSIG.
		• VERIFY SI Pumps discharge flow indicator.

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	20	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior
		• VERIFY RCS pressure > 325 PSIG.
	BOP	VERIFY Feedwater Isolation Complete:
		• VERIFY Feedwater Isolation Valves CLOSED.
		• VERIFY Feedwater Isolation Bypass Valves CLOSED.
		• VERIFY Feedwater Bypass Control Valves CLOSED.
		• VERIFY Feedwater Control Valves CLOSED.
	BOP	VERIFY both Diesel Generators – RUNNING.
	BOP	VERIFY Monitor Lights For SI Load Shedding – LIT.
	BOP	VERIFY Proper SI alignment per MLB light indication.
	BOP	VERIFY Components Properly Aligned per Table 1.

Location	Equipment	Description	Condition
CB-03	X-HS-5534	H2 PRG SPLY FN 4	STOPPED
CB-03	X-HS-5532	H2 PRG SPLY FN 3	STOPPED
CB-04	1/1-8716A	RHRP 1 XTIE VLV	OPEN
CB-04	1/1-8716B	RHRP 2 XTIE VLV	OPEN
CB-06	1/1-8153	XS LTDN ISOL VLV	CLOSED
CB-06	1/1-8154	XS LTDN ISOL VLV	CLOSED
CB-07	1/1-RTBAL	RX TRIP BKR	OPEN
CB-07	1/1-RTBBL	RX TRIP BKR	OPEN
CB-07	1/1-BBAL	RX TRIP BYP BKR	OPEN/DEENERGIZED
CB-07	1/1-BBBL	RX TRIP BYP BKR	OPEN/DEENERGIZED
CB-08	1-HS-2397A	SG 1 BLDN HELB ISOL VLV	CLOSED
CB-08	1-HS-2398A	SG 2 BLDN HELB ISOL VLV	CLOSED
CB-08	1-HS-2399A	SG 3 BLDN HELB ISOL VLV	CLOSED
CB-08	1-HS-2400A	SG 4 BLDN HELB ISOL VLV	CLOSED
CB-08	1-HS-2111C	FWPT A TRIP	TRIPPED
CB-08	1-HS-2112C	FWPT B TRIP	TRIPPED

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	21 of 22
Event Description:	Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure						

Time	Position	Applicant's Actions or Behavior

	CB-09	1-HS-2490	CNDS XFER PUMP	STOPPED (MCC deenergized on SI)
	CV-01	X-HS-6181	PRI PLT SPLY FN 17 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6188	PRI PLT SPLY FN 18 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6195	PRI PLT SPLY FN 19 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6202	PRI PLT SPLY FN 20 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6209	PRI PLT SPLY FN 21 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6216	PRI PLT SPLY FN 22 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6223	PRI PLT SPLY FN 23 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6230	PRI PLT SPLY FN 24 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-3631	UPS & DISTR RM A/C FN 1 & BSTR FN 42	STARTED
	CV-01	X-HS-3632	UPS & DISTR RM A/C FN 2 & BSTR FN 43	STARTED
	CV-01	1-HS-5600	ELEC AREA EXH FN 1	STOPPED/DEENERGIZED
	CV-01	1-HS-5601	ELEC AREA EXH FN 2	STOPPED/DEENERGIZED
	CV-01	1-HS-5602	MS & FW PIPE AREA EXH FN 3 & EXH DMPR	STOPPED/DEENERGIZED
	CV-01	1-HS-5603	MS & FW PIPE AREA EXH FN 4 & EXH DMPR	STOPPED/DEENERGIZED
	CV-01	1-HS-5618	MS & FW PIPE AREA SPLY FN 17	STOPPED/DEENERGIZED
	CV-01	1-HS-5620	MS & FW PIPE AREA SPLY FN 18	STOPPED/DEENERGIZED
	CV-03	X-HS-5855	CR EXH FN 1	STOPPED/DEENERGIZED
	CV-03	X-HS-5856	CR EXH FN 2	STOPPED/DEENERGIZED
	CV-03	X-HS-5731	SFP EXH FN 33	STOPPED/DEENERGIZED
	CV-03	X-HS-5733	SFP EXH FN 34	STOPPED/DEENERGIZED
	CV-03	X-HS-5727	SFP EXH FN 35	STOPPED/DEENERGIZED
	CV-03	X-HS-5729	SFP EXH FN 36	STOPPED/DEENERGIZED

Operating Test :	NRC	Scenario #	1	Event #	4, 5, 6, 7, 8, & 9	Page	22	of	22

Event Description: Main Steam Header Leak / Automatic And Manual Safety Injection Failure / Main Steam Isolation Valve Failure / RHR Pump Start Failure

Time	Position	Applicant's Actions or Behavior

<u>Examiner Note</u>: The next four (4) steps would be performed on Unit 2.

	CB-03	2-HS-5538	AIR PRG EXH ISOL DMPR	CLOSED
	CB-03	2-HS-5539	AIR PRG EXH ISOL DMPR	CLOSED
	CB-03	2-HS-5537	AIR PRG SPLY ISOL DMPR	CLOSED
	CB-03	2-HS-5536	AIR PRG SPLY ISOL DMPR	CLOSED

	BOP	NOTIFY Unit Supervisor attachment instructions complete and to IMPLEMENT FRGs as required.

Facility: CPNPP 1 & 2 Scenario No.: 2 Op Test No.: June 2011 NRC

Examiners: ______________ Operators: ______________

______________ ______________

______________ ______________

Initial Conditions:	- 72% power MOL - RCS Boron is 975 ppm by Chemistry sample.
Turnover:	Maintaining 72% power per Load Controller direction. Rod Control in AUTO.
Critical Tasks:	- Identify Excess Reactor Coolant System Leakage and Manually Trip Reactor Prior to Reaching 0% Pressurizer Level. - Identify and Isolate Flow from the Ruptured Steam Generator Prior to Commencing an Operator Induced Cooldown. - Initiate Cooldown of the Reactor Coolant System Prior to Exiting EOP-3.0A.

Event No.	Malf. No.	Event Type*	Event Description
1 +5 min		N (BOP, SRO)	Recirculate RWST using Containment Spray Pump (1-01).
2 +10 min	CV16A	I (RO, SRO)	Volume Control Tank Level Transmitter (LT-112) Failure low.
3 +15 min	MS13D	I (BOP, SRO)	Atmospheric Relief Valve (1-04) Fails Open due to Steam Pressure Transmitter (PT-2328) Failure.
4 +25 min	CV01B	C (RO, SRO) TS (SRO)	Centrifugal Charging Pump (1-01) Trip.
5 +45 min	SG01D	R (RO) N (BOP, SRO) TS (SRO)	Steam Generator (1-04) Tube Leak at 2.5 GPM (180 second ramp). Rapid Down Power Required.
6 +48 min	SG01D	M (RO, BOP, SRO)	Steam Generator (1-04) Tube Rupture at 500 GPM (180 second ramp).
7 +50 min	RP01	I (RO)	Automatic Reactor Trip Failure.
8 +55 min	RP09A RP09B	C (BOP)	Containment Isolation Phase A Train A and Train B Auto Actuation Failure.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	**Target Quantitative Attributes**
7	Total malfunctions (5-8)
2	Malfunctions after EOP entry (1-2)
3	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 #2

The crew will assume the watch at 72% power per IPO-003A, Power Operations. The Grid Controller has requested that power remain at this level due to transmission line overload until further notice.

The scenario begins with a recirculation of the Refueling Water Storage Tank per SOP-204A, Containment Spray System, following makeup to restore tank level. The Containment Spray Pump will remain operating during the scenario.

The next event is a low failure of the Volume Control Tank Level Transmitter. The crew will reference annunciator ALM-0061A-4.5, VCT LEVEL LO, and ABN-105, Chemical and Volume Control System Malfunction, and establish an Alternate Operating Mode for the Reactor Makeup System.

When conditions are stable, the Atmospheric Relief Valve (ARV) on Steam Generator 1-04 will fail open. This event is recognized by a Reactor power increase, ARV Controller indicating 100% demand, and a Plant Computer System alarm. The BOP will place the affected Controller in MANUAL and close the ARV. ABN-709, Steam Line Pressure Instrument Malfunction, may be referenced.

When plant parameters are stable, a loss of the running Centrifugal Charging Pump will occur. The crew will enter ABN-105, Chemical and Volume Control System Malfunction, and perform actions to immediately restore Charging flow. The SRO will refer to Technical Specifications.

The next event is a Steam Generator tube leak of ~2.5 GPM. Crew actions are per ABN-106, High Secondary Activity. Given the size of the leak, a rapid power reduction will be performed. The SRO will refer to Technical Specifications.

When Technical Specifications are referenced and power has been reduced from 3% to 5%, a Steam Generator Tube Rupture occurs and leakage rises to 500 GPM. Pressurizer pressure and level will lower uncontrollably and require a manual Reactor trip, initiation of Safety Injection, and entry into EOP-0.0A, Reactor Trip or Safety Injection. At Step 13, a transition to EOP-3.0A, Steam Generator Tube Rupture, will occur to isolate the ruptured Steam Generator. The event is complicated by a failure of Train A and B Containment Isolation Phase A.

The scenario is terminated when the ruptured Steam Generator is isolated, feedwater flow is properly aligned, and a Reactor Coolant System cooldown is initiated.

Risk Significance:

- Failure of risk important system prior to trip: Centrifugal Charging Pump Trip
- Risk significant core damage sequence: Steam Generator Tube Rupture
- Risk significant operator actions: Manually Trip Reactor

  Identify and Isolate Ruptured SG

  Manually Initiate Containment Isolation

  Cooldown and Depressurize the RCS

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC #35 and Event File for NRC Scenario #2.

EVENT	**TYPE**	**MALF #**	**DESCRIPTION**	**DEMAND VALUE**	**INITIATING PARAMETER**
SETUP		RP01	Automatic Reactor Trip failure	-	K0
		RP09A/B	Containment Isolation Train A/B actuation failure	-	K0
NOTE: Ensure Rod Control is in AUTO					
1		N/A	Recirculate the RWST	-	K0
2		CV16A	VCT Level Transmitter (LT-112) failure	0%	K2
2		CV16A	VCT Level Transmitter (LT-112) vented by I&C	DELETE	K2
3		MS13D	ARV (1-04) fails open due to PT-2328 failure	1300 psia	K3
4		CV01B	Centrifugal Charging Pump (1-01) trip	-	K4
	CVR05		CCP (1-01) Auxiliary Lube Oil Pump	OFF	K10
	CVR06		CCP (1-02) Auxiliary Lube Oil Pump	AUTO	K10
5		SG01D	Steam Generator #1 Tube Leak	2.5 gpm	K5 (180 sec. ramp)
6		SG01D	Steam Generator #1 Tube Rupture	500 gpm	MODIFY K5 (180 sec. ramp)
7		RP01	Automatic Reactor Trip failure	-	K0
8		RP09A/B	Containment Isolation Train A/B actuation failure	-	K0

<u>Booth Operator</u>: INITIALIZE to IC #35 and NRC Scenario #2 SETUP file.
ENSURE all Simulator Annunciator Alarms are ACTIVE.
ENSURE all Control Board Tags are removed.
ENSURE Operator Aid Tags reflect current boron conditions.
ENSURE Rod Bank Update (RBU) is performed.
ENSURE Turbine Load Rate set at 10 MWe/minute.
ENSURE 60/90 buttons DEPRESSED on ASD.
RE-SCALE Main Control Board CRT on CB-07 for 72% power.
ENSURE Reactivity Briefing Sheet printout provided with Turnover.
ENSURE procedures in progress are on SRO desk:

- **COPY of IPO-003A, Power Operations, Section 5.5, Operating at Constant Turbine Load.**

ENSURE Control Rods are in AUTO with Bank D at 178 steps.

Control Room Annunciators in Alarm:

PCIP-1.1 – SR TRN A RX TRIP BLK
PCIP-1.2 – IR TRN A RX TRIP BLK
PCIP-1.4 – CNDSR AVAIL STM DMP ARMED C-9
PCIP-1.6 – RX ≥ 10% PWR P-10
PCIP-2.1 – SR TRN B RX TRIP BLK
PCIP-2.2 – IR TRN B RX TRIP BLK
PCIP-2.5 – SR RX TRIP BLK PERM P-6
PCIP-3.2 – PR TRN A LO SETPT RX TRIP BLK
PCIP-4.2 – PR TRN B LO SETPT RX TRIP BLK

Operating Test : NRC Scenario # 2 Event # 1 Page 5 of 23

Event Description: Recirculate the Refueling Water Storage Tank

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: ENSURE Simulator in RUN when crew is ready to assume the watch.		
+1 min	US	DIRECT performance of SOP-204A, Containment Spray System, Section 5.1.3, Recirculation Through the Recirculation Header.
	BOP	ENSURE the system is in STANDBY per Section 5.1.1.
	BOP	VERIFY Train A Chemical Additive Tank Discharge Valve is CLOSED.
		• 1-HS-4754, CHEM ADD TK DISCH VLV, Train A.
	BOP	INITIATE a trend of CSP 1-01 parameters on the Plant Computer.
	BOP	VERIFY CSP 1-01 Recirculation Valve is OPEN.
		• 1-HS-4772-1, CSP 1 RECIRC VLV.
	BOP	START Containment Spray Pump 1-01.
		• 1-HS-4764, CSP 1.
+5 min	BOP	MONITOR Containment Spray Pump parameters.
When recirculation has been started, or at Lead Examiner discretion, PROCEED to Event 2.		

<table>
<tr><td>Operating Test :</td><td>NRC</td><td>Scenario #</td><td>2</td><td>Event #</td><td>2</td><td>Page</td><td>6</td><td>of</td><td>23</td></tr>
<tr><td>Event Description:</td><td colspan="9">Volume Control Tank Level Transmitter Failure</td></tr>
</table>

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td colspan="3"><u>Booth Operator</u>: When directed, EXECUTE Event 2.
- CV16A, Volume Control Tank (LT-112) fails low.</td></tr>
<tr><td colspan="3"><u>Indications Available</u>:
6A-3.5 – VCT LVL LO
6A-4.5 – VCT LVL LO-LO
1-LI-112A – VCT LVL level indication fails low</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td>+1 min</td><td>RO</td><td>RESPOND to Annunciator Alarm Procedures.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>RECOGNIZE VCT level transmitter (LT-112) failed low.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>US</td><td>DIRECT performance of ALM-0061A, 1-ALB-6A, Window 4.5 - VCT LVL LO-LO.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td colspan="3"><u>Examiner Note</u>: The following steps are from 1-ALB-6A, Window 5.5 - VCT LVL LO-LO.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>MONITOR VCT level on 1-LI-112A, VCT LVL and 1-LI-185, VCT LVL.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>VERIFY 1-PI-115, VCT PRESS is approximately 30 psig.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>CHECK 1-LT-112, CVCS VCT Level Transmitter for malfunction.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>STOP Auto Makeup; PLACE 1/1-MU, RCS MU MAN ACT in STOP.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>REDUCE VCT level to between 46% and 56%.</td></tr>
<tr><td></td><td></td><td>• If necessary, PLACE 1/1-LCV-112A, VCT LVL CTRL VLV in HUT.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>RO</td><td>ENSURE 1-LI-185, VCT LVL and 1-PI-115, VCT PRESS are both lowering.</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td></td><td>US</td><td>DIRECT performance of ABN-105, Chemical and Volume Control System Malfunction, Section 6.0.</td></tr>
<tr><td colspan="3"></td></tr>
</table>

Operating Test :	NRC	Scenario #	2	Event #	2	Page	7	of 23
Event Description:	Volume Control Tank Level Transmitter Failure							

Time	Position	Applicant's Actions or Behavior
Booth Operator: When maintenance is contacted, DELETE malfunction CV16A and REPORT I&C vented the transmitter and it appears to be operating normally.		
	RO	PLACE 1/1-MU, RCS MU MAN ACT in AUTO.
+5 min	RO	VERIFY Automatic Operating Mode in service per SOP-104A, Reactor Makeup and Chemical Control System.
When the VCT level control is restored, or at Lead Examiner discretion, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	2	Event #	3	Page	8	of	23

Event Description: Steam Pressure Control Channel Fails High

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>: When directed, EXECUTE Event 3. -MS13D, SG 1-04 Steam Pressure Channel (PT-2328) fails high.		
<u>Indications Available</u>: 1-PI-2328, MSL 4 PRESS failed high 1-ZL-2328 SG 4 ATMOS RLF VLV read OPEN light lit Y6704D Plant Computer alarm		
+1 min	BOP	RESPOND to Dynamic Alarm Display (DAD) Alarm.
	BOP	RECOGNIZE Steam Generator 1-04 Steam Pressure Transmitter (PT-2328) failed high.
	US	DIRECT performance of ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st-Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 2.0.
	BOP	DETERMINE Steam Generator Atmospheric Relief Valve - OPEN.
	US	DIRECT closing of Steam Generator 1-04, Atmospheric Relief Valve.
	BOP	PLACE 1-PK-2328, SG 4 ATMOS RLF VLV CTRL in MANUAL and 0% DEMAND to CLOSE Valve.
+5 min	US	NOTIFY Chemistry that a release has occurred.
When the Atmospheric Relief Valve is closed, or at Lead Examiner discretion, PROCEED to Event 4.		

Operating Test : NRC	Scenario # 2	Event # 4	Page 9 of 23				
Event Description: Centrifugal Charging Pump Trip							

Time	Position	Applicant's Actions or Behavior

Booth Operator: **When directed, EXECUTE Event 4.**
- CV01B, Centrifugal Charging Pump 1-01 trip.

Indications Available:

5A-1.6 – ANY RCP SEAL WTR INJ FLO LO
6A-1.7 – ANY CHG PMP OVRLOAD / TRIP
6A-3.4 – CHG FLO HI / LO
CCP 1 amber MISMATCH and white TRIP lights lit

+1 min	RO	RESPOND to Annunciator Procedure Alarms.
	RO	RECOGNIZE Charging Pump 1-01 trip.

Examiner Note: **The next step is an Initial Operator Action.**

	RO	START Centrifugal Charging Pump 1-02.
	US	DIRECT performance of ABN-105, CVCS Malfunction, Section 3.0.
	RO	VERIFY one Centrifugal Charging Pump running.
	RO	VERIFY Seal Injection Flow to each RCP between 6 gpm and 13 gpm.
	RO/BOP	VERIFY RCP parameters in normal operating range.
	RO	VERIFY PRZR level > 17% and rising.

Booth Operator: **When contacted about status of Centrifugal Charging Pump 1-01, REPORT Phase B 50/51 over current relays are tripped and an acrid odor is present.**

Booth Operator: **When contacted, EXECUTE remote functions CVR05 and CVR06 for the Centrifugal Charging Pump (1-01 & 1-02) Auxiliary Lube Oil Pumps.**

	RO	VERIFY RCS leakage normal.
		• DETERMINE PRZR level stable at or trending to program.

Operating Test :	NRC	Scenario #	2	Event #	4	Page	10	of 23

Event Description: Centrifugal Charging Pump Trip

Time	Position	Applicant's Actions or Behavior
		• DETERMINE Charging flow < 15 gpm above Letdown flow.

Operating Test :	NRC	Scenario #	2	Event #	4	Page	11	of	23
Event Description:	Centrifugal Charging Pump Trip								

Time	Position	Applicant's Actions or Behavior
	US	EVALUATE Technical Specifications.
		• LCO 3.5.2.A, ECCS - Operating.
		• CONDITION A - One train inoperable because of the inoperability of a centrifugal charging pump. • ACTION A.1 - Restore pump to OPERABLE status within 7 days.
+10 min	US	INITIATE a work request per STA-606.

When Technical Specifications are addressed, or at Lead Examiner discretion, PROCEED to Event 5.

Booth Operator: **When directed, EXECUTE Event 5.**
- SG01D, Steam Generator 1-04 Tube Leak at ~2.5 gpm.

Indications Available:

PC-11 – MSL-181 (1-RE-2328) is RED
PC-11 – N16-177 MSL #4 (1-RE-2328A) is RED
PC-11 – 182 COG (1-RE-2959) is RED (Condenser Off Gas is delayed)

Time	Position	Applicant's Actions or Behavior
+1 min	RO/BOP	RESPOND to Digital Radiation Monitoring System alarms.
	RO/BOP	RECOGNIZE radiation monitor alarms associated with Steam Generator 1-04.
	US	DIRECT performance of ABN-106, High Secondary Activity, Section 3.0.
	RO/BOP	DETERMINE Main Steam Line 1-04 radiation alarm 1-RE-2328 is RED on PC-11.
	RO/BOP	CORRELATE monitor readings to leak rate and rate of change as necessary.

Examiner Note: **Crew may implement the Reactivity Briefing Sheet for a Rapid Plant Shutdown within one (1) hour. This guidance includes a boration of ~ 700 gallons and a Main Turbine load reduction to 250 MWe at 10 MWe/min.**

Time	Position	Applicant's Actions or Behavior
	US	REDUCE power to ≤ 50% in 1 hour AND be in MODE 3 in the next 2 hours and GO TO Step 5b.
	RO	DETERMINE PRZR level is stable.
	BOP	ADJUST Steam Generator 1-04 Atmospheric Relief Controller setpoint to 1160 PSIG per TDM-501A/B.
	BOP	PLACE 1-HV-2452-1, TDAFW Pump Steam Supply Valve from SG 1-04 in PULLOUT.

Examiner Note: **Technical Specification LCO 3.4.13 is NOT addressed in ABN-106 and has been identified as a Procedure Enhancement.**

Operating Test :	NRC	Scenario #	2	Event #	5	Page	13	of 23
Event Description:	Steam Generator Tube Leak and Power Descension							

Time	Position	Applicant's Actions or Behavior
+10 min	US	EVALUATE Technical Specifications.
		• LCO 3.7.5.A, Auxiliary Feedwater System
		• CONDITION A - One steam supply valve to turbine driven AFW pump inoperable. • ACTION A.1 - Restore steam supply to OPERABLE status within 7 days.
		• LCO 3.4.13, RCS Operational Leakage
		• CONDITION B - Primary to secondary LEAKAGE not within limits. • ACTION B.1 - Be in MODE 3 within 6 hours. • ACTION B.2 - Be in MODE 5 within 36 hours.
	US	DIRECT load reduction to 250 MWe per IPO-003A, Power Operations, Section 5.6, Reducing Turbine Power from 100% to MODE 3.
	RO	INITIATE RCS boration per SOP-104A, Reactor Make-up and Chemical Control System.
	BOP	SET Turbine Load Rate Setpoint Controller as desired and Load Target to 250 MWe.
<u>Examiner Note</u>: Boration amount is determined per the Reactivity Briefing Sheet.		
	RO	PERFORM the following to COMMENCE RCS boration:
		• ENSURE 1/1-MU, RCS Makeup Manual Actuation is in STOP.
		• PLACE 43/1-MU, RCS Makeup Mode Select in BORATE.
		• SET 1-FK-110, BA Blender Flow Control to desired flowrate.
		• SET 1-FY-110B, BA Batch Flow counter for the desired number of gallons.
		• ENSURE 1/1-FCV-110A, Boric Acid Blender Flow Control Valve is in AUTO.
		• PLACE 1/1-MU, RCS Makeup Manual Actuation in START.
		• VERIFY 1/1-APBA1, Boric Acid Transfer Pump starts.
		• VERIFY 1/1-FCV-110A, Boric Acid Blender Flow Control Valve throttles to the preset flow rate.

Operating Test :	NRC	Scenario #	2	Event #	5	Page	14	of	23
Event Description:	Steam Generator Tube Leak and Power Descension								

Time	Position	Applicant's Actions or Behavior
		• VERIFY 1/1-FCV-110B, RCS Makeup to Charging Pump Suction Isolation Valve OPEN.
		• VERIFY 1-FR-110, Boric Acid Flow to Blender RED pen operating properly.
		• VERIFY 1-FY-110B, Batch Flow counter operating properly.
		• When desired amount of boric acid is added, PLACE 1/1-MU, RCS Makeup Manual Actuation in STOP.
		• FLUSH the blender with approximately 50 gallons makeup water when boration is complete.
	BOP	PERFORM the following to LOWER Turbine Load:
		• CHANGE Turbine Load Rate to ~10 MWe/min.
		• OPEN "Load Target" OSD.
		• SELECT blue bar and ENTER 250 MWe.
		• DEPRESS "Accept" then VERIFY value in blue bar is desired "Load Target" (magnitude and direction).
		• DEPRESS "Execute" then VERIFY "Load Target" changes to desired load.
+20 min	CREW	MONITOR load change.
When power has been reduced 3% to 5%, or at Lead Examiner discretion, PROCEED to Events 6, 7, and 8.		

Operating Test : NRC	Scenario # 2 Event # 6, 7, & 8 Page 15 of 23
Event Description:	Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior
<u>Booth Operator</u>:		**When directed, EXECUTE Events 6, 7, and 8. - SG01D, SG 1-04 Tube Rupture @ 500 gpm on 180 second ramp. - RP01, Automatic Reactor Trip failure. - RP09A/B, Containment Isolation Phase A Train A/B fails to auto actuate.**
<u>Indications Available</u>:		**6A-3.4 – CHRG FLO HI / LO 5C-1.2 – PRZR LVL DEV LO 5C-3.3 – PRZR PRESS LO BACKUP HTRS ON PC-11 – 178 MSL #4 (1-RE-2328) is RED Main Steam Line Radiation level rising Pressurizer pressure lowering**
+2 min	RO/BOP	RECOGNIZE Pressurizer level and pressure decreasing at an increasing rate.
	RO/BOP	RECOGNIZE PRZR pressure decreasing with Steam Line Radiation Monitors in alarm and steam / feed mismatch.
CRITICAL TASK STATEMENT		**Identify Excess Reactor Coolant System Leakage and Manually Trip Reactor Prior to Reaching 0% Pressurizer Level.**
CRITICAL TASK	RO	Manually INITIATE a Reactor Trip.
	US	DIRECT performance of EOP-0.0A, Reactor Trip or Safety Injection.
	RO	VERIFY Reactor Trip:
		• DETERMINE Reactor Trip Breakers – OPEN.
		• DETERMINE Neutron flux – DECREASING.
	RO	DETERMINE all Control Rod Position Rod Bottom Lights – ON.
	BOP	VERIFY Turbine Trip:
		• DETERMINE all HP Turbine Stop Valves – CLOSED.
	BOP	VERIFY Power to AC Safeguards Buses:

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	16	of 23
Event Description:	Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure							

Time	Position	Applicant's Actions or Behavior
		• DETERMINE both AC Safeguards Buses –ENERGIZED.
	RO	Manually INITIATE both Trains of Safety Injection.
Examiner Note:		**EOP-0.0A, Attachment 2 steps performed by the BOP are identified later in the scenario. Ensure CRITICAL TASK listed is performed during Attachment 2.**
	US/BOP	INITIATE Proper Safeguards Equipment Operation Per Attachment 2.
	RO	VERIFY AFW Alignment:
		• DETERMINE both MDAFW Pumps – RUNNING.
		• DETERMINE Turbine Driven AFW Pump – NOT RUNNING.
		• DETERMINE AFW total flow – GREATER THAN 460 GPM.
		• DETERMINE AFW valve alignment – PROPER ALIGNMENT.
	RO	DETERMINE Containment Spray NOT Required:
		• VERIFY 1-ALB-2B window 1-8, CS ACT NOT ILLUMINATED.
		• VERIFY 1-ALB-2B window 4-11, CNTMT ISOL PHASE B ACT NOT ILLUMINATED.
		• VERIFY Containment pressure < 18.0 PSIG.
		• VERIFY Containment Spray Heat Exchanger Out Valves – CLOSED.
		• VERIFY Containment Spray Pumps – RUNNING.
	RO	CHECK If Main Steam lines should be Isolated:
		• DETERMINE Containment pressure 0 PSIG and stable.
		• DETERMINE Main Steam pressure ~ 1000 PSIG and stable.
	RO	CHECK RCS Temperature:
		• DETERMINE RCS T_{AVE} – STABLE at OR trending to 557ºF.
	RO	CHECK PRZR Valve Status:
		• DETERMINE PRZR Safeties – CLOSED.

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	17 of 23

Event Description: Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior
		• DETERMINE Normal PRZR Spray Valves – CLOSED.
		• DETERMINE PORVs – CLOSED.
		• DETERMINE Power to both Block Valves – AVAILABLE.
		• VERIFY both PORV Block Valves – OPEN.
	RO	CHECK If RCPs Should Be Stopped:
		• DETERMINE ECCS Pumps – AT LEAST ONE RUNNING.
		• DETERMINE CCP Pump 1-02 and SI Pumps – RUNNING.
		• DETERMINE RCS subcooling – GREATER THAN 25ºF.
	US/RO	DETERMINE RCPs should remain running.
	US/RO	CHECK if any SG is Faulted:
		• DETERMINE pressure in all SGs – NORMAL.
	US/RO	CHECK if SG Tubes are Ruptured:
		• DETERMINE SG 1-04 is ruptured and TRANSITION to EOP-3.0A, Steam Generator Tube Rupture, Step 1.
Examiner Note: **EOP-3.0A, Steam Generator Tube Rupture steps begin here.**		
+15 min	US/RO	CHECK If RCPs Should Be Stopped:
		• OBSERVE ECCS Pumps – AT LEAST ONE RUNNING.
		• OBSERVE CCP 1-02 and both SI Pumps – RUNNING.
		• DETERMINE RCS subcooling – GREATER THAN 25ºF.
	US/RO	DETERMINE RCPs should remain RUNNING.
	US/BOP	DETERMINE Steam Generator 1-04 is ruptured.
		• OBSERVE increase in Steam Generator 1-04 narrow range level.
		• OBSERVE high radiation from Steam Generator 1-04 Main Steam line.

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	18	of 23

Event Description: Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior
CRITICAL TASK STATEMENT		**Identify and Isolate Flow from the Ruptured Steam Generator Prior to Commencing an Operator Induced Cooldown.**
CRITICAL TASK	RO/BOP	ISOLATE Flow From Ruptured Steam Generator 1-04:
		• VERIFY SG 1-04 Atmospheric Controller Setpoint in MANUAL & CLOSE.
		• CHECK SG 1-04 Atmospheric Relief Valve – CLOSED.
		• CLOSE SG 1-04 Main Steam Line Isolation Valve.
		• CLOSE SG 1-04 Drip Pot Isolation Valves.
		• VERIFY SG 1-04 TDAFW Pump Steam Supply Valve – CLOSED.
		• CLOSE SG 1-04 Blowdown Valves.
	RO/BOP	CHECK Ruptured SG 1-04 Level:
		• VERIFY narrow range level > 43%.
		• ISOLATE AFW flow to SG 1-04.
	RO/BOP	VERIFY SG 1-04 Pressure > 420 PSIG.
<u>EOP-3.0A Caution</u>:		**If RCPs are NOT running, the following steps may cause a false INTEGRITY STATUS TREE (FRP) indication for the ruptured loop. Disregard ruptured loop Cold Leg Wide Range Temperature indication until after performing Step 32.**
CRITICAL TASK STATEMENT		**Initiate Cooldown of the Reactor Coolant System Prior to Exiting EOP-3.0A.**
CRITICAL TASK	RO/BOP	INITIATE RCS Cooldown using Steam Dump System.
	RO/BOP	When PRZR pressure decreases to less than 1960 psig, BLOCK the Low Steam Line Pressure SI Signal.
		• PLACE 1/1-SLS-1RBA and 1/1-SLS-1RBB, Main Steam Line Isolation Safety Injection Reset / Block in BLOCK position.

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	19	of 23
Event Description:	Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure							

Time	Position	Applicant's Actions or Behavior
	US	DETERMINE required Core Exit Thermocouple (CET) temperature from Table 1.
		• OBSERVED Steam Generator pressure = _____ PSIG
		• TARGET Core Exit Thermocouple (CET) temperature = _____ ºF
	BOP	DUMP steam to Condenser from intact SG(s) at maximum rate using the Steam Dump Valves.
	BOP	TRANSFER the Steam Dump Valves to STEAM PRESSURE Mode.
	BOP	ENSURE 1-PK-507, Steam Dump Pressure Controller in MANUAL and INCREASE demand.
	US/RO	DETERMINE required CET temperature is met.
	BOP	STOP RCS cooldown.
	RO/BOP	MAINTAIN required CET temperature.
+30 min	RO/BOP	CHECK Intact SG Levels:
		• VERIFY Narrow Range Level > 43%.
		• CONTROL AFW flow to maintain level between 50% and 60%.
When the Steam Generator is isolated and required CET temperature is met, TERMINATE the scenario.		

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	20	of	23

Event Description: Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior
Examiner Note:		**These steps are performed by the BOP as required per EOP-0.0A, Attachment 2. EOP-3.0A steps are identified later in the scenario.**
	BOP	VERIFY SSW Alignment:
		• VERIFY SSW Pumps – RUNNING.
		• VERIFY Diesel Generator Cooler SSW return flow.
	BOP	VERIFY Safety Injection Pumps – RUNNING.
CRITICAL TASK STATEMENT		**Manually Initiate Containment Isolation Phase A due to Failure to Automatically Actuate Prior to Exiting EOP-0.0.**
CRITICAL TASK	BOP	Manually INITIATE both Trains of Containment Isolation Phase A.
		• PLACE 1/1-CIPAA1 CNTMT ISOL – PHASE A / CNTMT VENT ISOL Switch in ACT position.
	BOP	VERIFY Containment Isolation Phase A.
	BOP	VERIFY Containment Ventilation Isolation.
	BOP	VERIFY CCW Pumps – RUNNING.
	BOP	VERIFY RHR Pumps – RUNNING.
	BOP	VERIFY Proper CVCS Alignment:
		• VERIFY CCP 1-02 – RUNNING.
		• VERIFY Letdown Relief Valve isolation:
		• DETERMINE Letdown Orifice Isolation Valves – CLOSED.
		• DETERMINE Letdown Isolation Valves 1/1-LCV-459 & 1/1-LCV-460 – CLOSED.

Operating Test :	NRC	Scenario #	2	Event #	6, 7, & 8	Page	21	of	23

Event Description: Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior
	BOP	VERIFY ECCS flow:
		• VERIFY CCP SI flow indicator.
		• VERIFY RCS pressure < 1800 PSIG.
		• VERIFY SI Pumps discharge flow indicator.
		• VERIFY RCS pressure > 325 PSIG.
	BOP	VERIFY Feedwater Isolation Complete:
		• VERIFY Feedwater Isolation Valves CLOSED.
		• VERIFY Feedwater Isolation Bypass Valves CLOSED.
		• VERIFY Feedwater Bypass Control Valves CLOSED.
		• VERIFY Feedwater Control Valves CLOSED.
	BOP	VERIFY both Diesel Generators – RUNNING.
	BOP	VERIFY Monitor Lights For SI Load Shedding – LIT.
	BOP	VERIFY Proper SI alignment per MLB light indication.
	BOP	VERIFY Components Properly Aligned per Table 1.

Location	Equipment	Description	Condition
CB-03	X-HS-5534	H2 PRG SPLY FN 4	STOPPED
CB-03	X-HS-5532	H2 PRG SPLY FN 3	STOPPED
CB-04	1/1-8716A	RHRP 1 XTIE VLV	OPEN
CB-04	1/1-8716B	RHRP 2 XTIE VLV	OPEN
CB-06	1/1-8153	XS LTDN ISOL VLV	CLOSED
CB-06	1/1-8154	XS LTDN ISOL VLV	CLOSED
CB-07	1/1-RTBAL	RX TRIP BKR	OPEN
CB-07	1/1-RTBBL	RX TRIP BKR	OPEN
CB-07	1/1-BBAL	RX TRIP BYP BKR	OPEN/DEENERGIZED
CB-07	1/1-BBBL	RX TRIP BYP BKR	OPEN/DEENERGIZED
CB-08	1-HS-2397A	SG 1 BLDN HELB ISOL VLV	CLOSED

	CB-08	1-HS-2398A	SG 2 BLDN HELB ISOL VLV	CLOSED
	CB-08	1-HS-2399A	SG 3 BLDN HELB ISOL VLV	CLOSED
	CB-08	1-HS-2400A	SG 4 BLDN HELB ISOL VLV	CLOSED
	CB-08	1-HS-2111C	FWPT A TRIP	TRIPPED
	CB-08	1-HS-2112C	FWPT B TRIP	TRIPPED
	CB-09	1-HS-2490	CNDS XFER PUMP	STOPPED (MCC deenergized on SI)
	CV-01	X-HS-6181	PRI PLT SPLY FN 17 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6188	PRI PLT SPLY FN 18 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6195	PRI PLT SPLY FN 19 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6202	PRI PLT SPLY FN 20 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6209	PRI PLT SPLY FN 21 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6216	PRI PLT SPLY FN 22 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6223	PRI PLT SPLY FN 23 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-6230	PRI PLT SPLY FN 24 & INTK DMPR	STOPPED/DEENERGIZED
	CV-01	X-HS-3631	UPS & DISTR RM A/C FN 1 & BSTR FN 42	STARTED
	CV-01	X-HS-3632	UPS & DISTR RM A/C FN 2 & BSTR FN 43	STARTED
	CV-01	1-HS-5600	ELEC AREA EXH FN 1	STOPPED/DEENERGIZED
	CV-01	1-HS-5601	ELEC AREA EXH FN 2	STOPPED/DEENERGIZED
	CV-01	1-HS-5602	MS & FW PIPE AREA EXH FN 3 & EXH DMPR	STOPPED/DEENERGIZED
	CV-01	1-HS-5603	MS & FW PIPE AREA EXH FN 4 & EXH DMPR	STOPPED/DEENERGIZED
	CV-01	1-HS-5618	MS & FW PIPE AREA SPLY FN 17	STOPPED/DEENERGIZED
	CV-01	1-HS-5620	MS & FW PIPE AREA SPLY FN 18	STOPPED/DEENERGIZED
	CV-03	X-HS-5855	CR EXH FN 1	STOPPED/DEENERGIZED
	CV-03	X-HS-5856	CR EXH FN 2	STOPPED/DEENERGIZED

Operating Test : NRC Scenario # 2 Event # 6, 7, & 8 Page 23 of 23

Event Description: Steam Generator Tube Rupture / Automatic Reactor Trip Failure / Train A & B Containment Isolation Phase A Actuation Failure

Time	Position	Applicant's Actions or Behavior			
		CV-03	X-HS-5731	SFP EXH FN 33	STOPPED/DEENERGIZED
		CV-03	X-HS-5733	SFP EXH FN 34	STOPPED/DEENERGIZED
		CV-03	X-HS-5727	SFP EXH FN 35	STOPPED/DEENERGIZED
		CV-03	X-HS-5729	SFP EXH FN 36	STOPPED/DEENERGIZED

Examiner Note: The next four (4) steps would be performed on Unit 2.

Time	Position	Applicant's Actions or Behavior			
		CB-03	2-HS-5538	AIR PRG EXH ISOL DMPR	CLOSED
		CB-03	2-HS-5539	AIR PRG EXH ISOL DMPR	CLOSED
		CB-03	2-HS-5537	AIR PRG SPLY ISOL DMPR	CLOSED
		CB-03	2-HS-5536	AIR PRG SPLY ISOL DMPR	CLOSED
	BOP	NOTIFY Unit Supervisor attachment instructions complete and to IMPLEMENT FRGs as required.			

Facility: CPNPP 1 & 2 Scenario No.: 3 Op Test No.: June 2011 NRC

Examiners: ____________ Operators: ____________

Initial Conditions:

- ~3% power BOL - RCS Boron is 1659 ppm by Chemistry sample.
- Steam Dump System in service for RCS Temperature Control.

Turnover: Raise Reactor Power from 3% to 8% in preparation for Turbine Startup.

Critical Tasks:

- Manually Trip the Reactor Upon Failure of Reactor to Trip Prior to Exiting FRS-0.1A.
- Emergency Borate due to Anticipated Transient Without Trip Prior to Exiting FRS-0.1A.

Event No.	Malf. No.	Event Type*	Event Description
1 +15 min		N (BOP, SRO)	Transfer from Auxiliary Feedwater System to Main Feedwater System and Place Feedwater Bypass Control Valves in AUTO.
2 +30 min		R (RO) N (BOP, SRO)	Raise power to 8% in preparation for synchronizing the Main Generator to the electrical grid.
3 +40 min	RX08A	I (RO, SRO) TS (SRO)	Pressurizer Pressure Transmitter (PT-455) fails low.
4 +50 min	RX04D	I (BOP, SRO) TS (SRO)	Steam Generator (1-04) Level Transmitter (LT-554) Fails Low.
5 +51 min	RC07A	M (RO, BOP, SRO)	Reactor Coolant Pump (1-01) Trip.
6 +52 min	RP13C	I (RO)	Manual Reactor Trip Failure (both). Commence Inserting Control Rods at ≥ 48 steps/minute.
7 +52 min	OVRDE	C (BOP)	Bus Breaker CS-1B4-1 Fails to Open. Initiate Emergency Boration.
8 +62 min	CV01B	C (RO)	Centrifugal Charging Pump (1-01) Trip after Transition Brief to EOS-0.1A.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
6	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
2	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO SUMMARY NRC #3

The crew will assume the watch with power at approximately 3% per IPO-002A, Plant Startup from Hot Standby. The crew will transfer Feedwater flow from the Auxiliary Feedwater System to the Main Feedwater System in preparation for raising power to 8%. This is followed by entry into SOP-304A, Auxiliary Feedwater System, Section 5.2, Shutdown and Standby of the Auxiliary Feedwater System.

When transfer of Feedwater has been completed, the crew will enter IPO-003A, Power Operations, Section 5.1, Warmup and Synchronization of the Turbine Generator and perform a power ascension using the Rod Control and Steam Dump Systems.

When power has been raised 3% to 5%, a Pressurizer Pressure Channel will fail low. Response is per ABN-705, Pressurizer Pressure Malfunction, Section 2.0, to ensure Pressurizer Heaters are controlled and Power Operated Relief Valves remain closed. The SRO will refer to Technical Specifications.

The next event is a Steam Generator Level Transmitter failure. Actions are per ABN-710, Steam Generator Level Instrumentation Malfunction. The BOP will be required to take manual control of the Feedwater Bypass Control Valve and then select an alternate controlling channel to return the Feedwater System to automatic control. The SRO will refer to Technical Specifications.

When Technical Specifications are addressed, a Reactor Coolant Pump will trip. Entry into ABN-101, Reactor Coolant Pump Trip / Malfunction, may be performed. Although an automatic Reactor trip is not generated, the RO should recognize the requirement to manually trip the Reactor. An attempt will be made to manually trip the Reactor via the normal Trip Switches and by deenergizing both buses supplying the Control Element Drive Mechanism Motor Generators. Once it is determined that neither of these methods have been successful, the crew will transition from EOP-0.0A, Reactor Trip or Safety Injection, to FRS-0.1A, Response to Nuclear Power Generation/ATWT.

When FRS-0.1A is entered, Control Rods are manually inserted, emergency boration is initiated, and operators are dispatched to locally trip the Reactor. The crew then transitions from FRS-0.1A to EOP-0.0A. After it is determined that Safety Injection is not required the crew will enter EOS-0.1A, Reactor Trip Response and perform actions to restore Charging and Letdown flow. When in EOS-0.1A, a Centrifugal Charging Pump will trip and must be restarted to continue emergency boration.

The scenario is terminated when IPO-009A, Plant Equipment Shutdown Following a Trip, is referenced while in EOS-0.1A.

Risk Significance:

• Risk significant core damage sequence:	Manual Reactor Trip Failure
	Anticipated Transient Without Trip (ATWT)
• Risk significant operator actions:	Manually Trip Reactor Due to RCP Trip
	Manually Insert Control Rods During ATWT
	Emergency Borate Due to ATWT
	Centrifugal Charging Pump Trip

BOOTH OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC #34 and Event File for NRC Scenario #3.

EVENT	TYPE	MALF #	DESCRIPTION	DEMAND VALUE	INITIATING PARAMETER
SETUP		RP13C	Manual Reactor trip failure	FAIL	K0
	OVRDE		Bus Breaker CS-1B4-1 Fails to Open	OPEN	K0
1		-	Transfer from AFW to Main Feedwater System	-	N/A
1	FWR106		PV-2242 FWP SUCT HDR PRESS override	NORMAL	-
2		-	Raise power to 8%	-	N/A
2	MSR04		1MS-451 & 1MS-454 MSR A & B Auxiliary Steam Isolation Valve	-	K10
3		RX08A	Pressurizer Pressure Transmitter (PT-455) failure	1700 psig	K3
4		RX04D	SG (1-04) Level Transmitter (LT-554) failure	0%	K4
5		RC07A	Reactor Coolant Pump (1-01) trip	TRIP	K5
6		RP13C	Manual Reactor trip failure	FAIL	K0
7		OVRDE	Bus Breaker CS-1B4-1 failure	CLOSE	K0
7	RPR112		Reactor Trip Breaker Train A	OPEN	K7
7	RPR113		Reactor Trip Breaker Train B	OPEN	K7
8		CV01B	Centrifugal Charging Pump (1-01) trip **(NOTE: 1)**	TRIP	K8
NOTE 1: CCP 1-01 is tripped after transition brief in EOS-0.1 AND Pressurizer level > 10%.					

Booth Operator: **INITIALIZE to IC #34 and NRC Scenario #3 SETUP file.**
ENSURE all Simulator Annunciator Alarms are ACTIVE.
ENSURE all Control Board Tags are removed.
ENSURE Operator Aid Tags reflect current boron conditions.
ENSURE Control Rods are in MANUAL with Control Rod Bank C @ 228 steps and Bank D @ 115 steps.
ENSURE Rod Bank Update (RBU) is performed.
REMOVE N-16 detectors from POLL on PC-11.
ENSURE 1-HS-2484 & 1-HS-2485, Condensate Storage Tank Isolation Valves are OPEN.
SET Plant Computer screen for MODE 2.
ENSURE Reactivity Briefing Sheet printout provided with Turnover.
PLACE Plant Computer, right hand RO and US Computer screens for MODE 2.
PLACE Group Display LPTDIFF on the BOP Desktop Computer.
ENSURE all PRZR Heaters energized.
ENSURE procedures in progress are on SRO desk:
- COPY of IPO-002A, Plant Startup from Hot Standby, INITIALED to Step 5.4.8.
- COPY of SOP-304A, Auxiliary Feedwater System, Section 5.2, with N/As as required in preparation for placing the AFW System in Standby.
- COPY of IPO-003A, Power Operations, Section 5.1, Warmup and Synchronization of the Turbine Generator, INITIALED as appropriate.

Significant Control Room Annunciators in Alarm:

PCIP-1.1 – SR TRN A RX TRIP BLK
PCIP-1.3 – AMSAC BLK TURB < 40% PWR C-20
PCIP-1.4 – CNDNSR AVAIL STM DUMP ARMED C-9
PCIP-1.7 – RX ≤ 50% PWR TURB TRIP PERM P-9
PCIP-2.1 – SR TRN B RX TRIP BLK
PCIP-2.4 – LO TURB PWR ROD WTHDRWL BLK C-5
PCIP-2.5 – SR RX TRIP BLK PERM P-6
PCIP-3.5 – RX & TURB ≤ 10% PWR P-7
PCIP-4.5 – RX ≤ 48% PWR 3-LOOP FLO PERM P-8
PCIP-4.6 – TURB ≤ 10% PWR P-13
6D-1.1 – SR HI VOLT FAIL
7B-4.8 – FWP A/B RECIRC VLV NOT CLOSED
8A-1.3 – FWPT B TRIP
8A-1.10 – 1 OF 4 TURB STOP VLV CLOSE
Numerous 9A Feedwater alarms

Operating Test :	NRC	Scenario #	3	Event #	1	Page 5	of 20
Event Description:	Transfer from the Auxiliary Feedwater System to the Main Feedwater System / Shutdown AFW System						

Time	Position	Applicant's Actions or Behavior
Booth Operator: ENSURE Simulator in RUN when crew is ready to assume the watch.		
+1 min	US	DIRECT performance of IPO-002A, Plant Startup from Hot Standby, Step 5.4.10.
	BOP	ENSURE all Steam Generator Feedwater Flow Control Valve Controllers are in MANUAL and the valves are CLOSED.
	BOP	ENSURE all Steam Generator Feedwater Bypass Control Valve Controllers are in MANUAL and 0% demand.
	BOP	ENSURE the Steam Generator Feedwater Bypass Control Valve handswitches are in AUTO and the valves are CLOSED:
		• 1-HS-2162, SG 1 FW BYP & CTRL VLV.
		• 1-HS-2163, SG 2 FW BYP & CTRL VLV.
		• 1-HS-2164, SG 3 FW BYP & CTRL VLV.
		• 1-HS-2165, SG 4 FW BYP & CTRL VLV.
	BOP	RESET the Feedwater Isolation signal by DEPRESSING pushbuttons:
		• 1/1-FWIRA, FW ISOL RESET.
		• 1/1-FWIRB, FW ISOL RESET.
	BOP	VERIFY alarm 1-ALB-8A, 1.13, LO T_{AVE} & RX TRIP FW ISOL ACT is OFF.
IPO-002A Note: When the Feedwater Bypass Control Valves are open, the SG will be fed by two sources, which will require the operator to manipulate Auxiliary Feedwater flow to prevent SG level oscillations. The following three steps should be performed simultaneously in order to maintain proper SG level.		
	BOP	Throttle OPEN Feedwater Bypass Control Valve Controllers in MANUAL:
		• 1- LK-550, SG 1 FW BYP CTRL.
		• 1- LK-560, SG 2 FW BYP CTRL.
		• 1- LK-570, SG 3 FW BYP CTRL.

Time	Position	Applicant's Actions or Behavior
		• 1- LK-580, SG 4 FW BYP CTRL.
	BOP	VERIFY flow to each Steam Generator through the Main Feed line:
		• 1- FI-510A, SG 1 FW FLO.
		• 1- FI-511A, SG 1 FW FLO.
		• 1- FI-520A, SG 2 FW FLO.
		• 1- FI-521A, SG 2 FW FLO.
		• 1- FI-530A, SG 3 FW FLO.
		• 1- FI-531A, SG 3 FW FLO.
		• 1- FI-540A, SG 4 FW FLO.
		• 1- FI-541A, SG 4 FW FLO.
	BOP	Throttle CLOSED the Auxiliary Feedwater Flow Control Valve Controllers:
		• 1- FK-2453A, MD AFWP 1 SG FLO 1 CTRL.
		• 1- FK-2453B, MD AFWP 1 SG FLO 2 CTRL.
		• 1- FK-2454A, MD AFWP 2 SG FLO 3 CTRL.
		• 1- FK-2454B, MD AFWP 2 SG FLO 4 CTRL.
<u>IPO-002A Note</u>: The SG level control system is selected to the preferred channels to preserve the 2/3 coincidence on high level Turbine Trip in the event the alternate level control channel fails.		
	BOP	ENSURE Steam Generator Level Control Switches are in the following positions:
		• 1- LS-519C, SG 1 LVL CHAN SELECT - LQY-551.
		• 1- LS-529C, SG 2 LVL CHAN SELECT - LQY-552.
		• 1- LS-539C, SG 3 LVL CHAN SELECT - LQY-553.
		• 1- LS-549C, SG 4 LVL CHAN SELECT - LQY-554.
	BOP	VERIFY Main Feedwater flow is sufficient to maintain Steam Generator level and TERMINATE AFW flow in PLACE in STANDBY per SOP-304A.

Operating Test :	NRC	Scenario #	3	Event #	1	Page	7	of	20

Event Description: Transfer from the Auxiliary Feedwater System to the Main Feedwater System / Shutdown AFW System

Time	Position	Applicant's Actions or Behavior
	US/BOP	CONTACT Radwaste Operations to PLACE the Condensate Polishing Control System FWP Suction Header Pressure Low Trip Override Enabled Circuit to NORMAL per RWS-109A.
Booth Operator:		**When contacted, EXECUTE remote function FWR106, PV-2242 FWP SUCT HDR PRESS OVERRIDE.**
	BOP	PLACE Feedwater Bypass Control Valve Controllers in AUTO:
		• 1- LK-550, SG 1 FW BYP CTRL.
		• 1- LK-560, SG 2 FW BYP CTRL.
		• 1- LK-570, SG 3 FW BYP CTRL.
		• 1- LK-580, SG 4 FW BYP CTRL.
Examiner Note:		**The following steps are from SOP-304A, Auxiliary Feedwater System.**
	US	DIRECT performance of SOP-304A, Auxiliary Feedwater System.
	BOP	ENSURE both Motor Driven AFW Pump handswitches in AUTO after STOP.
		• 1-HS-2450A, MD AFWP 1.
		• 1-HS-2451A, MD AFWP 2.
	BOP	PLACE AFW Flow Control Valve Controllers at 100% output and MANUAL:
		• 1- FK-2453A, MD AFWP 1 SG 1 FLO CTRL.
		• 1- FK-2453B, MD AFWP 1 SG 2 FLO CTRL.
		• 1- FK-2454A, MD AFWP 2 SG 3 FLO CTRL.
		• 1- FK-2454B, MD AFWP 2 SG 4 FLO CTRL.
	US/BOP	VERIFY proper Flow Control and Isolation Valve position per OPT-206A.
Floor Cue:		**If requested, REPORT another operator will perform the actions of OPT-206A, AFW System.**

Operating Test : NRC Scenario # 3 Event # 1 Page 8 of 20

Event Description: Transfer from the Auxiliary Feedwater System to the Main Feedwater System / Shutdown AFW System

Time	Position	Applicant's Actions or Behavior
+15 min	US/BOP	MONITOR the temperature of the Auxiliary Feedwater System for approximately 30 minutes to detect any Steam Generator back leakage.
When the AFW System alignment is complete, or at Lead Examiner discretion, PROCEED to Event 2.		

Operating Test :	NRC	Scenario #	3	Event #	2	Page	9	of	20
Event Description:	Raise Reactor Power / Prepare Turbine for Operation								

Time	Position	Applicant's Actions or Behavior
Booth Operator: **MONITOR Simulator parameters while the crew transitions to IPO-003A.**		
+1 min	US	DIRECT performance of IPO-003A, Power Operations, Section 5.1, Warmup and Synchronization of the Turbine Generator STARTING at Step 5.1.3.
	BOP	OPEN Turbine Drain Valves.
		• 1-HS-2418, HP CTRL VLV 3/4 AFT SEAT DRN VLV.
		• 1-HS-2419, TURB SIDE XOVER DRN VLV.
		• 1-HS-2420, MSR SIDE XOVER DRN VLV.
	US	DETERMINE OPT-410A has been completed within the previous 31 days (already initialed).
	US	DETERMINE Moisture Separator Reheater pre-warming is complete per SOP-301A (already initialed).
	US	NOTIFY Chemistry and Radiation Protection if Reactor power will be increased greater than 15% in a one hour period (already initialed).
	US	Prior to increasing Reactor power above 10%, PERFORM Flow Control and Isolation Valve Position verification per OPT-206A to ensure each AFW flow control valve and isolation valve is fully open (already initialed).
	BOP	OPEN 1-HS-2611/12, FW HTR 5A & 6A/5B & 6B BYP VLV.
	US	VERIFY the following annunciators are OFF (already initialed):
		• 1-ALB-9B, 3.9, EHC FLUID TEMP HI.
		• 1-ALB-9B, 5.6, TURB L/O TEMP HI.
	US	DETERMINE lube oil temperature is >95°F on TURB BRG TEMP RCDR 1 (already initialed).
	BOP	OPEN 1-HS-2417, HP CTRL VLV 1 ● 4 BEF SEAT DRN VLV.

Operating Test : NRC Scenario # 3 Event # 2 Page 10 of 20

Event Description: Raise Reactor Power / Prepare Turbine for Operation

Time	Position	Applicant's Actions or Behavior
	BOP	ENSURE controllers on GEN TEMP/LEAK WATER Display in AUTO:
		• PLACE 1-TV-3097, Primary Water TEMP Controller in AUTO.
		• PLACE 1-TV-3118, Hydrogen TEMP Controller in AUTO.
	BOP	ENSURE the Turbine controls ready for Start-up by PERFORMING the following:
		• ENSURE the Load Control Subloop Controller is OFF.
		• ENSURE the Load Target Setpoint Controller SET at 30 MWe.
		• ENSURE Load Rate Setpoint Controller SET at 10 MWe/MIN.
		• ENSURE Turbine in Speed Control by VERIFYING SPEED bar is red.
	BOP	VERIFY the Turbine Trip is RESET and OBSERVE Turbine Trip bar is white.
	BOP	VERIFY the following Turbine parameters:
		• DETERMINE 1-PI-6559, TURB L/O PRESS > 25 psig.
		• DETERMINE 1-PI-6561, EHC FLUID PRESS at least 114 psig.
		• DETERMINE 1-PI-6566, HP EHC FLUID PRESS ~455 psig.
	BOP	If desired, ENSURE Feedwater Bypass Control Valve Controllers in AUTO.
	US	DETERMINE Attachment 1 was COMPLETED & REVIEWED by the Shift Manager per Turnover Sheet prior to exceeding 5% power.
	US	Direct WITHDRAWAL of Control Rods in no more than five (5) step increments to raise power.
	RO	WITHDRAW Control Rods in no more than five (5) step increments while monitoring Reactor power level.
	RO	VERIFY Power Range Channels respond appropriately as power level rises.

Operating Test : NRC	Scenario # 3	Event # 2	Page 11 of 20				
Event Description:	Raise Reactor Power / Prepare Turbine for Operation						

Time	Position	Applicant's Actions or Behavior
	BOP	As Reactor power increases, VERIFY Steam Dump System continues to maintain Main Steam pressure at approximately 1092 psig.
	US	When reactor power is greater than 5%, LOG entry into MODE 1.
	US	PERFORM OPT-102A for MODE 1 Surveillances.
<u>Floor Cue</u>:	**If requested, REPORT OPT-102A, Operations Shiftly Routine Tests was completed last shift.**	
+15 min	RO	Slowly RAISE Reactor power to between 6% and 8%.
When power level is stabilized at 6% to 8%, or at Lead Examiner discretion, PROCEED to Event 3.		

Operating Test :	NRC	Scenario #	3	Event #	3	Page	12 of 20
Event Description:	Pressurizer Pressure Transmitter Failure						

Time	Position	Applicant's Actions or Behavior
Booth Operator: When directed, EXECUTE Event 3. - RX08A, Pressurizer Pressure Channel (PT-455) fails low.		
Indications Available: 5B-3.4 – PRZR 1 OF 4 PRESS LO 5B-4.4 – PRZR 1 OF 4 SI PRESS LO 5C-3.3 – PRZR PRESS LO BACKUP HTRS ON		
+1 min	RO	RESPOND to Annunciator Alarm Procedures.
	RO	RECOGNIZE PRZR pressure rising with PRZR heaters ON.
	US	DIRECT performance of ABN-705, Pressurizer Pressure Malfunction, Section 2.0.
Examiner Note: The next three (3) steps are Initial Operator Actions.		
	RO	VERIFY PORV closed.
	RO	PLACE 1-PK-455A, PRZR Master Pressure Control in MANUAL.
	RO	ADJUST 1-PK-455A for current RCS pressure.
	RO	TRANSFER to an alternate controlling channel, 1/1-PS-455F, PRZR Press Control Channel Select.
	RO	PLACE 1-PK-455A in AUTO.
	RO	VERIFY automatic control restoring Pressurizer pressure to 2235 psig.
	RO	ENSURE a valid channel selected to recorder 1/1-PS-455G, 1-PR-455 PRZR Pressure Select.

Operating Test :	NRC	Scenario #	3	Event #	3	Page	13	of 20
Event Description:	Pressurizer Pressure Transmitter Failure							

Time	Position	Applicant's Actions or Behavior
	US	Within 1 hour, VERIFY PCIP window 2.6, PRZR PRESS SI BLK PERM P-11 in required state for current pressure (DARK).
	US/RO	VERIFY other instruments on common instrument line – NORMAL.
		• DETERMINE LT-459, LT-459F, and PT-455F readings are NORMAL.
+10 min	US	EVALUATE Technical Specifications.
		• LCO 3.3.1.E, Reactor Trip System Instrumentation.
		• CONDITION E - One channel inoperable. • ACTION E.1 - Place channel in trip within 72 hours.
		• LCO 3.3.2.L, ESFAS Instrumentation.
		• CONDITION L - One channel inoperable. • ACTION L.1- Verify interlock in required state for existing condition within 1 hour.
		• LCO 3.3.2.D, ESFAS Instrumentation.
		• CONDITION D - One channel inoperable. • ACTION D.1 - Place channel in trip within 72 hours.

When Technical Specifications are addressed, or at Lead Examiner discretion, PROCEED to Event 4.

Operating Test : NRC Scenario # 3 Event # 4 Page 14 of 20

Event Description: Steam Generator Level Transmitter Failure

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Event 4.** **- RX04D, Steam Generator 1-04 Level Transmitter (LT-554) fails low.**		
Indications Available: **8A-4.6 – SG 4 LVL LO** **8A-4.8 – SG 4 STM & FW FLO MISMATCH (power level dependent)** **8A-4.12 – SG 4 LVL DEV (power level dependent)** **8A-4.14 – SG 4 1 OF 4 LVL LO-LO** **1-LI-554, SG 4 LVL (NR) CHAN I indication failed low**		
+1 min	BOP	RESPOND to Annunciator Alarm Procedures.
	BOP	RECOGNIZE Steam Generator 1-04 Level Transmitter (LT-554) failed low.
	US	DIRECT performance of ABN-710, Steam Generator Level Instrumentation Malfunction, Section 2.0.
	BOP	DETERMINE controlling level channel has failed.
	BOP	Manually CONTROL 1-LK-580, SG 4 BYP CTRL as necessary to maintain Steam Generator 1-04 at programmed level.
	BOP	VERIFY instruments on common instrument line indicate NORMAL.
		• VERIFY Loop 3 Instruments FT-542, LT-549, and FT-543 responding normally per Attachment 1.
ABN-710 Caution: **Turbine Trip AND Feedwater Isolation will occur if 2 or more of the 3 HI-HI level bistables for the SAME Steam Generator are TRIPPED.**		
	BOP	DETERMINE all HI-HI level bistable windows on TSLB-3 for Steam Generator 1-04 are DARK.
	BOP	VERIFY automatic SG level control available:
		• OBSERVE alternate level control channel 1-LI-549A indication NORMAL.
		• DETERMINE automatic level control desired by Unit Supervisor.

Operating Test :	NRC	Scenario #	3	Event #	4	Page 15	of 20
Event Description:	Steam Generator Level Transmitter Failure						

Time	Position	Applicant's Actions or Behavior
	BOP	PLACE 1-LS-549C, Steam Generator 4 Level Channel Select to the LY-549 position.
	BOP	PLACE 1-FK-580, SG 4 BYP CTRL in AUTO and MONITOR operation.
+10 min	US	EVALUATE Technical Specifications.
		• LCO 3.3.1.E, Reactor Trip System Instrumentation.
		• CONDITION E - One channel inoperable (Channel 4 LO-LO). • ACTION E.1 - Place channel in trip within 72 hours.
		• LCO 3.3.2.D, ESFAS Instrumentation.
		• CONDITION D - One channel inoperable (Channel 4 LO-LO). • ACTION D.1 - Place channel in trip within 72 hours.
		• LCO 3.3.2.I, ESFAS Instrumentation.
		• CONDITION I - One channel inoperable (Channel 4 HI-HI). • ACTION I.1 - Place channel in trip within 72 hours.
When Technical Specifications are addressed, or at Lead Examiner discretion, PROCEED to Events 5, 6, and 7.		

Operating Test :	NRC	Scenario #	3	Event #	5, 6, & 7	Page	16	of 20
Event Description:	Reactor Coolant Pump Trip / Manual Reactor Trip Failure / Anticipated Transient Without Trip / Centrifugal Charging Pump Trip							

Time	Position	Applicant's Actions or Behavior
Booth Operator: **When directed, EXECUTE Events 5, 6, and 7.** **- RC07A, Reactor Coolant Pump trip.** **- RP13C, Manual Reactor Trip failure.** **- CONDITIONAL, Bus Breaker CS-1B4-1 Fails to Open.**		
Indications Available: **5B-1.1 – ANY RCP TRIP** **5B-1.2 – 1 of 4 RCP UNDRVOLT** **5A-1.3 – RC LOOP 1 1 OF 3 FLO LO**		
Examiner Note: **The crew may enter ABN-101, Reactor Coolant Pump Trip / Malfunction, if it is not immediately determined that a Reactor Trip is required.**		
+30 sec	RO	RECOGNIZE Reactor Coolant Pump trip and INFORM US Reactor trip is required.
	US	DIRECT a Reactor Trip and performance of EOP-0.0A, Reactor Trip or Safety Injection.
Examiner Note: **The Reactor fails to trip when the breaker supplying a CEDM Motor Generator Set remains closed. Opening then closing these breakers would normally trip the CEDM MG set.**		
CRITICAL TASK STATEMENT		**Manually Trip Reactor upon Failure of Reactor to Trip Prior to Exiting FRS-0.1A.**
CRITICAL TASK	RO	Manually INITIATE a Reactor Trip.
		• PLACE 1/1-RTC, RX TRIP Switch in TRIP.
		• PLACE 1/1-RT, RX TRIP Switch in TRIP.
		• DETERMINE Reactor is NOT tripped.
		• [RNO] OPEN CS-1B3-1, INCOMING BKR 1B3-1 and OBSERVE green TRIP light lit.
		• [RNO] RECLOSE CS-1B3-1, INCOMING BKR 1B3-1 and OBSERVE red CLOSE light lit.
		• [RNO] OPEN CS-1B4-1, INCOMING BKR 1B4-1 and OBSERVE red CLOSE light lit.

Operating Test :	NRC	Scenario #	3	Event #	5, 6, & 7	Page	17	of 20
Event Description:	Reactor Coolant Pump Trip / Manual Reactor Trip Failure / Anticipated Transient Without Trip / Centrifugal Charging Pump Trip							

Time	Position	Applicant's Actions or Behavior
		• [RNO] RECLOSE CS-1B4-1, INCOMING BKR 1B4-1.
	RO	VERIFY Reactor Trip:
		• DETERMINE Reactor Trip Breakers – CLOSED.
		• DETERMINE Neutron flux – NOT DECREASING.
	RO	DETERMINE all Control Rod Position Rod Bottom Lights – OFF.
+2 min	US	TRANSITION to FRS-0.1A, Response To Nuclear Power Generation/ATWT, Step 1.
Examiner Note:		**The following steps are from FRS-0.1A, Response To Nuclear Power Generation/ATWT.**
	RO	VERIFY Reactor Trip:
		• DETERMINE Reactor Trip Breakers – CLOSED.
		• DETERMINE Neutron flux – NOT DECREASING.
		• DETERMINE all Control Rod Position Rod Bottom Lights – OFF.
	RO	• [RNO] INSERT Control Rods ≥ 48 steps/minute.
	BOP	VERIFY Turbine Trip:
		• DETERMINE all HP Turbine Stop Valves – CLOSED.
	BOP	VERIFY Total AFW Flow – GREATER THEN 860 GPM:
		• Manually START both Motor Driven Auxiliary Feedwater Pumps.
	RO	INITIATE Emergency Boration.
CRITICAL TASK STATEMENT		**Initiate Emergency Boration During Anticipated Transient Without Trip Prior to Exiting FRS-0.1A.**

Operating Test :	NRC	Scenario #	3	Event #	5, 6, & 7	Page	18 of 20
Event Description:	Reactor Coolant Pump Trip / Manual Reactor Trip Failure / Anticipated Transient Without Trip / Centrifugal Charging Pump Trip						

Time	Position	Applicant's Actions or Behavior
CRITICAL TASK	RO	INITIATE Emergency Boration of Reactor Coolant System.
Examiner Note: The following steps are from FRS-0.1A, Response To Nuclear Power Generation/ATWT, Attachment 1.F, Initiate Emergency Boration.		
		• [1.F] ENSURE a Centrifugal Charging Pump – RUNNING.
		• [1.F] VERIFY Charging flow – GREATER THAN 30 GPM.
		• [1.F] START both Boric Acid Transfer Pumps.
		• [1.F] PLACE 1/1-APBA1, BA XFER PMP 1 in START.
		• [1.F] PLACE 1/1-APBA1, BA XFER PMP 2 in START.
		• [1.F] PLACE 1/1-8104, EMER BORATE VLV in OPEN.
		• [1.F] VERIFY flow on 1-FI-183A, EMER BORATE FLO.
Booth Operator: Two minutes after being contacted to locally trip the Reactor and once emergency boration is initiated, EXECUTE remote functions RPR112 and RPR 113 to locally trip Reactor.		
	US/RO	CHECK Pressurizer pressure – LESS THAN 2335 psig.
	US/RO	CHECK If The Following Trips Have Occurred:
		• VERIFY Reactor – TRIPPED.
	RO	• [RNO] DISPATCH operator to locally trip Reactor.
		• DETERMINE Turbine – TRIPPED.
	RO/BOP	VERIFY Containment Ventilation Isolation – APPROPRIATE MLB LIGHT INDICATION.
	US/RO	CHECK If Reactor Is Subcritical:
		• DETERMINE Power Range indication – LESS THAN 5%.
		• DETERMINE Intermediate Range Channels – NEGATIVE STARTUP RATE.

Operating Test : NRC Scenario # 3 Event # 5, 6, & 7 Page 19 of 20

Event Description: Reactor Coolant Pump Trip / Manual Reactor Trip Failure / Anticipated Transient Without Trip / Centrifugal Charging Pump Trip

Time	Position	Applicant's Actions or Behavior
		• GO to Step 18.
	US/RO	DETERMINE RCPs Should NOT Be Stopped.
	US/RO	RETURN to Procedure and Step in Effect.
Examiner Note: **The following steps are from EOP-0.0A, Reactor Trip or Safety Injection.**		
	RO	VERIFY Reactor Trip:
		• DETERMINE Reactor Trip Breakers – OPEN.
		• DETERMINE Neutron flux – DECREASING.
	RO	DETERMINE all Control Rod Position Rod Bottom Lights – LIT.
	BOP	VERIFY Turbine Trip:
		• DETERMINE all HP Turbine Stop Valves – CLOSED.
	BOP	VERIFY Power to AC Safeguards Buses:
		• DETERMINE both AC Safeguards Buses – ENERGIZED.
	US/RO	DETERMINE Safety Injection – NOT REQUIRED.
		• [RNO] If SI is NOT required, GO to EOS-0.1A, Reactor Trip Response, Step 1.
Examiner Note: **EOS-0.1A, Reactor Trip Response, steps begin here.**		
	RO	CHECK RCS Temperature:
		• DETERMINE RCS average temperature stable at or trending to 557ºF.

Operating Test :	NRC	Scenario #	3	Event #	5, 6, & 7	Page	20	of 20

Event Description: Reactor Coolant Pump Trip / Manual Reactor Trip Failure / Anticipated Transient Without Trip / Centrifugal Charging Pump Trip

Time	Position	Applicant's Actions or Behavior
	RO/BOP	CHECK FW Status:
		• VERIFY Reactor Trip Breakers – OPEN.
		• CHECK RCS average temperatures < 564ºF.
		• VERIFY Feedwater Isolation – ISOLATION COMPLETE.
	BOP	DETERMINE total AFW flow to SGs – GREATER THAN 460 GPM **<u>or</u>** MAINTAIN any SG narrow range level greater than 43%.
<u>Examiner Note</u>:		**Pressurizer level is low due to greater than 860 GPM of Auxiliary Feedwater flow during FRS-0.1A entry and minimal core decay heat.**
<u>Booth Operator</u>:		**When Pressurizer level is verified greater than 10%, EXECUTE malfunction CV01B, Centrifugal Charging Pump 1-01 trip.**
	RO	DETERMINE Centrifugal Charging Pump 1-01 has tripped.
		• Manually START Centrifugal Charging Pump 1-02.
	RO	CHECK PRZR Level Control:
		• DETERMINE PRZR Level – LESS THAN 17%.
		• [RNO] CLOSE Letdown Orifice Isolation Valves.
		• [RNO] CLOSE Letdown Orifice Isolation Valves.
		• [RNO] CLOSE Excess Letdown Isolation Valves.
		• [RNO] VERIFY Pressurizer Heaters – OFF.
		• VERIFY Charging – IN SERVICE.
When actions to restore Pressurizer level are in progress, TERMINATE the scenario.		