Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-1
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 90% power. LPCI pump 2RHS*P1C is out of service for maintenance.		
Turnover:	Start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0. The procedure is in progress up to step F.3.2. Secure CCS pump 2CCS-P1A per N2-OP-14 section F.4.0. Then, raise power to 95% using Recirculation flow per the provided ReMA.		
Critical Tasks:	CT-1: Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5. RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig. CT-2: Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC. Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap CCS Pumps N2-OP-14
2	N/A	R – ATC, SRO	Raise Reactor Power with Recirculation Flow N2-OP-101D
3	RR46A	C – ATC, SRO TS – SRO	Recirculation Pump High Vibration ARP 602219, N2-SOP-29.1, N2-SOP-29, Technical Specifications
4	Overrides	TS – SRO	RCIC Steam Line Isolation Valve 2ICS*MOV128 Breaker Trips ARP 601305, ARP 601319, Technical Specifications
5	Remote CW27	I – BOP, SRO MC – BOP	2CCS-TIK104 Automatic Setpoint Failure N2-OP-14, N2-SOP-14
6	OV0953 OV0954	I – ATC, SRO MC – ATC	Feedwater Level Control Drifts High in Automatic N2-SOP-06
7	MS04 PC10A	M – All	Steam Leak in Drywell, Pair of Drywell/Wetwell Vacuum Breakers Fail Open N2-SOP-101C, N2-EOP-RPV, N2-EOP-PC
8	RP03	I – ATC, SRO MC – ATC	RPS Fails to Scram the Reactor N2-SOP-101C, N2-EOP-C5
9	FW:BA (B)(C): MT:I	C – All MC – BOP	Condensate Booster Pumps Trip N2-SOP-101C, N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-1**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 8 & 9**	2	
2. Abnormal events (2-4) **Events 3, 5, 6**	3	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV, EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)	0	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 1** Revision: **0.0**

Title: **November 2023 NRC Scenario 1**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/11/23
	N/A / Dan Delaney	7/11/23
	N/A Mike Carno	7/11/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-14, Turbine Building Closed Loop Cooling System
2. N2-OP-101D, Power Changes
3. N2-SOP-29.1, Recirc Pump Failures
4. N2-SOP-29, Sudden Reduction in Core Flow
5. N2-SOP-14, Loss or Degraded CCS System
6. N2-SOP-06, Feedwater Failures
7. N2-SOP-101C, Reactor Scram
8. N2-EOP-RPV (1-3), RPV Control - Flowchart
9. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
10. N2-EOP-6.1, Low RPV Water Level/High Drywell Pressure Actuations
11. N2-EOP-6.22, Containment Sprays
12. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
13. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
14. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
15. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 90% power with LPCI pump 2RHS*P1C out of service.

   b. Event 1 is the normal evolution performed by the BOP operator to swap CCS pumps per N2-OP-14.

   c. Event 2 is a reactivity evolution. The ATC operator will raise Reactor power using Recirculation flow to approximately 95% per the provided ReMA.

   d. Event 3 occurs when Recirculation pump 1A develops a high vibration condition. The crew will respond per ARP 602219, execute N2-SOP-29.1, trip the pump, and execute N2-SOP-29. The SRO will evaluate Technical Specifications.

   e. Event 4 occurs when the breaker for RCIC steam line isolation valve 2ICS*MOV128 trips. The crew will respond per ARPs and the SRO will evaluate Technical Specifications.

   f. Event 5 occurs when the CCS temperature controller, 2CCS-TIK104, automatic setpoint fails high. This results in a temperature rise in all CCS cooled components. The crew will perform the associated ARP actions for the failure. The crew will take manual control of 2CCS-TIK104 and control CCS outlet temperature between 80-95°F (ARP) or 80-85°F (SOP).

   g. Event 6 occurs when a Feedwater level control (FWLC) malfunction causes Reactor water level to drift high. The crew will evaluate Feedwater indications, enter N2-SOP-06, determine that FWLC is not responding correctly, and take manual control of the master controller. The crew will then stabilize and restore Reactor water level to the normal band.

h. Events 7, 8, & 9 begin when a steam leak develops in the Drywell. RPS will fail to scram the Reactor (automatically and manually). The crew will insert control rods by manually initiating RRCS (*critical task*). The crew will respond after the scram by executing N2-SOP-101C and N2-EOP-RPV (1-3). Condensate Booster pumps will trip. This will complicate Reactor water level control. A pair of failed open Drywell/Wetwell vacuum breakers will cause elevated Containment bypass leakage will amplify the Containment pressure response to the steam leak. The Drywell pressure rise will require entry into N2-EOP-PC (1-3). The crew will spray the Suppression Chamber and the Drywell (*critical task).*

2. Termination Criteria

   a. All control rods are inserted.

   b. Reactor level is being controlled in assigned band.

   c. Containment pressure is being controlled per N2-EOP-PC (1-3).

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because with Drywell pressure above 1.68 psig, a Reactor scram is required. Continued power generation further challenges the Primary Containment. With RPS failing to scram the Reactor, RRCS provides alternate means to insert control rods and shutdown the Reactor.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Primary Containment pressure is elevated and RPS has failed to scram the Reactor. N2-EOP-PC (1-3) and N2-EOP-RPV (1-3) provide direction to scram the Reactor.*
Success Path:	*Manually arm and depress RRCS pushbuttons*
Performance Feedback:	*Control rod position and Reactor power indications will provide performance feedback regarding the success of RRCS.*
Measurable Performance Standard:	*Arming and depressing of the RRCS pushbuttons will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

   a. CT-1.0, Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually

initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5 (1-3). RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because initiating Drywell spray reduces Primary Containment pressure. This reduces stresses on the Drywell and Suppression Chamber, assists in avoiding "chugging" that may cause fatigue failure of the LOCA downcomers, and avoids the need for a blowdown. These benefits reduce challenges to the fuel cladding, the RPV, and the Primary Containment.*
Initiating Cue:	*Multiple Primary Containment pressure and temperature indications and annunciators will indicate degrading conditions. N2-EOP-PC (1-3) provides direction to initiate Drywell spray.*
Success Path:	*Drywell spray is initiated using N2-EOP-6.22.*
Performance Feedback:	*Drywell spray flow and lowering Primary Containment pressure and temperature indications will provide performance feedback regarding success of crew actions to initiate Drywell spray.*
Measurable Performance Standard:	*Manipulation of RHR control switches will provide observable actions for the evaluation team.*
Bounding Conditions:	*Based on N2-EOP-PC (1-3) requirements*

b. CT-2.0, Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC (1-3). Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

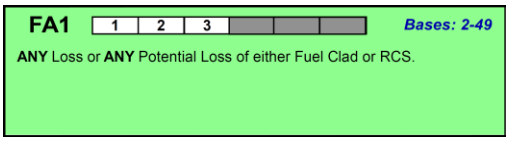
   a. DMS-PC-03, high containment pressure; drywell sprays required; maintain safe region PSP; no blowdown required

6. Technical Specifications (Applicable actions for initial conditions only)

   a. TS 3.5.1 Condition A for LPCI pump 2RHS*P1C (Day 1 of 7)

7. EAL Classification

   a. Alert, EAL FA1:

1) 

B. Initial Conditions

1. IC Number

   a. IC-021 or equivalent (IC-200 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**RH:PC:BK:I**, RHR Pump 1C Breaker Failure, Final = trip	**Inserted**
2)	**RR46A**, RCS PMP1A VIBRATION (MILS), Initial = 3, Final = 25, Ramp = 5 minutes	**TRG1**
3)	**MS04**, Steam Line Rupture Inside Primary Containment, Final = 1.5, Ramp = 15:00	**TRG5**
4)	**PC10A**, Drywell/Wetwell Vacuum Bkr Pair Failed Open, Final = true	**TRG6**
5)	**RP03**, Reactor Protection System Failure To Scram, Final = true	**Inserted**
6)	**FW:BA:MT:I**, Booster Pmp 2A Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
7)	**FW:BB:MT:I**, Booster Pmp 2B Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
8)	**FW:BC:MT:I**, Booster Pmp 2C Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
9)	**RH08**, Group 5 Isolation Failure (RHS*MOV122/115), Final = true	**TRG7**
10)	**OV0953**, FCS-MASTER Setpoint Increment Pushbutton, Final = pressed	**TRG4**
11)	**OV0954**, FCS-MASTER Setpoint Decrement Pushbutton, Final = disabled	**Inserted**

b. Remotes

1) **CW27**, 2CCS-TIK104 Auto Setpoint, Final = 110, Ramp = 2 minutes — **TRG3**

c. Overrides

1) **DO-806**, OFF RHR PUMP P1C GREEN, Final = off — **Inserted**

2) **DO-1023**, OFF MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) GREEN, Final = off — **TRG2**

3) **DO-1024**, ON MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) RED, Final = off — **TRG2**

4) **DI-901**, CLOSE MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD), Final = off — **TRG2**

5) **DO-532**, OFF STEAM SUPPLY INBD ISOL MOV 128 INOP AMBER, Final = on — **TRG2**

d. Annunciators

1) **an601305**, rcic system inoperable, Final = crywolf — **TRG2**

2) **an601319**, rcic valves mtr overload, Final = crywolf — **TRG2**

e. Event Triggers

Event #	Event Action	Command
6	rd_fi==1 (All rods full in)	None
30	Iscla106>187 & ov_slim0spincpb==2 (Reactor water level >187" with event 6 active)	dmf ov0953

f. Equipment Out of Service

1) LPCI pump 2RHS*P1C

g. Support Documentation

1) Provide N2-OP-14 sections F.1.0, F.3.0, and F.4.0. Have section F.3.0 completed up to step F.3.2.

2) Provide ReMA for raising Reactor power from 90% to 95% with Recirc flow.

3) Provide N2-OP-101D completed up to step F.3.7.d.

4) Provide N2-OP-29 section F.4.0

h. Miscellaneous

1) Place clearance reference tag on the LPCI pump 2RHS*P1C control switch in P-T-L.

2) Place Protected pathway signs on 2RHS*P1A and 2RHS*P1B control switches.

C. Shift Turnover Information

1. Reactor Power: 90%
2. Rodline: Above 100%
3. Technical Specification LCOs in effect:
   a. TS 3.5.1 Condition A for LPCI pump 2RHS*P1C (Day 1 of 7)
4. Significant Problems / Abnormalities / Equipment Out of Service:
   a. LPCI pump 2RHS*P1C out of service for maintenance
5. Evolutions / Maintenance Scheduled for this Shift:
   a. Swap CCS pumps per N2-OP-14 sections F.3.0 and F.4.0.
   b. Then, raise Reactor power to 95% using Recirculation flow per the provided ReMA.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 90%.
- LPCI pump 2RHS*P1C is out of service for maintenance. TS 3.5.1 Condition A applies (Day 1 of 7).

PART III: Remarks/Planned Evolutions:

- Start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0. The procedure is in progress up to step F.3.2. Secure CCS pump 2CCS-P1A and place it in standby per N2-OP-14 section F.4.0.
- Then, raise Reactor power to 95% using Recirculation flow per the provided ReMA.

Event #1 – Swap CCS Pumps

Event Information	• Presumed or required initial plant operating conditions - Reactor at power - 2CCS-P1A and 2CCS-P1B running, 2CCS-P1C shutdown • Verifiable Actions - Start 2CCS-P1C - Stop 2CCS-P1A • Final (expected) operating result - 2CCS-P1B and 2CCS-P1C running, 2CCS-P1A in standby

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0 and secure CCS pump 2CCS-P1A per N2-OP-14 section F.4.0.
Role Play: If asked about operating indications after 2CCS-P1C start, report that all indications are SAT. **Role Play:** When directed to slowly open 2CCS-V303C, wait 1 minute, then report 2CCS-V303C is open. **Role Play:** When directed to close 2CCS-V303A, wait 1 minute, then report 2CCS-V303A is closed. **Role Play:** If asked about operating indications after 2CCS-P1A stop, report that all indications are SAT. **Role Play:** When 2CCS-P1A is secured and asked to open 2CCS-V303A, wait 1 minute, then report 2CCS-V303A is open.	**BOP** • Acknowledges direction to start 2CCS-P1C and shutdown 2CCS-P1A • Starts 2CCS-P1C by placing control switch to Normal-After-START • Directs EO to slowly open 2CCS-V303C • Confirms normal operating indications per Subsection F.1.0: ◦ Pump discharge pressure 120-135 psig ◦ Pump current <58.5 amps ◦ 2CCS-TIK104 in Auto controlling ~85°F • Confirms 2CCS-P1A is NOT required for CCS System flow • Directs EO to close 2CCS-V303A • Secures 2CCS-P1A by placing control switch in Normal-After-STOP • Confirms normal operating indications per Subsection F.1.0: ◦ Pump discharge pressure 100-125 psig ◦ Pump current <58.5 amps

Instructor Actions / Plant Response	Operator Actions
	◦ 2CCS-TIK104 in Auto controlling ~85°F

Event Termination Criteria	• 2CCS-P1C is running and 2CCS-P1A is in standby

Event #2 – Raise Reactor Power with Recirculation Flow

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~90% power • Verifiable Actions - Raise Recirculation flow • Final (expected) operating result - Reactor power at ~95%

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to raise Reactor power to 95% using Recirc flow per ReMA and N2-OP-101D
Note: T.S. 3.4.1 requires that loop flow mismatch be maintained ≤ 5% of rated core flow (108.5 mlbm/hr) when operating at an effective core flow ≥ 70% of rated core flow. In this case the loop flow mismatch allowable would be 5.425 mlbm/hr.	**ATC** • Acknowledges direction to raise Reactor power to 95% using Recirc flow • Raises Reactor power to 95% by raising core flow as follows: ◦ Moves 2RCS*HYV17A&B individually in the open direction, maintaining loop flow differential at a minimal value by alternating between the two valves • Monitors APRMs and rate of power change
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• Reactor power has been raised sufficiently as determined by the Lead Evaluator

Event #3 – Recirculation Pump High Vibration

Event Information	• Presumed or required initial plant operating conditions - Plant on-line near rated power with Recirc pump 1A running • Verifiable Actions - Trip Recirc pump 1A • Final (expected) operating result - Operating at reduced power in single loop operation on Recirc pump 1B

Instructor Actions / Plant Response	Operator Actions
When directed by the lead evaluator, **insert** the following **malfunction**: **TRG1** **RCS PMP1A VIBRATION (MILS)**, Initial = 3, Final = 25, Ramp = 5 minutes	
The following annunciator alarms: • *602219, RECIRC 1A/1B VIBRATION HIGH* *The following computer point is generated:* • *RCSNC03, RCRC PMP1A VIBRATION HIGH*	**ATC** • Silences, acknowledges and reports annunciator 602219
Note: N2-RESP-07 may be referenced for Tech Specs. **Role Play:** As RE contacted, acknowledge direction and inform the control room you will start working on it. **Spec** / **Condition** / **Applicable Actions**: 3.4.1 / C / C.1 **Action** / **Description**: C.1 / Satisfy the requirements of the LCO (4 hours)	**SRO** • Acknowledges high vibration on Recirc pump 1A • Directs ATC to enter and execute N2-SOP-29.1 • Directs trip of Recirc pump 1A • Directs ATC to enter and execute N2-SOP-29 • Refers to TS 3.4.1 and determines the LCO is not met because RPS flow instrumentation is not reset for single loop operation. Enters a four hour LCO, Condition C. • Notifies RE to check thermal limits and to make adjustments to the rod pattern based on operation in the Heightened Awareness Region

Instructor Actions / Plant Response	Operator Actions
Role Play: As EO dispatched to monitor vibrations, wait 2 minutes, then report that Recirc pump 1A motor shaft vibrations are 21 mils and rising slowly, motor frame vibrations are 6 mils and stable, Recirc pump 1B vibrations are all normal. **Note:** The first 8 CRAM rods are: • 30-39 • 30-23 • 22-31 • 38-31 • 14-39 • 46-23 • 14-23 • 46-39	**ATC** • Executes ARP 602219 • Dispatches EO to monitor vibrations • Coordinates with BOP to monitor motor temps • Determines / reports N2-SOP-29.1 entry is required • Acknowledges direction to enter N2-SOP-29.1 • Determines / reports that Recirc pump 1A must be tripped • Trips Recirc pump 1A • Acknowledges direction to enter N2-SOP-29 • Determines a Recirc pump is in service • Determines core flow and power are NOT within the Scram Region • Determines Core flow AND power are NOT left of Line K • Inserts the first four cram rods • Performs N2-SOP-29, Attachment 1: • If plant is operating in the EXIT region: • Shifts APRM recorders to fast speed and monitors for core oscillations • Inserts more CRAM rods and/or lowers core flow • Determines cause of reduced core flow is NOT FCV runback • Verifies closed flow control valve for RCS loop A • Verifies RCS loop B is operating <41,800 gpm • Contacts I&C for adjustment to APRMs, Rod Blocks, and Rod Block Monitor • Refers to N2-OP-29, Section H.6.0 for single loop operations. • Posts operating in “Heightened Awareness Region”
	BOP • Coordinates with ATC to monitor motor winding and bearing temperatures

Event Termination Criteria	• Recirc pump 1A tripped • 4 CRAM rods inserted • Technical Specifications have been addressed

Event #4 – RCIC Steam Line Isolation Valve 2ICS*MOV128 Breaker Trips

Event Information	• Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Technical Specifications addressed • Final (expected) operating result - Plant on-line

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **overrides**: **TRG2** **an601305**, rcic system inoperable, Final = crywolf **an601319**, rcic valves mtr overload, Final = crywolf **DO-1023**, OFF MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) GREEN, Final = off **DO-1024**, ON MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) RED, Final = off **DI-901**, CLOSE MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD), Final = off **DO-532**, OFF STEAM SUPPLY INBD ISOL MOV 128 INOP AMBER, Final = on *The plant responds as follows:* • *2ICS*MOV128 indicating lights extinguish*	
The following annunciators alarm: • *601305, RCIC SYSTEM INOPERABLE* • *601319, RCIC VALVES MOTOR OVERLOAD*	**BOP** • Silences, acknowledges and reports annunciators • Reports 2ICS*MOV128 breaker is tripped

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.6.1.3</td><td>A</td><td>A.1 & A.2</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured (4 hours)</td></tr>
<tr><td>A.2</td><td colspan="2">Verify the affected penetration flow path is isolated (31 days)</td></tr>
</table>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.5.3</td><td>A</td><td>A.1 & A.2</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Verify by administrative means HPCS is operable (Immediately)</td></tr>
<tr><td>A.2</td><td colspan="2">Restore RCIC to operable status (14 days)</td></tr>
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SRO

- Acknowledges reports
- Reviews Tech. Spec. 3.6.1.3 Condition A, required actions A.1 & A.2
- May direct closing 2ICS*MOV121
- May review Tech Spec. 3.5.3, Condition A, required actions A.1 & A.2

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Role Play:

As EO dispatched to investigate 2ICS*MOV128 breaker (2EHS*MCC302 BKR 14A), wait 2 minutes, then report that the breaker is in the tripped condition and there is a slight acrid odor.

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BOP

- Executes ARP 601305:
  - Determines INOP Status Light is energized for 2ICS*MOV128
- Executes ARP 601319:
  - Determine 2ICS*MOV128 is the affected MOV
- Executes INOP STATUS LIGHT ARP (TURB STM SPLY INBD ISOL VLV ICS*MOV128):
  - Observes 2ICS*MOV128 red and green indicating lights extinguished
  - Dispatches EO to 2EHS*MCC302

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</table>

Event Termination Criteria	• Technical Specifications addressed

Event #5 – 2CCS-TIK104 Auto Setpoint Failure

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Taking manual control of 2CCS-TIK104 and lower temperature setting • Final (expected) operating result - Operating

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **remote**: **TRG3** **CW27**, 2CCS-TIK104 Auto Setpoint, Final = 110, Ramp = 2 minutes	
The following plant response occurs after event initiation: • *2CCS-TIK104 setpoint slowly drifts toward minimum cooling* *After approximately 30 seconds the following computer point is generated:* • *GMLTA01, GN LEADS CLR RTN AIR TEMP (HI)* *The following annunciator alarms ~1:30 after event initiation:* • *601244, TURBINE BLDG CLOSED LOOP COOLING SYS TROUBLE* *The following computer points are generated ~1:30 after event initiation:* • *CCSTC04, TBCLCW HX DISCH TEMP (HIGH)* • *CCSTA01, TBCLCW SYS HX DISCH TEMP (HIGH)*	**RO** • Silences, acknowledges and reports annunciator 601244 • Recognizes / reports 2CCS-TIK104 setpoint rising

Instructor Actions / Plant Response	Operator Actions
Note: Crew may opt to use ARP 601244 actions vs. using N2-SOP-14 guidance. The actions are the same with the exception of the allowed temperature band. In N2-SOP-14 the allowable temperature band is 80-85°F whereas the ARP guidance for computer point CCSTC04 allows 80-95°F.	**SRO** • Acknowledges reports • Directs BOP to execute ARP 601244 and/or N2-SOP-14 • May direct BOP to execute N2-SOP-68 to monitor in the event a generator runback occurs
Role Play: If directed as field operator to investigate stator water coolant flow and temperature at the stator water cooling/hydrogen monitor panel, wait 5 minutes and report the following: • If the CCS-TIK104 setpoint has been dialed down in manual and annunciator 851156 has cleared, then report that stator water inlet temperature is ~38°C, stator water outlet temperature is ~56°C and stator water flow is ~760 gpm • If the CCS-TIK104 setpoint has not yet been dialed down in manual and annunciator 851156 is still in alarm, then report that stator water inlet temperature is ~40°C, stator water outlet temperature is ~59°C and stator water flow is ~760 gpm	**BOP** • Performs N2-SOP-14 actions: ◦ Determines that all CCS Pumps have not tripped ◦ Determines that SWP supply to the turbine building is not isolated ◦ Determines CCS system conditions have not degraded to the point that system loads will NOT have adequate cooling ◦ Determines that main turbine high vibration alarm is not in ◦ Determines that a failure of the standby pump to auto start has not occurred ◦ Determines that the CCS expansion tank is not the problem ◦ Determines at least one CCS pump is operating ◦ Determines that cooling capacity is degraded: - Monitors system loads (Detail A, Major System Loads) - Secures loads as necessary ◦ Performs the following to stabilize the CCS system: - Places CCS-TIK104 in manual - Manually controls temperature using CCS-TIK104 between 80-85°F • Provides update to the control room that manual control has been established and that CCS temperature is lowering • Performs ARP 851112 actions, as required:

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched to CCS temperature control valve, wait 2 minutes and then report that there is nothing abnormal at the valve.	◦ Determines the cause of the event to be rising CCS inlet temperature due to the failed CCS temperature controller ◦ Determines that stator water cooling is lowering • Provides crew update for the requirement of ARP 851112 to enter N2-SOP-68 and the N2-SOP-68 entry, as required • Performs N2-SOP-68 actions, as required: ◦ Reviews N2-SOP-68 and determines that no action is required unless a generator runback has occurred ◦ Notifies the CRS of N2-SOP-68 requirements

Event Termination Criteria	• 2CCS-TIK104 in manual with temperature setting in required band

Event #6: Feedwater Master Controller Failure - High

Event Information	• Presumed or required initial plant operating conditions - Plant on-line with FWLC in master auto • Verifiable Actions - Take manual control of feedwater master controller and stabilize/restore feedwater level back in normal band • Final (expected) operating result - Operating with Feedwater Master controller in manual

Instructor Actions / Plant Response	Operator Actions
When directed by lead evaluator **insert** the following **malfunction**: **TRG4** **OV0953**, FCS-MASTER Setpoint Increment Pushbutton, Final = pressed	
The following annunciator alarms ~30 seconds after event initiation: • *603139, REACTOR WATER LEVEL HIGH/LOW* *The following computer points are generated after event initiation:* • *FWSLC01, REACTOR WTR LEVEL HI/LO*	**Crew** • Recognizes / reports the following: ◦ Annunciator 603139 in alarm ◦ RPV water level slowly rises ◦ Steam flow / feed flow mismatch
Note: The Feedwater master auto setpoint decrement pushbutton is failed off. When Reactor water level rises above ~187", **TRG30** will automatically **delete malfunction OV0953**. Reactor water level will stabilize ~191" and the Feedwater master auto setpoint will NOT be able to be lowered. **Note:** Crew may enter N2-SOP-08 and N2-SOP-101D for unplanned power changes depending on the timing of initial event diagnosis.	**SRO** • Provides crew update for entry into N2-SOP-06 • Directs ATC to enter and execute N2-SOP-06 • Directs RPV level band

Instructor Actions / Plant Response	Operator Actions
	ATC • Monitors the following: ◦ Reactor power ◦ Reactor water level ◦ Reactor pressure • Performs N2-SOP-06 actions and determines the following: ◦ FWLC is NOT responding correctly • Takes manual control of FWLC master controller • Adjusts feedwater master controller in manual to restore feedwater level to directed level band

Event Termination Criteria	• FWLC in master manual • Reactor water level in normal band

Constellation

Event #7, 8, and 9 – Steam Leak in Drywell, Pair of Drywell/Wetwell Vacuum Breakers Fail Open; RPS Fails to Scram the Reactor; Condensate Booster Pumps Trip

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Control rods inserted using RRCS - Reactor water level controlled - Suppression Pool and Drywell sprayed • Final (expected) operating result - All control rods inserted - Reactor level is being controlled in assigned band - Containment pressure is being controlled per N2-EOP-PC (1-3) • Mitigation Strategy code (if applicable) - DMS-PC-03, high containment pressure; drywell sprays required; maintain safe region PSP; no blowdown required
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5 (1-3). RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig. 2. Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC (1-3). Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunction**: **TRG5** **MS04**, Steam Line Rupture Inside Primary Containment, Final = 1.5, Ramp = 15:00 *Drywell pressure rises* *Drywell temperature rises*	

Instructor Actions / Plant Response	Operator Actions
The following malfunction is preset to prevent RPS from inserting control rods: ***RP03**, Reactor Protection System Failure To Scram, Final = true* *The following malfunctions will automatically insert when all control rods are full in:* **TRG6** **PC10A**, Drywell/Wetwell Vacuum Bkr Pair Failed Open, Final = true **FW:BA:MT:I**, Booster Pmp 2A Motor Failure, Final = short, Delay = 2 minutes **FW:BB:MT:I**, Booster Pmp 2B Motor Failure, Final = short, Delay = 2 minutes **FW:BC:MT:I**, Booster Pmp 2C Motor Failure, Final = short, Delay = 2 minutes	**Crew** • Recognizes / reports rising Drywell pressure and temperature
Note: N2-SOP-101C has an immediate action for the ATC to manually initiate ARI once RPS fails to depressurize the scram air header and insert rods. Alternately, the SRO may direct this based on guidance in N2-EOP-C5 (1-3). N2-EOP-C5 (1-3) entry is NOT expected, since the guidance in N2-SOP-101C will likely be carried out prior to EOP entry.	**SRO** • Acknowledges reports • May direct ATC to lower power using Recirc flow IAW with N2-SOP-101D • Directs manual Reactor scram • Directs entry into N2-SOP-101C • Acknowledges failure of RPS to insert control rods • **If required, directs manual initiation of RRCS (CT-1.0)** • Enters N2-EOP-RPV (1-3) on low Reactor water level and high Drywell pressure • Enters N2-EOP-PC (1-3) on high Drywell pressure and temperature • Performs N2-EOP-RPV (1-3) actions: ◦ Directs scram procedure entered (N2-SOP-101C) ◦ Performs level control actions: - Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1

Instructor Actions / Plant Response	Operator Actions
	- Determines that RPV water level is known - Determines that RPV water level can be restored and maintained between 159.3 and 202.3 inches using Feedwater, HPCS, RCIC, CRD, and/or SLC - Directs an RPV level band of 160 – 200 inches - May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: - Determines that no SRVs are cycling - Determines that RPV pressure can be stabilized below 1052 psig using the main turbine bypass valves ◦ Directs a pressure band of 800-1000 psig • Performs N2-EOP-PC (1-3) primary containment pressure leg actions: • Determines cannot maintain containment pressure below 1.68 psig • Determines that suppression chamber spray is required before suppression chamber pressure reaches 10 psig • Determines that suppression pool water level is below EL 217 ft. • Determines that suppression chamber sprays are warranted • Directs RO to place RHR in suppression chamber sprays • Directs pneumatics restored to the Drywell • Establishes a benchmark for closing the outboard MSIVs to control cooldown rate • May direct low pressure ECCS pumps terminated and prevented with a high drywell pressure and reactor pressure lowering (N2-EOP-RPV (1-3), step P-1) • Directs Suppression Chamber Spray secured prior to Suppression Chamber pressure lowering to 0 psig • Waits until suppression chamber pressure is above 10 psig

Instructor Actions / Plant Response	Operator Actions
	• Determines that suppression pool water level is below EL 217 ft. • Directs RO to verify the following: - Both recirculation pumps tripped - All drywell unit coolers tripped • **Directs RO to place RHR in drywell sprays in accordance with N2-EOP-6.22 (CT-2.0)**
Booth Operator: If directed to defeat the HPCS Level 8 interlocks by placing HPCS INJ VLV LEVEL 8 BYPASS TEST SWITCH in the TEST position, **manually insert** the following **remote function**: • **CS14**, OPS-CSH01 PNL625 TST.SW CSH*MOV107, Final = Test Wait 1 minute, then report that the HPCS Level 8 interlocks have been defeated. **Booth Operator:** If directed to defeat the RCIC Level 8 interlocks, **manually insert** the following **remote function**: • **RC02A,** RCIC Level 8 Trip Defeat: Withdraw Trip Units N693A, E, Final = defeated Wait 1 minute, then report that the RCIC Level 8 interlocks have been defeated.	**ATC** • Acknowledges SRO direction to scram the reactor • Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603. • Places the Reactor Mode Switch in the SHUTDOWN position • Recognizes / reports failure of RPS to insert rods • **Manually initiates RRCS (CT-1.0)**: ◦ Arms and depresses pushbuttons: ▪ DIVISION I CHANNEL A MANUAL INITIATION ▪ DIVISION I CHANNEL B MANUAL INITIATION ▪ DIVISION II CHANNEL A MANUAL INITIATION ▪ DIVISION II CHANNEL B MANUAL INITIATION • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs open ◦ Feedwater pumps are running (initially) ◦ All control rods fully inserted • Reports EOP entry conditions on low RPV water level, high Drywell pressure, high Drywell temperature • Performs N2-SOP-101C actions: ◦ Verifies: - All rods full in - Reactor power lowering - Turbine tripped/TSVs & TCVs shut

Instructor Actions / Plant Response	Operator Actions
	- Generator tripped and house loads transferred - SDV V & D Valves closed - RCS pumps downshift - RPV pressure on TBVs or SRVs FWLC controlling level > 159.3 ◦ Fully inserts IRMs and SRMs ◦ Energizes 2WCS-MOV107 ◦ If required, secures makeup to the cooling tower ◦ Shuts down HWC ◦ Performs Level control leg: - Recognizes that level will be stabilized using HPCS, RCIC, CRD, and/or SLC - If directed defeats HPCS Level 8 trips by directing a field operator to perform the actions of N2-EOP-6.20
Note: The candidate may also start HPCS by starting the pump and opening 2CSH*MOV107, rather than using the manual initiation pushbutton. **Booth Operator:** If directed to defeat group 5 isolations for RHR A and B, wait 2 minutes, then **insert malfunction**: **TRG7** **RH08,** Group 5 Isolation Failure – (RHS*MOV122/113), Final = true Then report task completion.	**ATC / BOP** • May manually start RCIC: ◦ Rotate RCIC MANUAL INITIATION pushbutton collar to ARMED position ◦ Depress RCIC MANUAL INITIATION pushbutton ◦ Verify GLAND SEAL SYSTEM AIR COMPRESSOR starts ◦ Verify 2ICS*MOV116, LUBE OIL COOLING WTR SUPPLY, opens ◦ Verify 2ICS*MOV120, TURB STM SUPPLY VLV, opens to start RCIC Turbine ◦ Verify 2ICS*MOV126, PMP 1 DISCH TO REACTOR, opens ◦ WHEN RCIC system total flow exceeds 220 GPM, THEN verify 2ICS*MOV143, PMP MINIMUM FLOW TO SUPPRESSION POOL, closes ◦ WHEN RCIC discharge pressure is greater than reactor pressure, THEN verify 2ICS*V156, REACTOR INJECTION OUTBD CHECK VLV, open ◦ IF at any time, automatic flow control of RCIC is NOT desired, is functioning improperly, OR flow is

Instructor Actions / Plant Response	Operator Actions
	less than 400 GPM, THEN return flow control to manual by placing 2ICS*FC101 to M ◦ Verify RCIC FLOW CONTROLLER establishes AND maintains a RCIC total flow of 600 GPM injecting to Reactor pressure vessel • May manually start HPCS: ◦ At 2CEC*PNL601, ARM AND DEPRESS MANUAL INITIATION ◦ At 2CEC*PNL601, VERIFY the following: ◦ 2CSH*P1, HPCS PUMP 1, started. ◦ 2CSH*MOV107, PMP 1 INJECTION VLV, open. ◦ WHEN system flow is greater than 634 gpm, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV closed. ◦ Flow to RPV as indicated by 2CSH*FI104 flow rising AND indication the following valves are closed: - 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL - 2CSH*MOV110, TEST RETURN TO CONDENSATE TK - 2CSH*MOV112, TEST RETURN TO CONDENSATE TK ◦ At 2CEC*PNL852, VERIFY DIVISION 3 2EGS*EG2 started

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If requested to place SWP radiation monitor RE23A in service, **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = Online **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Power, Final = On If requested to place SWP radiation monitor RE23B in service, **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, Final = Online **RM03-041**, SWP23B Current Radiation Level Power, Final = On Then report 2SWP*RE23A(B) is in service	**ATC / BOP** • Initiates Suppression Chamber spray per N2-EOP-6.22: ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies 2RHS*P1A(B) running. ◦ Opens 2RHS*MOV33A(B) to establish suppression chamber spray flow. ◦ Opens 2RHS*FV38A(B) and establish approximately 7450 gpm – If suppression pool cooling is directed. ◦ Throttles open 2SWP*MOV33A(B) to establish flow not to exceed 7400 gpm - Starts 5th SWP pump (post LOCA). ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay. ◦ Directs RP to place 2SWP*RE23A(B) in service • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE ◦ OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD ◦ OPENS 2IAS*SOV164 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD ◦ OPENS 2IAS*SOV165 • If directed, terminates and prevents Low pressure ECCS injection using N2-EOP-HC attachment 4:

Instructor Actions / Plant Response	Operator Actions
	◦ PLACES 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ IF a LPCI A/LPCS initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI A/LPCS MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24A, LPCI A INJECTION VLV ◦ 2CSL*MOV104, PMP 1 INJECTION VLV ◦ PLACES 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK. ◦ IF a LPCI B & C initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI B & C MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24B, LPCI B INJECTION VLV ◦ 2RHS*MOV24C, LPCI C INJECTION VLV • Closes the outboard MSIVs at pre-established pressure value and shifts pressure control to the SRVs • Recognizes/reports to the SRO that suppression chamber pressure has reached 10 psig • Verifies and reports that both recirculation pumps are tripped • Verifies and reports that all drywell unit coolers are tripped • **Initiates Drywell spray per N2-EOP-6.22 (CT-2.0):** ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies closed 2RHS*FV38A(B) ◦ Verifies RHR A(B) pump running. ◦ Opens 2RHS*MOV33A(B) and verifies flow (If suppression chamber spray is required concurrently with drywell sprays). ◦ Opens 2RHS*MOV15A(B) ◦ Opens 2RHS*MOV25A(B)

Instructor Actions / Plant Response	Operator Actions
	◦ Verifies proper operation of 2RHS*MOV4A(B) ◦ Verifies ~7450 gpm on Drywell Spray Header flow ◦ Verifies open 2SWP*MOV90A(B) ◦ Throttle open 2SWP*MOV33A(B) to establish ~ 7400 gpm ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay.

Event Termination Criteria	• All control rods are inserted. • Reactor level is being controlled in assigned band. • Containment pressure is being controlled per N2-EOP-PC (1-3).

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-2
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 65% power with a plant shutdown in progress. APRM 1 is bypassed due to erratic indication. 2MSS*PSV129 is inoperable due to a failed 'C' solenoid.		
Turnover:	Place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from 48 to 00.		
Critical Tasks:	CT-1: Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5: • Rapidly lowering RPV water level • Tripping RCS pumps • Injecting boron The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram. CT-2: Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5. Control rod insertion must be completed within one hour of the start of the failure to scram.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Place Heater Drain Pump in Recirculation Mode N2-OP-8
2	N/A	R – ATC, SRO	Insert Control Rods N2-OP-96
3	RD11	I – ATC, SRO TS – SRO	Loss of Control Rod Position Indication N2-OP-96, Technical Specifications
4	CW:RA: MT:I CW16	C – BOP, SRO MC – BOP	CCP Pump Trips, Standby Pump Fail to Automatically Start N2-SOP-13
5	NM19	I – ATC, SRO TS – SRO	RBM Fails Inoperable N2-OP-92, Technical Specifications
6	MC01	C – All	Loss of Main Condenser Vacuum N2-SOP-9, N2-SOP-101C, N2-EOP-RPV
7	RP03 RP12	M – All MC – All	Electrical ATWS N2-EOP-C5
8	RP12 SL01	C – BOP, SRO	SLC Fails to Start on RRCS, Delayed SLC Pump Trips N2-EOP-C5

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-2** Op-Test No.: **2023-2**

1. Malfunctions after EOP entry (1-2) **Event 8**	1	
2. Abnormal events (2-4) **Events 4 & 6**	2	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV**	1	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-C5**	1	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 2** Revision: **0.0**

Title: **November 2023 NRC Scenario 2**

	Signature / Printed Name		Date
Developed By	Signature on File	/ Mike Alexander	10/27/23
Validated By	N/A	Mecki Ghabarou	7/11/23
	N/A	/ Dan Delaney	7/11/23
	N/A	/ Mike Carno	7/11/23
Facility Reviewer	Signature on File	/ John Toothaker	10/27/23

References

1. N2-OP-8, Feedwater Heaters and Extraction Steam System
2. N2-OP-101C, Plant Shutdown
3. N2-SOP-13, Loss or Degraded CCP System
4. N2-OP-92, Neutron Monitoring
5. N2-SOP-9, Loss of Condenser Vacuum
6. N2-SOP-101C, Reactor Scram
7. N2-EOP-RPV (1-3), RPV Control - Flowchart
8. N2-EOP-C5 (1-3), RPV Control – ATWS - Flowchart
9. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
10. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
11. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
12. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 65% power during a shutdown with APRM 1 bypassed and 2MSS*PSV129 is inoperable.

   b. Event 1 is the normal evolution performed by the BOP operator to place the “C” heater drain pump in recirculation mode per N2-OP-8.

   c. Event 2 is a reactivity evolution. The ATC operator will continue the shutdown by inserting control rods per the shutdown sequence.

   d. Event 3 occurs when control rod position indication is lost to a control rod. The ATC operator will insert the control rod one notch and control rod indication will be restored. The SRO will evaluate Technical Specifications for the time the control rod position indication was lost.

   e. Event 4 occurs when a running CCP pump trips and the standby pump fails to automatically start. The crew will execute N2-SOP-13 and start the standby pump.

   f. Event 5 occurs when RBM A fails inop. The crew will bypass RBM A. The SRO will evaluate Technical Specifications.

   g. Event 6 occurs when Main Condenser vacuum begins to degrade. The crew will execute N2-SOP-09 and lower Reactor power. Vacuum will continue to degrade. The crew will attempt to scram the Reactor.

   h. Events 7-8 begin immediately following the scram attempt when control rods fail to insert. The crew will enter N2-EOP-RPV (1-3) and transition to N2-EOP-C5 (1-3). The crew will take actions to lower Reactor power *(critical task)* and insert control rods *(critical task)*. SLC pumps fail to automatically start on RRCS initiation. This will require

the crew to take manual actions to start SLC. SLC pumps will trip after 2 minutes of operation, limiting the boron injection and resulting power suppression.

2. Termination Criteria

   a. Reactor power downscale on APRMs.

   b. Control rod insertion completed.

   c. Reactor level being controlled in assigned band.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because, without operator action to lower Reactor power, the risk of damage to fission product barriers is significantly elevated.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that the Reactor has failed to scram and that Reactor power remains above 4%. N2-EOP-RPV (1-3) and N2-EOP-C5 (1-3) provide direction to lower Reactor power.*
Success Path:	*The crew will manipulate injection sources to lower Reactor water level, RCS pump control switches, and/or SLS controls.*
Performance Feedback:	*Reactor water level and injection source flow rates will provide performance feedback regarding the success of rapidly lowering RPV water level. RCS pump and SLS pump indications and flow rates will provide performance feedback regarding the success of operating those systems. Overall, multiple indications will provide performance feedback on the effect of these actions on Reactor power.*
Measurable Performance Standard:	*Manipulating injection sources, RCS pump control switches, and/or SLS controls will provide the evaluation team with observable actions.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

   a. CT-1.0, Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5 (1-3):

      - Rapidly lowering RPV water level
      - Tripping RCS pumps
      - Injecting boron

      The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because, without operator action to insert control rods, the Reactor will not achieve a stable, long-term shutdown condition.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence of the failure of control rods to insert on the scram.*
Success Path:	*The crew will utilize one or more methods in N2-EOP-6.14 to insert control rods. For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.*
Performance Feedback:	*Control rod position indication and Reactor power indication will provide performance feedback regarding the success of control rod insertion.*
Measurable Performance Standard:	*Manipulation of CRD/RMCS components and/or direction to field operators will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

b. CT-2.0, Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5 (1-3). Control rod insertion must be completed within one hour of the start of the failure to scram.

Note: For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-AT-01, high power ATWS (greater than APRM downscale); heat addition to suppression pool or torus requires entry into level power control; RPV level controlled below feedwater spargers; RPV blowdown not required

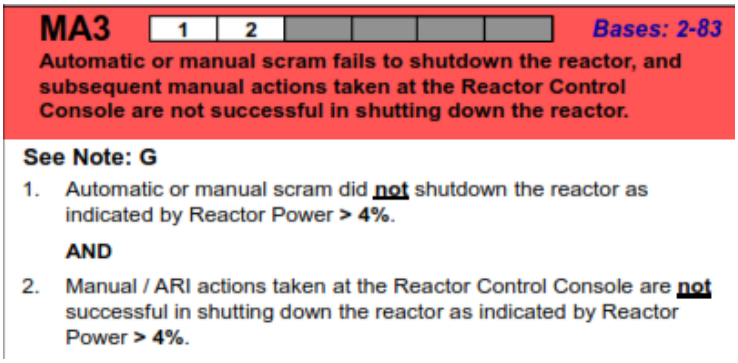
6. Technical Specifications (Applicable actions for initial conditions only)

   a. None

7. EAL Classification

   a. Alert, EAL MA3:

1)



B. Initial Conditions

1. IC Number

   a. IC-023 or equivalent (IC-201 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**AD02U**, Relief Valve Failure PSV129, Final = true	**Inserted**
2)	**RD11-38-39**, Control Rod Failure – RPIS (Any), Final = true	**TRG1**
3)	**CW16C**, CCP-P1C Fails to Auto Start, Final = true	**Inserted**
4)	**CW:RA:MT:I**, RBCLC Pump 1A Motor Failure, Final = short	**TRG2**
5)	**NM19A**, Rod Block Monitor Failure - Inop (A), Final = true	**TRG3**
6)	**MC01**, MAIN CONDENSER AIR INLEAKGE, Final = 50, Ramp = 10 minutes	**TRG4**
7)	**RP03**, Reactor Protection System Failure to Scram, Final = true	**Inserted**
8)	**SL01A**, Standby Liquid Control Pump Trip (P1A), Final = true, Delay = 5:30 minutes	**TRG10**
9)	**SL01B**, Standby Liquid Control Pump Trip (P1B), Final = true, Delay = 5:30 minutes	**TRG11**
10)	**RP12A**, RRCS Division Failure (Div I), Final = true	**Inserted**

11) **RP12B**, RRCS Division Failure (Div II), Final = true — **Inserted**

12) **RP13A**, RRCS Inadvertent Alternate Rod Insertion (Div I), Final = true — **TRG25**

13) **RP13B**, RRCS Inadvertent Alternate Rod Insertion (Div II), Final = true — **TRG26**

b. Remotes

1) **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, Final = 35, Ramp = 45 seconds — **TRG30**

2) **MS06A,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148A), Final = defeated — **TRG20**

3) **MS06B,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148B), Final = defeated — **TRG20**

4) **MS06C,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148C), Final = defeated — **TRG20**

5) **MS06D,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148D), Final = defeated — **TRG20**

c. Overrides

1) **DO-1225**, OFF SAFETY/RELIEF VALVE PSV 129 GREEN, Final = off — **Inserted**

2) **DO-1226**, OFF SAFETY/RELIEF VALVE PSV 129 RED, Final = off — **Inserted**

3) **DI-3963**, Alt_A RPS Power Selector Switch, Final = off — **Inserted**

4) **DI-3964**, Alt_A RPS Power Selector Switch, Final = off — **Inserted**

d. Annunciators

1) None

e. Event Triggers

Event #	Event Action	Command
1	zdrdserd(125)==1 (Activates when control rod 38-39 is selected)	None

Event #	Event Action	Command
10	hzlslcspump1a(2)==1 (Activates when SLC pump 1A red light energizes)	None
11	hzlslcspump1b(2)==1 (Activates when SLC pump 1B red light energizes)	None
15	rdvposb(125)==736 (Activates when control rod 38-39 reaches position 46)	dmf rd11

f. Equipment Out of Service

1) APRM 1 bypassed

2) 2MSS*PSV129 inop

g. Support Documentation

1) Provide N2-OP-8 section G.1.0, completed up to step G.1.6

2) Provide ReMA for plant shutdown

3) Provide N2-OP-101C completed up to step G.1.9.

4) Provide N2-OP-101D section G completed up through step G.1.19

5) Provide N2-OP-96 section F.3.0

h. Miscellaneous

1) Place clearance reference tag on the 2MSS*PSV129 control switch

2) Place clearance reference tag on APRM bypass switch for APRM 1

C. Shift Turnover Information

1. Reactor Power: 65%
2. Rodline: Below 100%
3. Technical Specification LCOs in effect:

a. None

NOTES

4. Significant Problems / Abnormalities / Equipment Out of Service:

   a. Plant shutdown is in progress.

   b. APRM 1 is bypassed.

   c. 2MSS*PSV129 is inoperable.

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Place the “C” heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Perform through G.1.10 and then hold that evolution.

   b. Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from 48 to 00.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 65% during a plant shutdown
- N2-OP-101C is in progress up to step G.1.9
- N2-OP-101D is in progress up to step G.1.19
- APRM 1 is bypassed due to erratic indication
- 2MSS*PSV129 is inoperable due to a failed 'C' solenoid

PART III: Remarks/Planned Evolutions:

- Place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Perform through G.1.10 and then hold that evolution. An operator is standing by at 2CES-IPNL204.
- Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from position 48 to 00.

Event #1 – Place Heater Drain Pump in Recirculation Mode

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power • Verifiable Actions - Place heater drain pump in recirculation mode • Final (expected) operating result - Heater drain pump in recirculation mode

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1
Role Play As EO directed to lower tape setting (N2-OP-8, G.1.8), immediately **insert** the following **remote function**: **TRG30** **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, Final = 35, Ramp = 45 seconds Once the remote setpoint reaches 35, contact the control room and report that the tape setting is at 35% and 2HDL-LV24C is closed. **Role Play** As EO, observe the position of LV24C on the FW06, Extraction Steam II display and inform the control room when it begins to open and control level. **Role Play** As EO directed to lower tape setting (N2-OP-8, G.1.10), immediately **modify** the following **remote function** as follows: **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, FV=10, RT=1:00 Once the remote setpoint reaches 10, contact the control room and report that the tape setting is at 10%.	**BOP** • Acknowledges direction to place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1 • At 2CEC*PNL851, slowly raise 2HDL-LV4C, FD WTR HTR 2CNM-E4C WTR LEVEL CONTROL setpoint to 47% • Contacts EO at 2CES-IPNL204 and directs them to slowly lower 2HDL-LV24C, 4TH PT HTR E4C HIGH DR tape setting in Auto UNTIL 2HDL-LV24C begins to open OR tape setting is at 35% • Places 2HDL-LV4C in MAN AND slowly closes the valve WHILE verifying the following: ◦ At 2CES-IPNL204, 2HDL-LV24C, 4TH PT HTR E4C HIGH DR, opens to control heater level. ◦ 2HDL-FV35C, HTR DRAIN P1C RECIRC FV POSN opens AND maintains a minimum system flow of approximately 1400 gpm • Contacts EO and directs them to slowly lower tape setting for 2HDL-LV24C,4th PT HTR E4C controller set to 10%

Event Termination Criteria	• Heater Drain Pump C is in recirculation mode

Event #2 – Insert Control Rods

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power • Verifiable Actions - Insert control rods • Final (expected) operating result - Sufficient control rods inserted for adequate evaluation

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to continue inserting control rods per the shutdown sequence • Provides oversight of reactivity manipulation
Note: The next three control rods to be moved are 38-23, 38-39, and 22-23, going from position 48 to 00. **Note:** The next event (loss of rod position indication) will automatically begin once the 2nd control rod is selected (38-39).	**ATC** • Acknowledges direction to continue inserting control rods per the shutdown sequence • Inserts control rods per N2-OP-96 and the shutdown sequence: ◦ VERIFY there are no rod insert blocks ◦ VERIFY correct rod has been selected per sequence being used ◦ DEPRESS AND HOLD INSERT push button AND OBSERVE INSERT light remains on as long as push button is held down ◦ OBSERVE selected rod moves continuously inward on 4-Rod Display ◦ RELEASE push button AND ALLOW rod to settle into desired notch AND OBSERVE proper cycling of INSERT AND SETTLE lights ◦ MONITOR appropriate nuclear instruments for proper response

Instructor Actions / Plant Response	Operator Actions
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• A sufficient number of control rods have been inserted as determined by the Lead Evaluator

Event #3 – Loss of Control Rod Position Indication

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power - Control rod 38-39 has just been selected • Verifiable Actions - Insert control rod 38-39 to re-establish control rod position indication - Technical Specification 3.1.3 evaluated • Final (expected) operating result - Control rod 38-39 position indication restored

Instructor Actions / Plant Response	Operator Actions
When control rod 38-39 is selected, verify the following **malfunction automatically inserts**: **TRG1 RD11-38-39, Control Rod Failure – RPIS (Any), Final = True** *Control rod 38-39 position indication is lost*	
The following annunciator alarms: • *603443, Control Rod Drift*	**ATC** • Recognize / report control rod 38-39 has lost position indication
Note: During the time that rod position indication is not available (XX indication), TS 3.1.3 Condition C would be applicable. Once valid position indication is achieved, TS 3.1.3 Condition C would be exited. **Spec** | **Condition** | **Applicable Actions** 3.1.3 | C | C.1 & C.2 **Action** | **Description** C.1 | Fully insert inoperable control rod. (3 Hours) C.2 | Disarm the associated CRD. (4 Hours)	**SRO** • Acknowledges report • Directs ATC to execute N2-OP-96 section H.2.0 • Refers to TS 3.1.3 and determines Condition C applies during time position indication is not available • May contact Reactor Engineering and/or plant management

Instructor Actions / Plant Response	Operator Actions
Role Play: If asked to record lit LEDs in 2CEC-PNL615, report that another operator will perform that action. **Role Play:** Once control rod position indication has been restored, if asked as Reactor Engineering and/or plant management about desire to continue control rod insertions, report that it is desired to continue control rod insertions per the shutdown sequence.	**ATC** • Executes N2-OP-96 section H.2.0 • Coordinates with BOP to record lit LEDs in 2CEC-PNL615 • Notifies crew of need to suspend all control rod motion • Determines only one control rod is affected • Inserts control rod 38-39 one notch • Recognizes / reports control rod 38-39 has position indication at position 46 • Continues with control rod insertions per the shutdown sequence, as directed.

Event Termination Criteria	• Control rod 38-39 position indication has been restored • A sufficient number of control rods have been inserted as determined by the Lead Evaluator • Technical Specifications have been addressed

Event #4 – CCP Pump Trips, Standby Pump Fail to Automatically Start

Event Information	• Presumed or required initial plant operating conditions - Plant on-line - CCP system aligned for normal operation with 2CCP-P1A and P1B running, P1C in standby • Verifiable Actions - 2CCP-P1C is started • Final (expected) operating result - CCP system operating with 2CCP-P1B and P1C running

Instructor Actions / Plant Response	Operator Actions
Note: It is desired to wait until no control rods are being moved to insert this event. Wait until operators are signing off ReMA for the rod just moved. **Booth Operator:** When directed by the lead evaluator, **insert** the following **malfunction**: **TRG2** **CW:RA:MT:I**, RBCLC Pump 1A Motor Failure, Final = short The following malfunction is preset: **CW16C**, CCP-P1C Fails to Auto Start, Final = true *The plant responds as follows:* • *2CCP-P1A trips* • *2CCP-P1C fails to automatically start*	

Instructor Actions / Plant Response	Operator Actions
The following annunciators alarm: • *601252, RBCLC Pump 1A/1B/1C Auto Trip/Fail to Start* • *601253, RBCLC Pump 1A/1B/1C Motor Overload* • *601246, Reactor Bldg Closed Loop Cooling Sys Trouble* *The following annunciators may alarm and clear depending upon the time it takes the operators to respond and to manually start the standby RBCLC pump:* • *601254, RBCLC Pump 1A/1B/1C Disch Press Low* • *601255, RBCLCW to RHR Pump 1A/1B/!C Pressure Low* • *601257, RBCLCW to Reactor Recirc Pump A Clrs Pres Low* • *601258, RBCLCW to Reactor Recirc Pump B Clrs Pres Low*	**BOP** • Silences, acknowledges and reports annunciators • Reports trip of 2CCP-P1A • Reports failure of 2CCP-P1C to automatically start
Note: Crew may not enter N2-SOP-13 and simply start 2CCP-P1C due to making an action occur that failed.	**SRO** • Acknowledges reports • Directs entry into N2-SOP-13
Role Play: When contacted by the control room to investigate local indications of pump trip, wait 3 minutes and then report: • If dispatched to pump, no visible problems, however a moderate acrid odor is noticeable in the area of the 2CCP-P1A motor • If dispatched to the breaker, no visible problems, breaker (2NNS-SWG012-5) is tripped, the 50/51 relay is flagged and the 86 Relay on the breaker is rolled	**BOP** • Enters and executes N2-SOP-13 • Performs N2-SOP-13 actions: ◦ Evaluates second set of If/Then statements ◦ Determines that a standby CCP pump failed to auto start ◦ Starts 2CCP-P1C ◦ Reports to SRO that the standby CCP Pump (CCP-P1C) has been manually started • Continues with the actions of N2-SOP-13: ◦ Evaluates all 3 legs of N2-SOP-13 - Determines at least 1 main AND booster pump is operating

Instructor Actions / Plant Response	Operator Actions
	- Determines that cooling capacity is no longer degraded - Determines that no further actions are required to stabilize the CCP system ◦ Verifies CCP cooling is restored and that system parameters have stabilized ◦ Monitors loads cooled by RBCLC ◦ Dispatches an operator to investigate locally (breaker and pump) ◦ Verifies that the system is operating per N2-OP-13, F.1.0 • Perform ARP actions for tripped CCP pump ◦ Place 2CCP-P1A in Pull-to-Lock or Stop

Event Termination Criteria	• 2CCP-P1C started

Event #5 – RBM Fails Inoperable

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Bypass RBM A • Final (expected) operating result - Operating with RBM A bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **remote**: **TRG3** **NM19A**, Rod Block Monitor Failure - Inop (A), Final = true	
The following annunciators alarm immediately after event initiation: • *603204 RBM UPSCALE/ INOPERABLE* • *603442 CONTROL ROD OUT BLOCK*	**ATC** • Silences, acknowledges and reports annunciators • Recognizes / reports inop failure of RBM A
Note: Although RBM A is inoperable, it is only required to be operable when no peripheral (edge) rod is selected. If an edge rod is not already selected, the SRO may direct the RO to select an edge rod to exit the applicability requirements of TS 3.3.2.1. To continue the plant shutdown, it would be required to select a non-edge rod at some point.	**SRO** • Acknowledges reports • Directs execution of ARPs • Directs bypassing RBM A per N2-OP-92 • References TS 3.3.2.1 Condition A and determines a 24 hour LCO applies • May direct selecting an edge rod to change the applicability (If not already selected)

Spec	Condition	Applicable Actions
3.3.2.1	A	A.1
Action	**Description**	
A.1	Restore RBM channel to operable status (24 Hours)	

Instructor Actions / Plant Response	Operator Actions
Role Play: As booth when contacted for indications on RBM A Interface Module Top, provide the following information: LED indications on A3 are as follows: • PWR 1-Lit • PWR 2-Not Lit • PWR A-Not Lit • PWR B-Not Lit LED indications on the cards: • A4-Not Lit • A5- Not Lit • A6- Not Lit • A7-Lit • A8- Not Lit • A9- Not Lit • A10- Not Lit **Note:** If the SRO decides to not bypass RBM A, at the Lead Evaluator's discretion, call up the SRO as the SRO and direct bypassing RBM A. **Role Play:** When contacted to provide indications at P608, inform the operator that BYPASS MANUAL is displayed in inverse video header for RBM A	**ATC** • Executes ARPs • Determines RBM A is inop • Completes N2-OP-92 Attachment 4 • Informs SRO of the results of Attachment 4 • Informs SRO that the RBM may be bypassed per N2-OP-92 • Bypasses RBM A per N2-OP-92 section H.2.0 • Determines no other RBMs are bypassed • Places the RBM bypass joystick to the 'A' position • Verifies RBM A BYPASS light is lit at P603 • Verifies BYPASS MANUAL is displayed in inverse video in the header for RBM A at H13-P608

Event Termination Criteria	• RBM A bypassed • Technical Specifications addressed

Event #6: Loss of Main Condenser Vacuum

Event Information	• Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Lower Reactor power - Attempt manual scram • Final (expected) operating result - Manual scram attempted

Instructor Actions / Plant Response	Operator Actions
When directed by lead evaluator **insert** the following **malfunction**: **TRG4** **MC01**, MAIN CONDENSER AIR INLEAKGE, Final = 50, Ramp = 10 minutes	
• *Offgas flows rise* • *Condenser vacuum begins to lower* • *MWe lowers* • *MWth lowers* *The following annunciator alarms ~5 minutes after event initiation:* • *851358, TURBINE CNSR A/B/C VACUUM LOW* • *851359, LOAD DEPENDENT CNSR VACUUM LOW* **Note:** *Offgas flow rate will be high >65 scfm after 2 minutes. The Main Turbine will trip on low vacuum about 6 minutes after the start of the event.*	**Crew** • Recognizes / reports degrading Main Condenser vacuum

Instructor Actions / Plant Response	Operator Actions
Note: Once the manual Reactor scram is attempted, further actions are described in the next event.	**SRO** • Acknowledges report of degrading condenser vacuum • Directs BOP to enter N2-SOP-09 • May direct ATC to lower power using rods • Directs manual Reactor scram • Directs entry into N2-SOP-101C
	ATC • Lowers Reactor power using Recirc flow • May insert cram rods • Acknowledges SRO direction to scram the reactor • Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603. • Places the Reactor Mode Switch in the SHUTDOWN position • Recognizes / reports failure of RPS to insert rods
	BOP • Acknowledges direction to enter N2-SOP-09 • Determines ATC is lowering power • Verifies proper operation of the following: ◦ SJAE ◦ Offgas ◦ Circulating Water • Determines condenser vacuum cannot be maintained above 22.1”Hg • Informs the SRO that N2-SOP-09 directs tripping the turbine per N2-SOP-21

Event Termination Criteria	• Manual Reactor scram attempted

Event #7 and 8 – Electrical ATWS; SLC Fails to Start on RRCS, Delayed SLC Pump Trips

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power - Manual Reactor scram attempted • Verifiable Actions - Reactor power lowered - Control rods inserted • Final (expected) operating result - Reactor power <4% - Control rod insertion completed • Mitigation Strategy code (if applicable) - DMS-AT-01, high power ATWS (greater than APRM downscale); heat addition to suppression pool or torus requires entry into level power control; RPV level controlled below feedwater spargers; RPV blowdown not required
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5 (1-3): ◦ Rapidly lowering RPV water level ◦ Tripping RCS pumps ◦ Injecting boron The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram. 2. Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5 (1-3). Control rod insertion must be completed within one hour of the start of the failure to scram. **Note:** For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.

Instructor Actions / Plant Response	Operator Actions
The following malfunctions are preset to cause the failure to scram: **RP03**, Reactor Protection System Failure to Scram, Final = true **RP12A**, RRCS Division Failure (Div I), Final = true **RP12B**, RRCS Division Failure (Div II), Final = true	

Instructor Actions / Plant Response	Operator Actions
The following malfunctions will automatically insert on TRG10 or TRG11 to cause the trip of the SLC pumps: **TRG10** **SL01A**, Standby Liquid Control Pump Trip (P1A), Final = true, Delay = 5:30 minutes **TRG11** **SL01B**, Standby Liquid Control Pump Trip (P1B), Final = true, Delay = 5:30 minutes	
	Crew • Recognizes / reports failure to scram
Role Play: If requested prevent the main turbine trip from RCIC (N2-EOP-6.2) **manually insert** the following **remote**:	**SRO** • Enters & executes N2-EOP-RPV (1-3) ◦ Informs crew of N2-EOP-RPV (1-3) entry ◦ Determines the following: ▪ Scram was initiated ▪ Failure to scram has occurred • Exits N2-EOP-RPV (1-3) and enters & executes N2-EOP-C5 (1-3) • Informs crew that reactor has failed to scram, exiting N2-EOP-RPV (1-3), and entering N2-EOP-C5 (1-3) • Determines Reactor power is >4% • Directs rapidly lowering RPV water level (Detail J): ◦ Directs inhibiting ADS ◦ **Directs terminating and preventing all injection into the RPV except boron, CRD, and RCIC (CT-1.0)** ◦ Directs maintaining RPV water level between -39” and 100” using Condensate/Feedwater, HPCS, RCIC, SLS, and/or CRD ◦ **Directs boron injection using SLS (CT-1.0)** ◦ May direct preventing Main Turbine trip from RCIC (N2-EOP-6.2) ◦ May direct bypassing the following isolations per N2-EOP-6.10

Instructor Actions / Plant Response	Operator Actions
RC10, Defeat RCIC/MT Trip Intlk (EOP-6 ATT 2), Final = Defeated **Note:** Main Condenser vacuum will stabilize above 8.5" Hgv, such that the Main Condenser remains available. If the crew has concern over Main Condenser vacuum, they may eventually close MSIVs and transfer Reactor pressure control to RCIC/SRVs.	▪ MSIV low water level ▪ Offgas high radiation • Power Leg ◦ Verifies Mode Switch in SHUTDOWN ◦ Directs initiating RRCS (N2-EOP-6.13) ○ Acknowledges RRCS fails to start SLC pumps ○ Directs manual start of SLC pump(s) ◦ Directs reducing Recirc flow to minimum ◦ **Directs tripping Recirc pumps (CT-1.0)** ◦ **Directs inserting control rods per N2-EOP-6.14 (CT-2.0)** • Level Leg: ◦ Directs verification of auto actions per N2-EOP-6.1 for isolations, ECCS, and DGs ◦ Determines the Main Condenser is available ◦ May direct bypassing the following isolations per N2-EOP-6.10 ○ MSIV low water level ○ Offgas high radiation ◦ Directs maintaining RPV water level between -39" and 100" using Condensate/Feedwater, HPCS, RCIC, SLS, and/or CRD • Pressure Leg: ◦ Determines no SRVs are cycling ◦ Directs RPV pressure band of 800 - 1000 psig using EHC ◦ Directs restoring pneumatics to the drywell ◦ Assigns critical parameter for monitoring / reporting the following: ○ Suppression pool temperature (110°F is a key value for BIIT) • May enter N2-EOP-PC (1-3) due to high Suppression Pool temperature ◦ May direct initiation of Suppression Pool Cooling ◦ May direct start of 5th Service Water pump

Instructor Actions / Plant Response	Operator Actions
	ATC • Acknowledges direction to initiate RRCS per N2-EOP-6.13 • Manually initiate RRCS as follows: ◦ At PNL603, arms and depresses the Division I and II A and B RRCS MANUAL INITAITION SWITCHES ◦ Determines the Division I and II ARI initiation lights are lit ◦ Determines SLC did not initiate after 98 seconds and WCS did not isolate as expected ◦ Informs SRO that RRCS failed to start SLC pumps ◦ **Manually places the keylock switches for SLS P1A and P1B in start (CT-1.0)** ◦ Verifies other SLC pump starts and WCS isolates • Informs the SRO that one SLC pump is running and WCS is isolated • May recognize / report trip of SLC pumps (later) • **Terminates and prevents panel 603 (CT-1.0):** ◦ VERIFIES controller 2FWS-HIC1600 is in manual with 0% output ◦ VERIFIES the following controllers are in manual with 0% output: ○ 2FWS-HIC1010A ○ 2FWS-HIC1010B ○ 2FWS-HIC1010C ○ 2FWS-LIC1055A ○ 2FWS-LIC1055B ○ 2CNM-LIK1137 • Informs the SRO that PNL603 is terminated and prevented. • **Allows RPV level to lower to <100 inches and then reestablishes feed flow and an RPV level band as directed (CT-1.0)** • **Trips Recirc pumps (CT-1.0)**

Instructor Actions / Plant Response	Operator Actions
Note: Multiple attempts to scram the Reactor electrically will fail. Control rods will insert when the scram air header is manually vented. Manual insertion of control rods using RMCS will also be possible. **Role Play:** When dispatched as EO to manipulate POWER SOURCE SELECTOR at 2CEC*PNL610, acknowledge direction, wait 2 minutes, then report task completion. **Role Play:** When dispatched as EO to vent scram air header using Div 1 & 2 ARI, acknowledge direction, wait at least 2 minutes and until Lead Examiner is ready for control rods to insert, then **insert TRG25 and TRG26** and report task completion **Role Play:** When dispatched as EO to de-energize scram solenoids with HCU SRI switches, acknowledge direction. **Role Play:** When dispatched as EO to pull fuses in 2CEC*PNL609 and 2CEC*PNL611, acknowledge direction.	• As time permits, verifies auto actions occurred as expected • **Inserts control rods per N2-EOP-6.14 (CT-2.0):** • Determines 8 white scram solenoid power lights are NOT off • Performs the following at 2CEC*PNL610: ◦ Positions POWER SOURCE SELECTOR switch to ALT A ◦ Positions POWER SOURCE SELECTOR switch to ALT B ◦ Positions POWER SOURCE SELECTOR switch to NORM • Determines 8 white scram solenoid power lights are NOT off • Performs concurrently: ◦ Determines blue SCRAM lights NOT lit on Full Core Display: ▪ **Dispatches EO to vent scram air header using Div 1 & 2 ARI** ◦ Continues at Arrow D to de-energize scram solenoids with HCU SRI switches: ◦ Determines scram discharge volume vent and drain valves are open ▪ Dispatches EO to pull fuses in 2CEC*PNL609 and 2CEC*PNL611

Instructor Actions / Plant Response	Operator Actions
Role Play: If directed to defeat the RCIC level 8 logic, wait 1 minute then **manually insert** the following **remote**: **RC02A**, RCIC Level 8 Trip Defeat:Withdraw Trip Units N693A,E, Final = Defeated **Role Play** If contacted to defeat MSIV low water level isolation per N2-EOP-6.10, **insert** the following **remotes**: **TRG20** **MS06A**, Defeat Level One Isolation of MSIVs (Jumper K148A), Final = Defeated **MS06B**, Defeat Level One Isolation of MSIVs (Jumper K148B), Final = Defeated **MS06C**, Defeat Level One Isolation of MSIVs (Jumper K148C), Final = Defeated **MS06D**, Defeat Level One Isolation of MSIVs (Jumper K148D), Final = Defeated Wait 1 minute and report that the MSIV low water level jumpers have been installed. **Role Play** If contacted as field operator to defeat the Offgas high radiation isolation per N2-OP-42 Attachment 6, wait 1 minute and **insert** the following **remote**: **OG03**, Off Gas High Radiation Isolation Defeated, Final = Defeated	**BOP** • Inhibits ADS: ◦ Places DIV I ADS AUTOMATIC INITIATION DISABLE switch to ON and verifies the associated white light is lit ◦ Places DIV II ADS AUTOMATIC INITIATION DISABLE switch to ON and verifies the associated white light is lit ◦ Verifies 601521 DIVISION I ADS AUTOMATIC INITIATION DISABLED is lit ◦ Verifies 601522 DIVISION II ADS AUTOMATIC INITIATION DISABLED is lit • Informs SRO that ADS is inhibited • **Terminates and prevents panel 601 (CT-1.0):** ◦ Places 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ Arms and depresses LPCI A/LPCS MANUAL INITIATION pushbutton ◦ Closes and overrides 2RHS*MOV24A, LPCI A INJECTION VLV and 2CSL*MOV104, PMP 1 INJECTION VLV ◦ Places 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK ◦ Arms and depresses LPCI B & C MANUAL INITIATION pushbutton ◦ Closes and overrides 2RHS*MOV24B, LPCI B INJECTION VLV and 2RHS*MOV24C, LPCI C INJECTION VLV • Informs the SRO that PNL601 is terminated and prevented. • If directed, prevents Main Turbine trip from RCIC ◦ Removes relay E51A-K102 in 2CEC*PNL613 ◦ Delivers relay to SRO

Instructor Actions / Plant Response	Operator Actions
Then report that the Offgas high radiation isolation has been defeated.	• Informs SRO that Main Turbine trip from RCIC is defeated • If directed, defeats MSIV low water level isolations at 2CEC*PNL609 (611) ◦ Installs the following EOP Jumpers: - #20 from relay B22H-K148A to jumper block (P609) - #18 from relay B22H-K148C to jumper block (P609) - #13 from relay B22H-K148B to jumper block (P611) - #12 from relay B22H-K148D to jumper block (P611) ◦ Verifies 2IAS*SOV166, LOCA OVERRIDE VLV switch in OVERRIDE (P851) ◦ Verifies open 2IAS*SOV166 (P851) ◦ IF required, defeats OFG system high radiation isolation AND restores OFG system per N2-OP-42, Attachment 6 - Dispatches field operator to lift lead at 2OFG-IPNL122 • Informs SRO that MSIV low water level isolation is defeated • If defeated, informs SRO that OFG high radiation isolation is defeated • Maintains RPV pressure 800 - 1000 psig using EHC or appropriate pressure band using SRV's if the MSIVs close. • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: - PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE - OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 - PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD - OPENS 2IAS*SOV164

Instructor Actions / Plant Response	Operator Actions
	- PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD - OPENS 2IAS*SOV165
Role Play: If dispatched as field operator to perform pre-start checks on 2SWP-P1A(B-E), wait 3 minutes then report that 2SWP-P1A(B-E) is ready for start and that all pre-start checks are completed. **Role Play:** If requested to place SWP radiation monitor RE23A in service, wait 2 minutes and **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, FINAL=On **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, FINAL=On Then report 2SWP*RE23A is in service. **Role Play:** If requested to place SWP radiation monitor RE23B in service, wait 2 minutes **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, FINAL=On **RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, FINAL=On Then report 2SWP*RE23B is in service.	**ATC/BOP** • If directed, initiates Suppression Pool Cooling per N2-OP-31 section F.4.0: ◦ Notifies Shift Manager to declare RHS A(B) LPCI mode inoperable ◦ Directs radiation protection department to start SWP radiation monitors 2SWP*RE23A(B) ◦ Opens 2SWP*MOV90A(B) ◦ May start 2SWP*P1E(F) ◦ Throttles open 2SWP*MOV33A(B) to establish 7400 gpm service water flow to RHR A(B) ◦ Verifies 2RHS*MOV24A(B) closed ◦ Verifies 2RHS*MOV4A(B) open ◦ Starts 2RHS*P1A(B) ◦ Throttles open 2RHS*FV38A(B) to establish 7450 gpm RHR A(B) total flow ◦ Verifies 2RHS*MOV4A(B) closed ◦ May throttle 2RHS*MOV8A(B) closed

Event Termination Criteria	• Reactor power downscale on APRMs. • Control rods insertion completed. • Reactor level being controlled in assigned band.

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-3
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 100% power. 2MSS*PSV129 is inoperable due to failed 'A' and 'B' solenoids. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.		
Turnover:	Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7. Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA.		
Critical Tasks:	CT-1: Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout.		
	CT-2: Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV. Reactor water level must not be below -14" for more than 15 minutes.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Secure Raising Suppression Pool Water Level with HPCS N2-OP-33
2	N/A	R – ATC, SRO	Lower Reactor Power with Recirculation Flow N2-OP-101D
3	Overrides	C – SRO TS – SRO	SLC Squib Valve Continuity Loss ARP 601701, Technical Specifications
4	NM12	I – ATC, SRO	APRM Fails Downscale N2-OP-92
5	AD05	C – All MC – BOP TS – SRO	ADS SRV Inadvertently Opens N2-SOP-34, N2-SOP-101D, Technical Specifications
6	ED02 EG01	C – All	Loss of Offsite Power N2-SOP-03, N2-SOP-101C, N2-EOP-RPV
7	DG04	C – BOP, SRO MC – BOP	Emergency Diesel Generators Fail to Automatically Start N2-SOP-03, N2-SOP-01
8	RR20	M – All MC – All	Coolant Leak in Drywell N2-EOP-RPV, N2-EOP-PC
9	RC01 CS02 RH14	C – All MC – All	RCIC Fails to Automatically Start, HPCS Fails to Automatically Inject, Low Pressure ECCS Systems Fail to Automatically Inject N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-3**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 7 & 9**	2	
2. Abnormal events (2-4) **Events 5 & 6**	2	
3. Major transients (1-2) **Event 8**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV, EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)	0	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 3** Revision: **0.0**

Title: **November 2023 NRC Scenario 3**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/11/23
	N/A / Dan Delaney	7/11/23
	N/A / Mike Carno	7/11/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-33, High Pressure Core Spray System
2. N2-OP-101D, Power Changes
3. N2-OP-92, Neutron Monitoring
4. N2-SOP-34, Stuck Open Safety Relief Valve
5. N2-SOP-101D, Rapid Power Reduction
6. N2-SOP-29, Sudden Reduction in Core Flow
7. N2-SOP-08, Unplanned Power Changes
8. N2-SOP-03, Loss of AC Power
9. N2-SOP-101C, Reactor Scram
10. N2-EOP-RPV (1-3), RPV Control - Flowchart
11. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
12. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
13. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
14. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
15. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 100% power with 2MSS*PSV129 inoperable. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.

   b. Event 1 is the normal evolution performed by the BOP operator to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0.

   c. Event 2 is a reactivity evolution. The ATC operator will lower Reactor power to 95% using Recirculation flow per the provided ReMA.

   d. Event 3 occurs when an SLC squib valve loses continuity. The SRO will evaluate Technical Specifications.

   e. Event 4 occurs while the power reduction is in progress. APRM 2 will fail downscale. The SRO will evaluate Technical Specifications. The crew will bypass APRM 2 per N2-OP-92.

   f. Event 5 occurs when 2MSS*PSV127, an ADS SRV, inadvertently opens. The crew will execute N2-SOP-34. The crew to reduce Reactor power to approximately 90% per N2-SOP-101D. The crew will be able to close the open SRV. During the transient heat will be added to the suppression pool, requiring entry into N2-EOP-PC (1-3). The crew will place Suppression Pool cooling in service. The SRO will evaluate Technical Specifications.

   g. Events 6 & 7 begin when a loss of all off-site power occurs. 115 KV line 5 will be lost first, followed shortly by 115KV line 6. A Main Generator load reject results in a Reactor scram. Div 1 and II Diesel Generators will fail to automatically start. The crew must manually start one or more Diesel Generators *(critical task)*. The crew will

execute N2-SOP-03, N2-SOP-101C, and N2-EOP-RPV (1-3) to stabilize the plant.

h. Events 8 & 9 occur when a coolant leak develops in the Drywell. RCIC, HPCS, and low pressure ECCS will fail to automatically initiate. The crew will control injection system to restore and maintain Reactor water level *(critical task)*.

2. Termination Criteria

a. All control rods inserted.

b. Div I and/or Div II EDG powering their respective bus(es).

c. Reactor water level being controlled above top of active fuel.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because loss of all AC power significantly degrades the ability to control the plant.*
Initiating Cue:	*Multiple annunciators and equipment losses will indicate the loss of all AC power. N2-SOP-03 and N2-SOP-01 provide direction to start an EDG and restore power.*
Success Path:	*Manually start Div I and/or Div II EDG.*
Performance Feedback:	*Restoration of AC voltage and multiple systems will provide performance feedback to the crew regarding the start of EDG(s).*
Measurable Performance Standard:	*Manipulation of EDG controls will provide observable action for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

CT-1.0, Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because maintaining Reactor water level above -14 inches ensures adequate core cooling through the preferred method of core submergence. This protects the integrity of the fuel cladding.*

Initiating Cue:	*Multiple Reactor water level indicators and annunciators will provide indications of lowering Reactor water level. N2-EOP-RPV (1-3) provides multiple procedure steps directing injection.*
Success Path:	*Manually injection with RCIC, HPCS, SLC, and/or low pressure systems.*
Performance Feedback:	*Multiple Reactor water level indicators and annunciators will provide performance feedback regarding the success of injection with injection systems.*
Measurable Performance Standard:	*Manipulation of pumps and/or valves in the injection system(s) will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

a. CT-2.0, Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV (1-3). Reactor water level must not be below -14" for more than 15 minutes.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-SE-01, station blackout

   b. DMS-SE-02, loss of off-site power or loss of grid

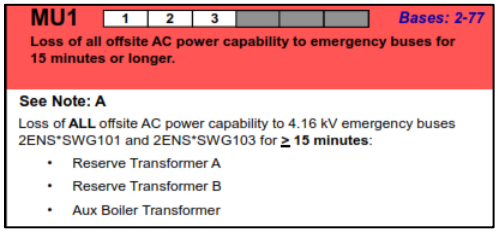
   c. DMS-RL-01, LOCA; RPV level maintained above TAF using high pressure preferred injection and/or high-pressure alternate injection systems; containment parameters controlled; no blowdown

6. Technical Specifications (Applicable actions for initial conditions only)

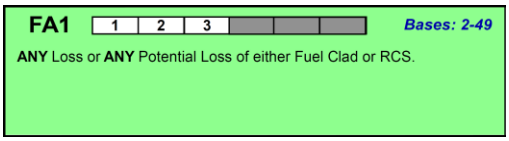
   a. Technical Specification 3.5.1 Condition B (14 days) was entered 1 hour ago.

7. EAL Classification

   a. Unusual Event, EAL MU1:

1) 

b. Alert, EAL FA1:

1) 

B. Initial Conditions

1. IC Number

   a. IC-021 or equivalent (IC-202 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

      1) **NM12B**, APRM Channel Failure – Downscale (2), Final = true — **TRG2**

      2) **ED02A**, Loss Of Off-Site 115KV Line 5, Final = true — **TRG5**

      3) **ED02B**, Loss Of Off-Site 115KV Line 6, Final = true, Delay = 30 seconds — **TRG5**

      4) **DG04A**, EDG 1 fail to UV/LOCA Auto-Start, Final = true — **Inserted**

      5) **DG04B**, EDG 3 fail to UV/LOCA Auto-Start, Final = true — **Inserted**

      6) **EG01**, Main Generator Trip, Final = true, Delay = 30 seconds — **TRG5**

      7) **RR20**, RR Loop Rupture – DBA LOCA, Final = 0.1 — **TRG6**

      8) **RC01**, Reactor Core Isolation Cooling Auto Start Failure, Final = true — **Inserted**

9)	**CS02**, High Pressure Core Spray Fails to Auto Start, Final = true	**Inserted**
10)	**RH14A**, ECCS Fails to Initiate (Div I), Final = true	**Inserted**
11)	**RH14B**, ECCS Fails to Initiate (Div II), Final = true	**Inserted**
12)	**SL02A**, 2SLS*VEX3A Fails to Fire When SLC Pump is Turned On, Final = true	**TRG1**
13)	**ann601701,** div 1 slc system inoperable, Final = crywolf	**TRG1**
b.	Remotes	
1)	**RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = online	**TRG10**
2)	**RM02-041**, SWP23B Current Radiation Level Online, Final = online	**TRG11**
3)	**RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, Final = on	**TRG10**
4)	**RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, Final = on	**TRG11**
c.	Overrides	
1)	**DO-1168**, OFF SQUIB VLV A READY (F004A) (WHITE), Final = off	**TRG1**
2)	**DI-358**, ON SLC SQUIB VALVE VEX3A INOP (AMBER), Final = on	**TRG1**
3)	**DI-1171**, OPEN ADS VALVE PSV 127, Final = on	**TRG3**
4)	**DI-2351**, OPEN ADS VALVE PSV 127, Final = on	**TRG3**
5)	**DI-3215**, INSERT FUSE, Final = off	**Inserted**
6)	**DI-3216**, INSERT 9 SOL B FUSE, Final = off	**Inserted**
7)	**DI-2095**, INSERT FUSE, Final = off	**Inserted**
8)	**DI-2096**, INSERT FUSE, Final = off	**Inserted**

d. Annunciators

1) None

e. Event Triggers

Event #	Event Action	Command
None		

f. Equipment Out of Service

1) 2MSS*PSV129 inop

g. Support Documentation

1) Provide N2-OP-33 section H.3.0 in progress up to step H.3.7.

2) Provide ReMA for lowering power to 95% with Recirc flow

3) Provide N2-OP-101D completed up to step F.3.6.b.3

h. Miscellaneous

1) Place clearance reference tags on the 2MSS*PSV129 control switches on P628 and P631

2) Place a clearance reference tag on the vertical section of P601 near the 2MSS*PSV129 solenoid indicating light

3) Protect RCIC

C. Shift Turnover Information

1. Reactor Power: 100%

2. Rodline: Above 100%

3. Technical Specification LCOs in effect:

a. Technical Specification 3.5.1 Condition B (14 days) was entered 1 hour ago

4. Significant Problems / Abnormalities / Equipment Out of Service:

   a. 2MSS*PSV129 is inoperable.

   b. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0.

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7.

   b. Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 100%
- 2MSS*PSV129 is inoperable due to failed 'A' and 'B' solenoids
- Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.

PART III: Remarks/Planned Evolutions:

- Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7.
- Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA. N2-OP-101D is in progress up to step F.3.6.b.3 in preparation for the power reduction.

Event #1 – Secure Raising Suppression Pool Water Level with HPCS

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~100% power - Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0 • Verifiable Actions - Secure raising of Suppression Pool water level with HPCS • Final (expected) operating result - Reactor at ~100% power - HPCS in standby lineup

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0, starting at step H.3.7 • Once lineup is secured, exits Technical Specification 3.5.1 Condition B
	BOP • Acknowledges direction to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0, starting at step H.3.7: • WHEN desired Suppression Pool Level is reached, THEN CLOSE 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL • WHEN HPCS SYSTEM FLOW is less than 634 GPM, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, is OPEN • STOP 2CSH*P1, HPCS PUMP 1 • VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, is CLOSED • NOTIFY SM that 2CSH*MOV111 is CLOSED AND operability concern PER IR 02083078 no longer exists • RESTORE CST Level, as required PER N2-OP-4

Event Termination Criteria	• Raising of Suppression Pool water level with HPCS secured

Event #2 – Lower Reactor Power with Recirculation Flow

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~100% power • Verifiable Actions - Lower Recirculation flow • Final (expected) operating result - Reactor power at ~95%

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to lower Reactor power to 95% using Recirc flow per ReMA and N2-OP-101D • Provides oversight for reactivity maneuver
Note: T.S. 3.4.1 requires that loop flow mismatch be maintained ≤ 5% of rated core flow (108.5 mlbm/hr) when operating at an effective core flow ≥ 70% of rated core flow. In this case the loop flow mismatch allowable would be 5.425 mlbm/hr.	**ATC** • Acknowledges direction to lower Reactor power to 95% using Recirc flow. • Lowers power to 95% by lowering core flow as follows: ◦ Moves 2RCS*HYV17A&B individually in the close direction, maintaining loop flow differential at a minimal value by alternating between the two valves. • Monitors APRMs and rate of power change
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• Reactor power has been lowered sufficiently as determined by the Lead Evaluator

Event #3 – SLC Squib Valve Continuity Loss

Event Information	• Presumed or required initial plant operating conditions - Both SLC systems in a normal standby lineup • Verifiable Actions - Technical Specifications addressed • Final (expected) operating result - SLC system A declared inoperable

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td><u>Booth Operator</u>
When directed by the lead evaluator, insert the following malfunction and overrides:
TRG1 SL02A, 2SLS*VEX3A Fails to Fire When SLC Pump is Turned On, Final = true
ann601701, div 1 slc system inoperable, Final = crywolf
DO-1168, OFF SQUIB VLV A READY (F004A) (WHITE), Final = off
DI-358, ON SLC SQUIB VALVE VEX3A INOP (AMBER), Final = on</td><td></td></tr>
<tr><td>2SLS*VEX3A loses power/continuity

The following annunciator alarms:
• 601701, DIVISION I STBY LIQ CONT SYSTEM INOPERABLE</td><td><u>ATC</u>
• Recognizes / reports annunciator
• Recognizes / reports loss of power/continuity to 2SLS*VEX3A</td></tr>
<tr><td><table><tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr><tr><td>3.1.7</td><td>A</td><td>A.1</td></tr><tr><th>Action</th><th colspan="2">Description</th></tr><tr><td>A.1</td><td colspan="2">Restore SLC subsystem to operable status (7 days)</td></tr></table></td><td><u>SRO</u>
• Acknowledges failure of 2SLS*VEX3A
• Refers to TS 3.1.7 and determines Condition A applies (7 days)</td></tr>
</table>

Instructor Actions / Plant Response	Operator Actions
Role Play: If requested to check milliamp reading on ammeter C41A-Z1A in back of 2CEC*PNL629, wait 1 minute, then report it indicated 0 mA. **Role Play:** If requested to check 2EHS*MCC102 BKR 16D, wait 2 minutes, then report the breaker is closed but the 10A fuse is blown.	**ATC** • Executes ARP • Coordinates with BOP / field operator to verify: ○ Milliamp reading on ammeter C41A-Z1A (0 amps)

Event Termination Criteria	• Technical Specifications addressed

Event #4 – APRM Fails Downscale

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~95% power • Verifiable Actions - APRM 2 bypassed • Final (expected) operating result - Reactor power at ~95% with APRM 2 bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction**: **TRG2** **NM12B**, APRM Channel Failure – Downscale (2), Final = true *APRM 2 fails downscale*	
The following annunciators alarm: • *603214, APRM TRIP SYSTEM DOWNSCALE* • *603442, CONTROL ROD OUT BLOCK* *The following computer points are generated:* • *RDSBC12, CONTROL ROD OUT BLOCK* • *NMP2C107, APRM DOWNSCALE ALARM* • *NMP2BC23, APRM 2 NEUTRON FLUX DOWNSCALE*	**ATC** • Recognizes / reports annunciators • Recognizes / reports APRM 2 indicates downscale
	SRO • Acknowledges failure of APRM 2 • Evaluates Technical Specification 3.3.1.1 • Directs bypass of APRM 2 per N2-OP-92

Instructor Actions / Plant Response	Operator Actions
	ATC • Bypasses APRM 2 per N2-OP-92 section H.2.0: • Verifies no other APRM in bypass • Performs a channel check to verify no other APRM is inoperable for the division being bypassed (APRMs within 2%) • Places APRM BYPASS joystick to the 2 position • Verifies APRM 2 BYPASS light is lit on 2CEC*PNL603 • Coordinates with BOP to verify the following: ◦ BYP is displayed in inverse video in the header for APRM 2 Chassis at H13-P608 ◦ Blue BYPASSED LED is lit for APRM 2 on each 2/4 MDL at H13-P608
Role Play: As the booth operator, inform the BOP that APRM 2 indicates bypassed on all 2/4 MDL modules.	**BOP** • Monitors plant parameters to verify proper operations • Provides peer checks as needed • Coordinates with ATC to verify the following: ◦ BYP is displayed in inverse video in the header for APRM 2 Chassis at H13-P608 ◦ Blue BYPASSED LED is lit for APRM 2 on each 2/4 MDL at H13-P608

Event Termination Criteria	• APRM 2 is bypassed

Event #5 – ADS SRV Inadvertently Opens

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~95% power - 2MSS*PSV129 inoperable • Verifiable Actions - Close 2MSS*PSV127 - Technical Specification 3.5.1 evaluated • Final (expected) operating result - Reactor at ~90% power - 2MSS*PSV127 closed - Suppression Pool Cooling in service, if required

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **overrides**: **TRG3** **DI-1171**, OPEN ADS VALVE PSV 127, Final = on **DI-2351**, OPEN ADS VALVE PSV 127, Final = on *2MSS*PSV127 opens*	
The following annunciators alarm: • *602553, ADS ACC TANK 32/33/34 PRESSURE LOW (comes in then clears)* • *601537, ADS VALVES / SAFETY VALVES LEAKING* • *601548, SAFETY / RELIEF VALVE OPEN* • *601543, SAFETY RELIEF VALVE SWITCH IN OFF POSITION (When switch taken to OFF)* *The following computer points are generated:* • *SVVBC08, SAF/RLF VLV PSV127 STAT (Open)* • *ADSBC20, ADS VLVS/SAF VLVS LEAK (Alarm)*	**Crew** • Recognizes / reports annunciators • Recognizes / reports 2MSS*PSV127 is open

Instructor Actions / Plant Response	Operator Actions
The following annunciator alarms ~ 3 minutes after event initiation: • *601560, Suppression Pool Water Temp High* *Suppression Pool temperature exceeds 90°F* **Note:** Based on the timeliness of crew actions, the crew may not reach 90°F in the suppression pool and therefore will not enter N2-EOP-PC (1-3).	**SRO** • Acknowledges reports • Directs BOP to execute N2-SOP-34 • Directs ATC to lower Reactor power to 90% per N2-SOP-101D • Provides oversight of reactivity manipulation • May enter N2-EOP-PC (1-3) on high Suppression Pool temperature • May direct initiation of Suppression Pool Cooling • May direct start of 5th Service Water pump • Refers to TS 3.5.1 and determines Condition E applies due to 2 of 7 ADS SRVs inoperable (14 days) • If places one division of RHR in Suppression Pool Cooling, determines TS 3.5.1 Condition F applies (72 hours) • If places two divisions of RHR in Suppression Pool Cooling, determines TS 3.5.1 Condition H applies (enter LCO 3.0.3 immediately) • If Suppression Pool average temperature exceeds 90°F, determines TS 3.6.2.1 Condition A applies (24 hours)

Spec	Condition	Applicable Actions
3.5.1	E	E.1
Action	**Description**	
E.1	Restore ADS valve to operable status (14 days)	

Spec	Condition	Applicable Actions
3.5.1	F	F.1 or F.2
Action	**Description**	
F.1	Restore ADS valve to operable status (72 hours)	
F.2	Restore low pressure ECCS injection / spray subsystem to operable status (72 hours)	

Spec	Condition	Applicable Actions
3.5.1	H	H.1
Action	**Description**	
H.1	Enter LCO 3.0.3 (immediately)	

Spec	Condition	Applicable Actions
3.6.2.1	A	A.1 and A.2
Action	**Description**	
A.1	Verify SP average temp ≤110°F (once per hour)	
A.2	Restore SP average temp ≤90°F (24 hours)	

Instructor Actions / Plant Response	Operator Actions
	ATC • Lowers Reactor power to approximately 90% using Recirc flow per N2-SOP-101D
Note: The candidate may remove the SRV fuses in any order. The scenario is scripted in the order listed in N2-SOP-34. The SRV will close when the 'A' and 'B' solenoid fuses are pulled.	**BOP** • Executes N2-SOP-34: ◦ Identifies 2MSS*PSV127 as the open SRV ◦ Places the keylock switch for 2MSS*PSV127 in Off ◦ Determines 2MSS*PSV127 failed to close by the use of one or more of the following indications: - SPDS Computer - ERF computer points MSSZC111 and MSSZC128 - Reactor power change - Generator output change - Steam flow/Feed flow mismatch - Acoustic monitor ◦ Informs crew of need to reduce Reactor power to approximately 90% ◦ Using detail 2 of N2-SOP-34 removes fuses in any of the following order until the SRV closes: - 'C' solenoid fuses - 'A' solenoid fuses - 'B' solenoid fuses ◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'C' solenoid fuses is panel 628 strip 'A' fuse numbers F49 & F50 ◦ Removes fuses F49 & F50 ◦ Determines that 2MSS*PSV127 did not close by pulling the 'C' solenoid fuses ◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'A' solenoid fuses is panel 628 strip 'K' fuse numbers F5A & F6A ◦ Removes fuses F5A & F6A

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched as field operator to perform pre-start checks on 2SWP-P1A(B-E), wait 3 minutes then report that 2SWP-P1A(B-E) is ready for start and that all pre-start checks are completed. **Role Play:** If requested to place SWP radiation monitor RE23A in service, wait 2 minutes and **insert** the following **remote**: **TRG10** **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, FINAL=On **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, FINAL=On Then report 2SWP*RE23A is in service. **Role Play:** If requested to place SWP radiation monitor RE23B in service, wait 2 minutes **insert** the following **remote**: **TRG11** **RM02-041**, SWP23B Current Radiation Level Online, FINAL=On **RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, FINAL=On Then report 2SWP*RE23B is in service	◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'B' solenoid fuses is panel 631 strip 'C' fuse numbers F5B & F6B ◦ Removes fuses F5B & F6B ◦ Determines that 2MSS*PSV127 did close by pulling the 'A' and 'B' solenoid fuses ◦ Informs crew that 2MSS*PSV127 closed after pulling the 'A' and 'B' solenoid fuses • If directed, initiates Suppression Pool Cooling per N2-OP-31 section F.4.0: ◦ Notifies Shift Manager to declare RHS A(B) LPCI mode inoperable ◦ Directs radiation protection department to start SWP radiation monitors 2SWP*RE23A(B) ◦ Opens 2SWP*MOV90A(B) ◦ May start 2SWP*P1E(F) ◦ Throttles open 2SWP*MOV33A(B) to establish 7400 gpm service water flow to RHR A(B) ◦ Verifies 2RHS*MOV24A(B) closed ◦ Verifies 2RHS*MOV4A(B) open ◦ Starts 2RHS*P1A(B) ◦ Throttles open 2RHS*FV38A(B) to establish 7450 gpm RHR A(B) total flow ◦ Verifies 2RHS*MOV4A(B) closed ◦ May throttle 2RHS*MOV8A(B) closed

Event Termination Criteria	• 2MSS*PSV127 closed • Suppression Pool Cooling is in service, if required • Technical Specifications have been addressed

Events #6, 7, 8, and 9 – Loss of Offsite Power; Emergency Diesel Generators Fail to Automatically Start; Coolant Leak in Drywell; RCIC Fails to Automatically Start, HPCS Fails to Automatically Inject, Low Pressure ECCS Systems Fail to Automatically Inject

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Emergency Diesel Generators started - Reactor water level controlled above top of active fuel - Suppression Pool and Drywell sprays initiated • Final (expected) operating result - Emergency buses energized. - All control rods inserted. - Reactor level being controlled in assigned band. - Containment pressure is being controlled per N2-EOP-PC (1-3). • Mitigation Strategy code (if applicable) - DMS-SE-01, station blackout - DMS-SE-02, loss of off-site power or loss of grid - DMS-RL-01, LOCA; RPV level maintained above TAF using high pressure preferred injection and/or high pressure alternate injection systems; containment parameters controlled; no blowdown
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout. 2. Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV (1-3). Reactor water level must not be below -14" for more than 10 minutes.

Instructor Actions / Plant Response	Operator Actions
The following malfunctions are preset to cause the Div I and II EDGs to fail to start: **DG04A**, EDG 1 fail to UV/LOCA Auto-Start, Final = true **DG04B**, EDG 3 fail to UV/LOCA Auto-Start, Final = true *The following malfunctions are preset to cause RCIC, HPCS, LPCS, and LPCI automatic injection to fail:* **RC01**, Reactor Core Isolation Cooling Auto Start Failure, Final = true **CS02**, High Pressure Core Spray Fails to Auto Start, Final = true **RH14A**, ECCS Fails to Initiate (Div I), Final = true **RH14B**, ECCS Fails to Initiate (Div II), Final = true When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG5** **ED02A**, Loss Of Off-Site 115KV Line 5, Final = true **ED02B**, Loss Of Off-Site 115KV Line 6, Final = true, Delay = 30 seconds **EG01**, Main Generator Trip, Final = true, Delay = 30 seconds	
Note: Due to the loss of offsite power, multiple indications, including on R*Time, will be inaccurate or unavailable.	**Crew** • Recognizes / reports loss of offsite power • Recognizes / reports Reactor scram • Recognizes / reports failure of Div I and II EDGs to automatically start

Instructor Actions / Plant Response	Operator Actions
	ATC • Places the Reactor Mode Switch in the SHUTDOWN position • Arms and depresses RPS pushbuttons • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs closed ◦ Feedwater pumps are NOT running ◦ All control rods fully inserted • Reports EOP entry condition on low RPV water level • Performs N2-SOP-101C actions: ◦ Verifies: ○ All rods full in ○ Reactor power lowering ○ Turbine tripped/TSVs & TCVs shut ○ Generator tripped ○ SDV V & D Valves closed ○ RPV pressure on TBVs or SRVs ◦ Fully inserts IRMs and SRMs

Instructor Actions / Plant Response	Operator Actions
	SRO • Acknowledges reports • Directs entry into N2-SOP-101C • Enters N2-EOP-RPV (1-3) on low Reactor water level • Directs entry into N2-SOP-03 • **Directs start of Div I and II EDGs (CT-1.0)** • Performs N2-EOP-RPV (1-3) actions: ◦ Directs scram procedure entered (N2-SOP-101C) ◦ Performs level control actions: ○ Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1 ○ Determines that RPV water level is known ○ **Directs that RPV water level be restored and maintained between 159.3 and 202.3 inches using RCIC, HPCS, and/or SLC (CT-2.0)** ○ Directs an RPV level band of 160 – 200 inches ○ May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: ○ Determines that SRVs are cycling ○ Determines that RPV pressure can be stabilized below 1052 psig using SRVs ○ Directs a pressure band of 800-1000 psig

Instructor Actions / Plant Response	Operator Actions

Booth Operator:
If directed to defeat the HPCS Level 8 interlocks by placing HPCS INJ VLV LEVEL 8 BYPASS TEST SWITCH in the TEST position, **manually insert** the following **remote function**:
 	
• **CS14**, OPS-CSH01 PNL625 TST.SW CSH*MOV107, Final = Test
 	
Wait 1 minute, then report that the HPCS Level 8 interlocks have been defeated.
 	
Booth Operator:
If directed to defeat the RCIC Level 8 interlocks, **manually insert** the following **remote function**:
 	
• **RC02A,** RCIC Level 8 Trip Defeat: Withdraw Trip Units N693A, E, Final = defeated
 	
Wait 1 minute, then report that the RCIC Level 8 interlocks have been defeated.
 	
Booth Operator:
If directed to defeat the RHR group 5 isolation, **manually insert** the following **malfunction**:
 	
• **RH08,** GROUP 5 ISOLATION FAILURE – (RHS*MOV122/113), Final = true
 	
Wait 1 minute, then report that the RHR group 5 isolation has been defeated.
	ATC / BOP
	• May recognize / report failure of RCIC and/or HPCS to automatically start
	• **Control injection to restore / maintain Reactor water level between 159.3 and 202.3 inches using RCIC, HPCS, and/or SLC (CT-2.0)**
	• **May manually start RCIC (CT-2.0)**:
	  ◦ Arms and depresses RCIC MANUAL INITIATION pushbutton
	  ◦ Verifies proper RCIC response
	  ◦ Controls RCIC operation as follows:
	  ◦ Automatic control with thumbwheel
	  ◦ Manual control using the pushbuttons
	  ◦ If injection with reject to CST is required:
	  ▪ Opens 2ICS*MOV124, TEST BYPASS TO CONDENSATE STORAGE TK
	  ▪ Controls injection flow to the Reactor by throttling 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK
	• **May manually start HPCS (CT-2.0)**:
	• At 2CEC*PNL601, ARM AND DEPRESS MANUAL INITIATION
	• At 2CEC*PNL601, VERIFY the following:
	• 2CSH*P1, HPCS PUMP 1, started.
	• 2CSH*MOV107, PMP 1 INJECTION VLV, open.
	• WHEN system flow is greater than 634 gpm, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV closed.
	• Flow to RPV as indicated by 2CSH*FI104 flow rising AND indication the following valves are closed:
	  ◦ 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL
	  ◦ 2CSH*MOV110, TEST RETURN TO CONDENSATE TK

Instructor Actions / Plant Response	Operator Actions
	◦ 2CSH*MOV112, TEST RETURN TO CONDENSATE TK • At 2CEC*PNL852, VERIFY DIVISION 3 2EGS*EG2 started
Once at least one EDG has been started and connected to its respective bus, and when directed by the lead evaluator, **insert** the following **malfunction**: **TRG6** **RR20**, RR Loop Rupture – DBA LOCA, Final = 0.1	**Crew** • Recognizes / reports degrading Primary Containment parameters
	SRO • Enters N2-EOP-PC (1-3) on high Drywell pressure and temperature • Re-enters N2-EOP-RPV (1-3) on high Drywell pressure • Performs N2-EOP-PC (1-3) primary containment pressure leg actions: • Determines cannot maintain containment pressure below 1.68 psig • Determines that suppression chamber spray is required before suppression chamber pressure reaches 10 psig • Determines that suppression pool water level is below EL 217 ft. • Determines that suppression chamber sprays are warranted • Directs RO to place RHR in suppression chamber sprays • Directs pneumatics restored to the Drywell • May direct low pressure ECCS pumps terminated and prevented with a high drywell pressure and reactor pressure lowering (N2-EOP-RPV (1-3), step P-1) • Directs Suppression Chamber Spray secured prior to Suppression Chamber pressure lowering to 0 psig

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If requested to place SWP radiation monitor RE23A in service, **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = Online **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Power, Final = On If requested to place SWP radiation monitor RE23B in service, **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, Final = Online **RM03-041**, SWP23B Current Radiation Level Power, Final = On Then report 2SWP*RE23A(B) is in service	**ATC / BOP** • Initiates Suppression Chamber spray per N2-EOP-6.22: ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies 2RHS*P1A(B) running. ◦ Opens 2RHS*MOV33A(B) to establish suppression chamber spray flow. ◦ Opens 2RHS*FV38A(B) and establish approximately 7450 gpm – If suppression pool cooling is directed. ◦ Throttles open 2SWP*MOV33A(B) to establish flow not to exceed 7400 gpm - Starts 5th SWP pump (post LOCA). ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay. ◦ Directs RP to place 2SWP*RE23A(B) in service • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE ◦ OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD ◦ OPENS 2IAS*SOV164 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD ◦ OPENS 2IAS*SOV165 • If directed, terminates and prevents Low pressure ECCS injection using N2-EOP-HC attachment 4:

Instructor Actions / Plant Response	Operator Actions
	◦ PLACES 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ IF a LPCI A/LPCS initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI A/LPCS MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24A, LPCI A INJECTION VLV ◦ 2CSL*MOV104, PMP 1 INJECTION VLV ◦ PLACES 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK. ◦ IF a LPCI B & C initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI B & C MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24B, LPCI B INJECTION VLV ◦ 2RHS*MOV24C, LPCI C INJECTION VLV

Event Termination Criteria	• All control rods inserted. • Div I and/or Div II EDG powering their respective bus(es). • Reactor water level being controlled above top of active fuel.

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-4
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 7% power during a startup.		
Turnover:	Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. The procedure is complete up to step F.5.4. Then, continue to withdraw control rods per the startup sequence. The next rod is 34-19 going from 12 to 48.		
Critical Tasks:	CT-1: Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC. The scram must be initiated before Suppression Pool water level reaches 192'. CT-2: Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC. The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Transfer Reboiler Steam Supply to Main Steam N2-OP-25
2	N/A	R – ATC, SRO	Withdraw Control Rods N2-OP-96
3	RW01	I – ATC, SRO TS – SRO	Rod Worth Minimizer Fails Inop ARP 603307, N2-OP-95A, Technical Specifications
4	CU06 CU08	C – BOP, SRO TS – SRO MC – BOP	Reactor Water Cleanup Leak into Reactor Building Closed Loop Cooling System, Automatic Isolation Fails ARP 602320, ARP 602313, Technical Specifications
5	RH13A RH15	C – BOP, SRO TS – SRO MC – BOP	Inadvertent Initiation of Div 1 ECCS with Min Flow Failure ARPs, N2-OP-100A, Technical Specifications
6	PC12	M – All	Suppression Pool Leak N2-EOP-PC, N2-EOP-SC, N2-EOP-RPV, N2-SOP-101C, N2-EOP-C2
7	CS:HP:MT:I	C – BOP, SRO	HPCS Pump Trip, Suppression Pool Makeup from RHR Unavailable N2-EOP-PC
8	TC:PA(B):MT:I	C – BOP, SRO MC – BOP	EHC Pumps Trip N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-4**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 7 & 8**	2	
2. Abnormal events (2-4) **Events 4 & 5**	2	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **N2-EOP-C2**	1	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 4** Revision: **0.0**

Title: **November 2023 NRC Scenario 4**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/12/23
	N/A / Dan Delaney	7/12/23
	N/A / Mike Carno	7/12/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-25, Auxiliary Steam, Auxiliary Condensate, and Gland Seal
2. N2-OP-96, Reactor Manual Control and Rod Position Indication System
3. N2-OP-95A, Rod Worth Minimizer System
4. N2-OP-100A, Standby Diesel Generators
5. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
6. N2-EOP-SC (1-3), Secondary Containment Control – Flowchart
7. N2-SOP-101C, Reactor Scram
8. N2-EOP-RPV (1-3), RPV Control – Flowchart
9. N2-EOP-C2 (1-3), RPV Blowdown - Flowchart
10. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
11. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
12. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
13. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 7% power during a plant startup.

   b. Event 1 is the normal evolution performed by the BOP operator to transfer Reboiler steam to Main Steam per N2-OP-25 section F.5.0.

   c. Event 2 is a reactivity evolution. The ATC operator will continue the startup by withdrawing control rods per the startup sequence.

   d. Event 3 occurs when the Rod Worth Minimizer (RWM) fails. This will result in a control rod block. The SRO will address Technical Specifications. The crew will bypass the RWM to allow continued control rod withdrawals.

   e. Event 4 occurs when WCS begins to leak into the CCP system in the NRHX. WCS will fail to automatically isolate. The crew will manually isolate WCS to stop the leak.

   f. Event 5 occurs when an inadvertent Division I ECCS signal is received. This causes the CSL and RHR 'A' pumps to automatically start and run on minimum flow. During the transient, 2RHS*MOV4A (2RHS*P1A minimum flow valve) fails closed. The crew will evaluate using redundant and independent indications that the ECCS signal is not valid and determine that 2RHS*P1A is running at shutoff head. The crew will then place 2RHS*P1A in P-T-L. The SRO will evaluate Technical Specifications for the inoperability of two ECCS injection systems.

   g. Events 6-8 begin when a leak develops from the Suppression Pool. The crew will enter N2-EOP-SC (1-3) and N2-EOP-PC (1-3). The crew will initiate makeup to the Suppression Pool, but RHR will be unavailable for makeup and HPCS will trip shortly after makeup is initiated. Suppression Pool water level will continue to lower. The

crew will scram the Reactor before Suppression Pool water level reaches 192' *(critical task)*. Suppression Pool water level will continue to lower. EHC pumps will trip when Reactor pressure lowers below 800 psig, limiting the crew's ability to depressurize using Turbine Bypass Valves. Suppression Pool water level will not be able to be maintained above 192', so the crew will perform an RPV Blowdown per N2-EOP-C2 (1-3) *(critical task)*.

2. Termination Criteria

   a. All control rods in.

   b. RPV Blowdown in progress or completed.

   c. Reactor level being controlled in assigned band.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because, given lowering Suppression Pool water level, a Reactor scram is required to allow subsequent action to lower Reactor pressure.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Suppression Pool water level is lowering and cannot be maintained above 192'.*
Success Path:	*Arming and depressing RPV pushbuttons and/or rotating the Reactor Mode Switch to SHUTDOWN*
Performance Feedback:	*Control rod position and Reactor power indications will provide performance feedback regarding the success of the scram.*
Measurable Performance Standard:	*Manipulating RPS pushbuttons and/or the Reactor Mode Switch will provide the evaluation team with observable actions.*
Bounding Conditions:	*N2-EOP-PC (1-3) directs entering N2-EOP-RPV (1-3) and scramming the Reactor before Suppression pool water level drops to 192' (bottom of scale).*

   a. CT-1.0, Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC (1-3). The scram must be initiated before Suppression Pool water level reaches 192'.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because loss of Suppression Pool water inventory jeopardizes the ability of the Primary Containment to handle the energy contained in the Reactor Coolant system. An RPV Blowdown is required to lower the energy state of the Reactor Coolant system in this situation.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Suppression Pool water level is lowering and cannot be maintained above 192'.*
Success Path:	*Manually initiating ADS and/or opening SRVs*
Performance Feedback:	*SRV and/or TBV instrumentation will provide indication that these valves are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.*
Measurable Performance Standard:	*Manipulation of ADS/SRV controls will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

b. CT-2.0, Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC (1-3). The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-PC-02, loss of inventory in suppression pool (Unit 2) or torus (Unit 1); RPV blowdown or anticipate blowdown and depressurize to the main condenser

6. Technical Specifications (Applicable actions for initial conditions only)

   a. None

7. EAL Classification

   a. FA1:

      1) Any loss of any potential loss of either fuel clad or RCS

B. Initial Conditions

1. IC Number

   a. IC-012 or equivalent (IC-203 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**RW01**, Rod Worth Minimizer Failure, Final = true	**TRG1**
2)	**CU08**, Reactor Water Cleanup Isolation Failure, Final = true	**Inserted**
3)	**CU06**, Reactor Water Cleanup Non-regenerative HX Tube Leak, Final = 50, Ramp = 2 minutes	**TRG2**
4)	**RH13A**, ECCS Inadvertently Initiates (Div I), Final = true	**TRG3**
5)	**RH15**, RHS*MOV4A Valve Fails Shut, Final = true	**TRG3**
6)	**PC12**, Suppression Pool To Reactor Building Leak, Final = 10	**TRG4**
7)	**TC:PA:MT:I**, Turbine EHC Pump A Motor Failure, Final = short	**TRG5**
8)	**TC:PB:MT:I**, Turbine EHC Pump B Motor Failure, Final = short	**TRG5**
9)	**CS:HP:MT:I**, HPCS Motor Failure, Final = short, Delay = 30 sec	**TRG30**

   b. Remotes

      1) None

   c. Overrides

      1) None

   d. Annunciators

      1) None

e. Event Triggers

Event #	Event Action	Command
5	rrpdome<800 (Activates when Reactor pressure lowers below 800 psig)	None
14	anntbl(1039)>1	imf pc12 100
30	hzlcshps2(2)==1 (HPCS pump red light on)	None

f. Equipment Out of Service

1) None

g. Support Documentation

1) Provide N2-OP-25 section F.5.0, completed up to step F.5.4. Also provide section F.1.0.

2) Provide ReMA for plant startup

3) Mark up N2-OP-101A with section E.1.0 initialed off, section E.2.0 initialed off, E.3.0 initialed off, and E.4.0 initialed off up to E.4.10. (open circle step E.4.10).

4) Provide N2-OP-96 section F.2.

h. Miscellaneous

1) None

C. Shift Turnover Information

1. Reactor Power: 7%
2. Rodline: Below 100%
3. Technical Specification LCOs in effect:
   a. None
4. Significant Problems / Abnormalities / Equipment Out of Service:
   a. Plant startup is in progress.

NOTES

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. An EO is standing by at the Auxiliary Boilers and by 2CES-IPNL204. The procedure is complete up to step F.5.4.

   b. Then, raise Reactor power by withdrawing control rods per the startup sequence. The next rod is 34-19 going from 12 to 48 (continuous).

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 7% during a plant startup
- Reactor pressure is 925 psig with 1 bypass valve open
- Feed pump A is running with level control on LV55A
- A Reactor startup is in progress per N2-OP-101A. Currently on step E.4.10. The Clean Steam Reboiler is still on the Aux Boilers.
- Currently on NM2C19-SU-1 sequence page 19, rod 34-19 withdrawing control rods from position 12 to position 48
- Reactor Engineering is present and available in the Control Room

PART III: Remarks/Planned Evolutions:

- Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. Place 'A' Clean Steam Reboiler in service per engineering recommendation. An EO is standing by at the Auxiliary Boilers and by 2CES-IPNL204. The procedure is complete up to step F.5.4.
- Then, continue to withdraw control rods per the startup sequence. The next rod is 34-19 going from 12 to 48 (continuous).
- Another team of Operators will be responsible for other startup activities in N2-OP-101A.

Event #1 – Transfer Reboiler Steam Supply to Main Steam

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Close 2ASS-AOV145, AUX BLR STM INLET VLV - Place the Reboiler 2ASS-STV112 and 2ESS-STV104 control switches in the AUTO • Final (expected) operating result - Operating with reboiler steam supply on main steam

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to transfer Reboiler steam supply to Main Steam per N2-OP-25, Section F.5.0, starting at step F.5.4
Role Play: As EO, when asked, provide indications from the Aux Boilers. Use normal operating indications from N2-OP-25, F.1.0 to respond to any additional requests for indications. **Role Play:** As EO, when asked to locally open 2ASS-V23, wait 1 minute and then report that 2ASS-V23 is open. **Role Play:** As EO directed to depress the reset pushbuttons for 2CNA-LV8A, LV-8B and LV-8C, wait 10 seconds and inform the control room that it has been completed. **Role Play:** As EO directed to confirm that 2CNA-LV8A, LV-8B and LV-8C are modulating to control Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed, wait 1 minute and inform the control room that 2CNA-LV8A, LV-8B and LV-8C are modulating and controlling Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed.	**BOP** • Acknowledges direction to transfer Reboiler steam supply to Main Steam • Directs EO to open 2ASS-V23 • Closes 2ASS-AOV145, AUX BLR STM INLET VLV • Places the Reboiler 2ASS-STV112 and 2ESS-STV104 control switches in the AUTO • Directs EO to depress the RESET PUSHBUTTONS for 2CNA-LV8A, LV-8B and LV-8C on the 2CES-IPNL204 • Opens 2CNA-HV34A • Directs EO to confirm that 2CNA-LV8A, LV-8B and LV-8C are modulating to control Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed • Monitors for proper Reboiler operation • Reports steam supply transferred

Instructor Actions / Plant Response	Operator Actions
Role Play: If asked, inform the candidate that there is no longer any load on the auxiliary boilers. **Role Play:** If asked as the EO about the following, wait 1 minute then report: • At Main Turbine, no steam is blowing from turbine shaft seals • A water level exists in the Gland Exhaust loop seal as indicated by air not being sucked in	

Event Termination Criteria	• Reboiler steam supply on main steam

Event #2 – Withdraw Control Rods

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Withdraw control rods • Final (expected) operating result - Sufficient control rods withdrawn for adequate evaluation

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to continue withdrawing control rods per the startup sequence • Provides oversight of reactivity manipulation
Note: The next four control rods to be moved are 34-19, 34-43, 26-19, and 18-35 going from 12 to 48 (continuous withdraw). **Note:** The next event may be inserted whenever the lead evaluator determines that sufficient control rods have been withdrawn for an adequate evaluation. It should be inserted when no rod motion is occurring.	**ATC** • Acknowledges direction to continue withdrawing control rods per the startup sequence • Reviews N2-OP-96 actions for first time rod selection and withdrawal actions: ◦ SELECTS the rod to be moved by momentarily depressing the pushbutton for that rod on the Rod Select Module at 2CEC*PNL603 ◦ Monitors RPV, CRD and Nuclear Instruments ◦ OBSERVES the following: - The pushbutton for selected rod is lit - The white select light on full core display is lit for control rod - The rod's position is indicated AND in correct location on 4-Rod Display - IF available, THEN Rod Worth Minimizer indicates correct rod AND rod position OTHERWISE MARK this step N/A

Instructor Actions / Plant Response	Operator Actions
Note: A coupling check may be done separately or as the rod is being initially withdrawn per the sequence.	◦ Continuous Withdrawal is performed as follows: - VERIFIES Reactor mode switch is not in SHUTDOWN AND there are no rod withdrawal blocks - VERIFIES correct rod has been selected per sequence being used AND that continuous rod withdrawal is allowed for given conditions - SIMULTANEOUSLY DEPRESSES WITHDRAW AND CONTINUOUS WITHDRAW push buttons AND OBSERVE WITHDRAW AND CONTINUOUS WITHDRAW lights are on • Continues control rod withdrawal per the startup sequence • Monitors NI indications, reactor level and reactor pressure
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• A sufficient number of control rods have been withdrawn as determined by the Lead Evaluator

Event #3 – Rod Worth Minimizer Fails Inop

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power - RWM in service • Verifiable Actions - Bypass RWM - Technical Specification 3.3.2.1 evaluated • Final (expected) operating result - RWM bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction**: **TRG1** **RW01**, Rod Worth Minimzer Failure, Final = true *RWM fails*	
The following annunciator alarms: • *603307, RWM ROD BLOCK*	**ATC** • Recognize / report annunciator 603307

Instructor Actions / Plant Response	Operator Actions

Spec	Condition	Applicable Actions
3.3.2.1	C	C.1 or C.2
Action	**Description**	
C.1	Suspend control rod movement except by scram (immediately)	
C.2	Verify ≥ 12 rods withdrawn (immediately) Or Verify by administrative means that startup with RWM inoperable has not been performed in the last calendar year (immediately AND Verify movement of control rods is in compliance with BPWS by a 2nd licensed operator or other qualified member of the technical staff (during control rod movement)	

Role Play:

If asked for assistance with troubleshooting RWM, report that the RWM has a critical fault and will not be available for the remainder of the startup. If asked, recommend/approve bypassing RWM per N2-OP-95A.

Role Play:

If asked if a startup with an inoperable RWM has been performed in the last calendar year, acknowledge request, wait 2 minutes, then report that a startup with an inoperable RWM has NOT been performed in the last calendar year.

Role Play:

If asked for an additional licensed operator or qualified member of the technical staff to verify control rod movement, acknowledge request.

SRO

- Acknowledges report
- Directs ATC to execute ARP
- Refers to TS 3.3.2.1 and determines Condition C applies
- May contact Reactor Engineering and/or plant management
- Directs ATC to bypass RWM per N2-OP-95A section H.1.0

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched to verify two position Computer Display keylock mode switch is in OPER, wait 1 minute, then report that the switch is in OPER.	**ATC** • Executes ARP 603307 • Bypasses RWM per N2-OP-95A section H.1.0: • Coordinates with SRO to review restrictions • Dispatches operator to verify two position Computer Display keylock mode switch is in OPER position at H13-P615, RPIS Panel • Places three position Operator Display keylock mode switch in BYPASS position at panel 2CEC*PNL603 • Observes BYPASS mode indication

Event Termination Criteria	• RWM bypassed • Technical Specifications have been addressed

Event #4 – Reactor Water Cleanup Leak into Reactor Building Closed Loop Cooling System, Automatic Isolation Fails

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power - WCS in service • Verifiable Actions - Isolate WCS • Final (expected) operating result - Reactor at ~7% power with WCS isolated

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction** (insert when rods are not being moved): **TRG2** **CU06**, Reactor Water Cleanup Non-regenerative HX Tube Leak, Final = 50, Ramp = 2 minutes The following **malfunction** is **preset** to cause the failure to isolate: **CU08**, Reactor Water Cleanup Isolation Failure, Final = true	
The following annunciator will alarm ~70 seconds after the malfunction is inserted: • *602320, RWCU DIFF FLOW TIMER BYPASS.* *The following annunciator will alarm 35 seconds after 602320 alarms:* • *602313, RWCU DIFFERENTIAL FLOW HIGH*	**Crew** • Recognizes / reports annunciators • Recognizes / reports WCS leak into CCP • Recognizes / reports failure of WCS to automatically isolate

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.3.6.1</td><td>A</td><td>A.1</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Place channel in trip (24 hours)</td></tr>
</table>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.3.6.1</td><td>B</td><td>B.1</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>B.1</td><td colspan="2">Restore isolation capability (1 hour)</td></tr>
</table>

Note:
Technical Specification 3.3.6.1 Condition F will also become applicable after 1 hour.

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SRO

- Acknowledges reports
- Directs execution of ARPs
- Directs isolation of WCS
- May direct entry into N2-SOP-83
- Evaluates Technical Specification 3.3.6.1 for failure of WCS to automatically isolate, as necessary:
  - Determines that Actions A and B are applicable
  - Determines the isolation flow path is already isolated

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Note:
The crew may choose to not wait for Annunciator 602313 to alarm. The crew may anticipate the RWCU isolation and take manual actions prior to the actual isolation signal. This is an acceptable action.

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BOP

- Executes ARPs
- May execute N2-SOP-83
- Isolates WCS:
  - Closes 2WCS*MOV102, Cleanup Suct Inboard Isol Vlv
  - Closes 2WCS*MOV112 Cleanup Suct Outboard Isol Vlv
  - Verifies 2WCS-P1A trips
  - Throttles open 2WCS-MOV110, CLEANUP DEMIN BYPASS VLV

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</table>

Event Termination Criteria	• WCS isolated

Event #5 – Inadvertent Initiation of Div 1 ECCS with Min Flow Failure

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Place 2RHS*P1A in P-T-L • Final (expected) operating result - Reactor at ~7% power with 2RHS*P1A in P-T-L

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG3** **RH13A**, ECCS Inadvertently Initiates (Div I), Final = true **RH15**, RHS*MOV4A Valve Fails Shut, Final = true • *Division I low pressure ECCS pumps receive initiation signals and auto start* • *Division I Diesel Generator starts* • *RHS*MOV4A closes* • *LPCI A / LPCS initiation white light illuminates*	
The following annunciators alarm: • *601413, LPCS PUMP 1 AUTO START* • *601426, LPCS SYSTEM ACTUATED* • *601442, RHR PUMP 1A AUTO START* • *601451, RHR A SYSTEM ACTUATED* • *601539, ADS A LPCS / RHR A PERMISSIVE* • *852117, EDG 1 RUNNING*	**BOP** • Silences, acknowledges and reports annunciators • Recognizes / reports Div I low pressure ECCS initiation • Recognizes / reports 2RHS*MOV4A, RHR 'A' minimum flow valve, is closed

Instructor Actions / Plant Response	Operator Actions
Note: The following Technical Specifications apply for placing 2RHS*P1A in P-T-L:	**SRO** • Acknowledges reports • Directs execution of ARPs • Directs 2RHS*P1A placed in P-T-L • May direct 2CSL*P1 placed in P-T-L. • May direct Division I Diesel Generator shutdown per N2-OP-100A • Evaluates Technical Specifications 3.5.1, 3.6.1.6, 3.6.2.3, and 3.6.2.4

Spec	Condition	Applicable Actions
3.5.1	A	A.1
Action	**Description**	
A.1	Restore low pressure ECCS injection/spray subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.1.6	A	A.1
Action	**Description**	
A.1	Restore RHR drywell spray subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.2.3	A	A.1
Action	**Description**	
A.1	Restore RHR suppression pool cooling subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.2.4	A	A.1
Action	**Description**	
A.1	Restore RHR suppression pool spray subsystem to OPERABLE status. (7 days)	

Note:
The following Technical Specification will also apply if 2CLS*P1 is placed in P-T-L:

Spec	Condition	Applicable Actions
3.5.1	C	C.1
Action	**Description**	
C.1	Restore one ECCS injection/spray subsystem to OPERABLE status. (72 hours)	

Note:
Technical Specification 3.3.5.1 would also possibly apply depending on troubleshooting results.

Instructor Actions / Plant Response	Operator Actions
Role Play: If requested to provide RHR and CSL trip unit indications (2CEC*PNL629), report that no trip units are in alarm and that no gross failures are present. **Role Play:** If dispatched as EO to monitor diesel running parameters, wait 5 minutes and then report that all diesel running indications are normal for the Division I EDG.	**BOP** • Executes ARPs • Verifies the validity of the LOCA initiation using at least two redundant indications • Ensures by two separate and redundant means that the initiation is not valid • Checks trip unit indications for status • Places 2RHS*P1A in P-T-L • Dispatches EO to perform running checks on the Div I EDG • If directed, places 2CSL*P1 in P-T-L • If directed, shuts down the Division I diesel generator per N2-OP-100A, Section H.16.0: ◦ Places EMERGENCY DSL GEN 1 LOCA SIGNAL BYPASS switch to ON ◦ Places DIVISION I 2EGS*EG1 START switch in PULL-TO-LOCK

Event Termination Criteria	• 2RHS*P1A has been placed in P-T-L • Technical Specifications addressed

Events #6, 7, & 8 – Suppression Pool Leak; HPCS Pump Trip, Suppression Pool Makeup from RHR Unavailable; EHC Pumps Trip

Event Information	• Presumed or required initial plant operating conditions - Reactor ~7% power • Verifiable Actions - Scram the Reactor - Depressurize the Reactor • Final (expected) operating result - Reactor power <4% - All control rods in - Reactor pressure lowered • Mitigation Strategy code (if applicable) - DMS-PC-02, loss of inventory in suppression pool (Unit 2) or torus (Unit 1); RPV blowdown or anticipate blowdown and depressurize to the main condenser
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC (1-3). The scram must be initiated before Suppression Pool water level reaches 192'. 2. Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC (1-3). The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG4** **PC12**, Suppression Pool To Reactor Building Leak, Final = 10 When annunciator 851443, Reactor Building General Area Tank 2A Flooding, is received (anntbl(1039) > 1), verify the following **malfunction modifies**: **PC12**, Suppression Pool To Reactor Building Leak, Final = 100 *The following malfunctions will automatically insert on **TRG5** when Reactor pressure lowers below 800 psig to cause EHC pumps to trip:* **TC:PA:MT:I**, Turbine EHC Pump A Motor Failure, Final = true **TC:PB:MT:I**, Turbine EHC Pump B Motor Failure, Final = true	
The following annunciators alarm: • *851443, REACTOR BLDG GENERAL AREA TANK 2A FLOODING* • *851444, REACTOR BLDG GENERAL AREA TANK 2E FLOODING* • *851453, REACTOR BLDG FLOOR DRAIN SYSTEM TROUBLE*	**Crew** • Recognizes / reports leakage in Reactor Building • Recognizes / reports lowering Suppression Pool water level

Instructor Actions / Plant Response	Operator Actions
Role Play: When dispatched to inspect the Reactor Building for indications of flooding, wait 3 minutes and then report there is a crack in the Suppression Pool wall on RB 175' elevation (~3 feet long and 5 feet above floor level) and there is a lot of water pouring out. **Role Play:** If contacted as Radwaste, report that both pumps are running for each affected tank (2A and 2E).	**SRO** • Acknowledges reports • Enters N2-EOP-SC (1-3) on high RB floor drain and area water levels: ◦ Determines no HVR isolation setpoint has been exceeded ◦ Directs operating sump pumps to restore and maintain drain sump level below high-high setpoint ◦ Determines cannot remove water from area and transitions to 28 ◦ Attempts to isolate all discharges into affected areas ◦ Determines leak is not from primary system ◦ Monitors for 2 areas above Max Safe Water level (flooding alarm) • Enters N2-EOP-PC (1-3) on low Suppression Pool water level • Directs filling suppression pool (N2-OP-31 section H.6.0 or N2-OP-33 section H.2.0 or H.3.0) • Determines cannot maintain Suppression Pool water level above 199.5’ and transitions to 17 • Enters N2-EOP-RPV (1-3) • **Directs manual Reactor scram (CT-1.0)** • Directs entry into N2-SOP-101C • Acknowledges scram report • Performs N2-EOP-RPV (1-3) actions: ◦ Performs level control actions: ○ Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1 ○ Determines that RPV water level is known ○ Determines that RPV water level can be restored and maintained between 159.3 and 202.3 inches using Feedwater, HPCS, RCIC, CRD, and/or SLC

Instructor Actions / Plant Response	Operator Actions
	○ Directs an RPV level band of 160 – 200 inches ○ May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: ○ Determines that no SRVs are cycling ○ Determines that RPV pressure can be stabilized below 1052 psig using the main turbine bypass valves ○ Directs a pressure band of 800-1000 psig
	ATC • Acknowledges SRO direction to scram the reactor • **Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603 (CT-1.0)** • **Places the Reactor Mode Switch in the SHUTDOWN position (CT-1.0)** • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs open ◦ Feedwater pumps are running ◦ All control rods fully inserted • Performs N2-SOP-101C actions: ◦ Verifies: ○ All rods full in ○ Reactor power lowering ○ SDV V & D Valves closed ○ RPV pressure on TBVs or SRVs ○ FWLC controlling level > 159.3 ◦ Fully inserts IRMs and SRMs ◦ Energizes 2WCS-MOV107 ◦ If required, secures makeup to the cooling tower

Instructor Actions / Plant Response	Operator Actions
	◦ Performs Level control leg: ○ Recognizes that level will be stabilized using Feedwater ○ If directed defeats HPCS Level 8 trips by directing a field operator to perform the actions of N2-EOP-6.20
Note: N2-OP-33 section H.3.0 would provide the highest flow rate makeup to the Suppression Pool, however it will not work due to the trip of the HPCS pump. The crew may still attempt to utilize N2-OP-33 section H.2.0 and/or N2-OP-31 to provide some Suppression Pool makeup. **Role Play:** When contacted as field operator to open Condensate Flush Supply valves, wait 3 minutes, then make the report that the valve is stuck, unable to open valve	**BOP** • May attempt Suppression Pool makeup using N2-OP-31 section H.6.0: • Verifies LPCI operability requirements • Performs the following: ◦ Depress RHR A (B, C) MANUALLY OUT OF SVCE pushbutton ◦ Verifies amber light is lit • Verifies annunciator 601431 (601, 631) RHR A (B,C) SYSTEM INOPERABLE is in alarm • Places 2RHS*P1A (B, C) control switch in PULL-TO-LOCK • For the applicable RHR system, unlock AND open the following valve: ◦ For RHR A, 2RHS*V71 ◦ For RHR B, 2RHS*V22 ◦ For RHR C, 2RHS*V21 • May attempt Suppression Pool makeup using N2-OP-33 section H.2.0: ◦ Verify SM has declared CSH System inoperable ◦ Verifies open 2CSH*MOV101, PUMP SUCT FROM CNDS TK ◦ Monitors AND maintains 2CSH*PI117, HPCS SYSTEM PRESS PMP, to greater than or equal to 65 psig ◦ Throttles open 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL to begin transferring water from CST to Suppression Pool ◦ Monitors CST AND Suppression Pool Levels

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If crew establishes Suppression Pool makeup with HPCS, verify the HPCS pump trips on the following **malfunction**: **TRG30, CS:HP:MT:I, HPCS Motor Failure, Final = short, Delay = 30 sec** **Role Play:** If directed to defeat the RCIC level 8 logic, wait 1 minute then **manually insert** the following **remote**: **RC02A**, RCIC Level 8 Trip Defeat:Withdraw Trip Units N693A,E, Final = Defeated **Role Play** If contacted as field operator to defeat the Offgas high radiation isolation per N2-OP-42 Attachment 6, wait 1 minute and **insert** the following **remote**: **OG03**, Off Gas High Radiation Isolation Defeated, Final = Defeated Then report that the Offgas high radiation isolation has been defeated. **Booth Operator:** If directed to defeat the RHR group 5 isolation, **manually insert** the following **malfunction**: • **RH08,** GROUP 5 ISOLATION FAILURE – (RHS*MOV122/113), Final = true Wait 1 minute, then report that the RHR group 5 isolation has been defeated.	• May attempt Suppression Pool makeup using N2-OP-33 section H.3.0: ◦ Verifies subsection F.1.0, Standby Condition Status Checks, is complete ◦ Verifies SM has declared CSH system inoperable ◦ Starts 2CSH*P1, HPCS PUMP 1 ◦ Verifies open2 CSH*MOV105, MINIMUM FLOW BYPASS VLV ◦ Monitors CST and Suppression Pool Levels ▪ Recognizes / reports trip of HPCS pump • Maintains RPV pressure 800 - 1000 psig using EHC and/or SRVs • Recognizes / reports trip of EHC pump ○ May attempt to start standby EHC pump ○ Recognize / report trip of 2nd EHC pump

Instructor Actions / Plant Response	Operator Actions
	SRO • Determines cannot maintain Suppression Pool water level above 192' • Enters N2-EOP-C2 (1-3) • Answers "Will the reactor stay shutdown without boron?" Yes • Determines Drywell pressure is <1.68 psig • **If Suppression Pool water level is above 192', directs open all 7 ADS valves (CT-2.0)** • **If Suppression Pool water level is at or below 192', then rapidly depressurizes the RPV using one or more RPV Blowdown Systems (CT-2.0)**
	ATC/BOP • Recognize / report Suppression Pool water level cannot be maintained above 192' • **Performs RPV Blowdown as directed by SRO (CT-2.0):** ◦ May open all 7 ADS valves: ◦ If an ECCS pump is operating performs the following to open all 7 ADS valves using N2-EOP-HC attachment 5: ◦ Determines that no SRVs are stuck open and that an ECCS pump is operating ◦ Arms and depresses both ADS logic pushbuttons for Division I: - ADS LOGIC "A" manual pushbutton - ADS LOGIC "E" manual pushbutton ◦ Arms and depresses both ADS logic pushbuttons for Division II: - ADS LOGIC "B" manual pushbutton - ADS LOGIC "F" manual pushbutton

Instructor Actions / Plant Response	Operator Actions
	◦ If an ECCS pump is not operating performs the following to open all 7 ADS valves using N2-EOP-HC attachment 5: ◦ Takes control switches to OPEN at EITHER of the following panels UNTIL a total of 7 SRVs are open: - 2CEC*PNL628 ○ 2MSS*PSV137 ○ 2MSS*PSV127 ○ 2MSS*PSV126 ○ 2MSS*PSV121 ○ 2MSS*PSV134 ○ 2MSS*PSV130 ○ 2MSS*PSV129 - 2CEC*PNL631 ○ 2MSS*PSV137 ○ 2MSS*PSV127 ○ 2MSS*PSV126 ○ 2MSS*PSV121 ○ 2MSS*PSV134 ○ 2MSS*PSV130 ○ 2MSS*PSV129 ◦ May rapidly depressurize the RPV using one or more RPV Blowdown Systems

Event Termination Criteria	• All control rods in. • RPV Blowdown in progress or completed. • Reactor level being controlled in assigned band.

Form 4.1-BWR Boiling-Water Reactor Examination Outline

Facility: Nine Mile Point 2	**K/A Catalog Rev. 3**	**Rev.** 0	**Date of Exam:** 11/06/2023

Tier	Group	RO K/A Category Points												SRO-Only Points			
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2		G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	4	3				3	4			3	20	4		3	7
	2	1	1	1				1	1			1	6	2		1	3
	Tier Totals	4	5	4				4	5			4	26	6		4	10
2. Plant Systems	1	2	3	3	2	2	2	3	2	3	2	2	26	2		3	5
	2	1	1	1	1	1	1	1	1	0	2	1	11	0	1	2	3
	Tier Totals	3	4	4	3	3	3	4	3	3	4	3	37	3		5	8

3. Generic Knowledge and Abilities Categories	CO	EC	RC	EM	
	2	2	1	1	6

SRO-Only Points				
CO	EC	RC	EM	
2	2	1	2	7

4. Theory	Reactor Theory	Thermodynamics	
	3	3	6

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EM = Emergency Procedures/Plan

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan.

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan.

ES-4.1-BWR	BWR Examination Outline (Nine Mile Point 2)									
	Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)									
Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
1	(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION			X				AK3.02 Knowledge of the reasons for the following responses or actions as they apply to (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reactor power response	4.0	1
2	(295003) (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER	X						AK1.07 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER: Load sequencing	3.8	2
3	(295004) (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER						X	2.4.31 Knowledge of annunciator alarms, indications, or response procedures (CFR: 41.10 / 45.3)	4.2	3
4	(295005) (APE 5) MAIN TURBINE GENERATOR TRIP				X			AA1.01 Ability to operate or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system	3.3	4
5	(295006) (APE 6) SCRAM					X		AA2.06 Ability to determine or interpret the following as they apply to SCRAM: Cause of reactor SCRAM	3.6	5
6	(295016) (APE 16) CONTROL ROOM ABANDONMENT		X					AK2.09 Knowledge of the relationship between the CONTROL ROOM ABANDONMENT and the following systems or components: RHR/LPCI: torus/suppression pool cooling mode	3.7	6
7	(295018) (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)					X		AA2.03 Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Partial or complete loss	3.7	7
8	(295019) (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR					X		AA2.01 Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air pressure	4.1	8
9	(295021) (APE 21) LOSS OF SHUTDOWN COOLING			X				AK3.01 Knowledge of the reasons for the following responses or actions as they apply to LOSS OF SHUTDOWN COOLING: Raising reactor water level	4.0	9
10	(295023) (APE 23) REFUELING ACCIDENTS						X	2.2.35 Ability to determine technical specification mode of operation	3.6	10
11	(295024) (EPE 1) HIGH DRYWELL PRESSURE	X						EK1.01 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the HIGH DRYWELL PRESSURE: Drywell integrity	4.3	11
12	(295025) (EPE 2) HIGH REACTOR PRESSURE				X			EA1.09 Ability to operate or monitor the following as they apply to HIGH REACTOR PRESSURE: RPS	4.2	12
13	(295026) (EPE 3) SUPPRESSION POOL HIGH WATER TEMPERATURE						X	2.1.29 Knowledge of how to conduct system lineups, such as valves, breakers, or switches	4.1	13
14	(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)			X				EK3.03 Knowledge of the reasons for the following responses or actions as they apply to HIGH DRYWELL TEMPERATURE: Drywell spray	3.8	14
15	(295030) (EPE 7) LOW SUPPRESSION POOL WATER LEVEL		X					EK2.05 Knowledge of the relationship between the LOW SUPPRESSION POOL WATER LEVEL and the following systems or components: HPCS	3.8	15

16	(295031) (EPE 8) REACTOR LOW WATER LEVEL		X					EK2.14 Knowledge of the relationship between the REACTOR LOW WATER LEVEL and the following systems or components: Emergency generators	4.1	16
17	(295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN					X		EA2.01 Ability to determine or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor power	4.3	17
18	(295038) (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE		X					EK2.03 Knowledge of the relationship between the HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components: Plant ventilation systems	3.8	18
19	(600000) (APE 24) PLANT FIRE ON SITE	X						AK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PLANT FIRE ON SITE: Firefighting methods for each type of fire	3.4	19
20	(700000) (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES				X			AA1.06 Ability to operate or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AC electrical distribution system	3.5	20
21	**(295004) (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER**					**X**		**AA2.01 Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Partial or complete loss of DC power**	**4.1**	**76**
22	**(295018) (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)**						**X**	**2.1.2 Knowledge of operator responsibilities during any mode of plant operation**	**4.4**	**77**
23	**(295019) (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR**						**X**	**2.1.27 Knowledge of system purpose and/or function**	**4.0**	**78**
24	**(295023) (APE 23) REFUELING ACCIDENTS**					**X**		**AA2.02 Ability to determine or interpret the following as they apply to REFUELING ACCIDENTS: Fuel pool level**	**3.8**	**79**
25	**(295024) (EPE 1) HIGH DRYWELL PRESSURE**					**X**		**EA2.04 Ability to determine or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression chamber pressure**	**4.3**	**80**
26	**(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)**						**X**	**2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm**	**4.3**	**81**
27	**(700000) (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES**					**X**		**AA2.10 Ability to determine or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions**	**3.5**	**82**
K/A Category Totals:		3	4	3	3	4/4	3/3	Group Point Total:		20/7

ES-4.1-BWR BWR Examination Outline (Nine Mile Point 2)

Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
28	(295002) (APE 2) LOSS OF MAIN CONDENSER VACUUM				X			AA1.07 Ability to operate or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Circulating water system	3.4	21
29	(295009) (APE 9) LOW REACTOR WATER LEVEL			X				AK3.01 Knowledge of the reasons for the following responses or actions as they apply to LOW REACTOR WATER LEVEL: Recirculation runback	3.7	22
30	(295010) (APE 10) HIGH DRYWELL PRESSURE						X	2.1.32 Ability to explain and apply system precautions, limitations, notes, or cautions	3.8	23
31	(295013) (APE 13) HIGH SUPPRESSION POOL TEMPERATURE.					X		AA2.02 Ability to determine or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE.: Localized heating/stratification	3.5	24
32	(295029) (EPE 6) HIGH SUPPRESSION POOL WATER LEVEL	X						EK1.01 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the HIGH SUPPRESSION POOL WATER LEVEL: Containment integrity	3.8	25
33	(295033) (EPE 10) HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS		X					EK2.07 Knowledge of the relationship between the HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following systems or components: Systems required for adequate core cooling	4.2	26
34	**(295007) (APE 7) HIGH REACTOR PRESSURE**						**X**	**2.4.14 Knowledge of general guidelines for emergency and abnormal operating procedures usage**	**4.5**	**83**
35	**(295035) (EPE 12) SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE**					**X**		**EA2.03 Ability to determine or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Lights and alarms**	**3.3**	**84**
36	**(500000) (EPE 16) HIGH CONTAINMENT HYDROGEN CONCENTRATION**					**X**		**EA2.02 Ability to determine or interpret the following as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATION: Oxygen monitoring availability**	**3.8**	**85**
	(295008) (APE 8) HIGH REACTOR WATER LEVEL / 2									
	(295012) (APE 12) HIGH DRYWELL TEMPERATURE / 5									
	(295014) (APE 14) INADVERTENT REACTIVITY ADDITION / 1									
	(295017) (APE 17) ABNORMAL OFFSITE RELEASE RATE / 9									
	(295020) (APE 20) INADVERTENT CONTAINMENT ISOLATION / 5 & 7									
	(295022) (APE 22) LOSS OF CONTROL ROD DRIVE PUMPS / 1									

	(295032) (EPE 9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE / 5									
	(295034) (EPE 11) SECONDARY CONTAINMENT VENTILATION HIGH RADIATION / 9									
	(295036) (EPE 13) SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL / 5									
K/A Category Totals:		1	1	1	1	1/2	1/1	Group Point Total:		6/3

ES-4.1-BWR	BWR Examination Outline (Nine Mile Point 2) Plant Systems—Tier 2/Group 1 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
37	(203000) (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE				X								K4.03 Knowledge of RHR/LPCI: INJECTION MODE design features and/or interlocks that provide for the following: Pump minimum flow protection	3.2	27
38	(205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)					X							K5.03 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Decay heat removal	3.9	28
39	(209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM		X										K2.02 LOW PRESSURE CORE SPRAY SYSTEM Knowledge of electrical power supplies to the following: Valves	3.8	29
40	(209002) (SF2, SF4 HPCS) HIGH PRESSURE CORE SPRAY SYSTEM									X			A3.01 Ability to monitor automatic operation of the HIGH PRESSURE CORE SPRAY SYSTEM including: Valve operation	4.2	30
41	(300000) (SF8 IA) INSTRUMENT AIR SYSTEM		X										K2.01 Knowledge of electrical power supplies to the following: Instrument air compressor	3.3	31
42	(211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM			X									K3.01 Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the following systems or system parameters: Reactor power	4.0	32
43	(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM											X	REACTOR PROTECTION SYSTEM (291008K1.06) BREAKERS, RELAYS, AND DISCONNECTS: Interpreting one-line diagram of control circuitry	3.6	33
44	(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM	X											K1.05 Knowledge of the physical connections and/or cause and effect relationships between the REACTOR PROTECTION SYSTEM and the following systems: Process radiation monitoring system	3.2	34
45	(215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM									X			A3.04 Ability to monitor automatic operation of the INTERMEDIATE RANGE MONITOR SYSTEM including: Control rod block signals	3.9	35

46	(215004) (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM										X		A4.08 Ability to manually operate and/or monitor the SOURCE RANGE MONITOR SYSTEM in the control room: SRMS channel bypass	3.4	36
47	(215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR						X						K6.08 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR: Uninterruptable power supply (AC/DC)	3.5	37
48	(217000) (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM			X									K3.06 Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM will have on the following systems or system parameters: Condensate storage tank level	3.2	38
49	(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM		X										K2.01 AUTOMATIC DEPRESSURIZATION SYSTEM Knowledge of electrical power supplies to the following: ADS logic	4.0	39
50	(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF			X									K3.14 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF will have on the following systems or system parameters: Recirculation system	3.1	40
51	(239002) (SF3 SRV) SAFETY RELIEF VALVES									X			A3.02 Ability to monitor automatic operation of the SAFETY RELIEF VALVES including: SRV operation on high reactor pressure	4.2	41
52	(259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM	X											K1.04 Knowledge of the physical connections and/or cause and effect relationships between the REACTOR WATER LEVEL CONTROL SYSTEM and the following systems: Reactor feedwater flow	3.9	42
53	(259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM						X						K6.05 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor water level input	3.8	43
54	(261000) (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM							X					A1.07 Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY GAS TREATMENT SYSTEM including: SGTS train temperature	3.0	44

55	(261000) (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM					X							K5.04 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the STANDBY GAS TREATMENT SYSTEM: Secondary containment pressure changes	3.4	45
56	(262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION							X					A1.03 Ability to predict and/or monitor changes in parameters associated with operation of the AC ELECTRICAL DISTRIBUTION including: Bus voltage	3.6	46
57	(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)				X								K4.01 Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks that provide for the following: Transfer of power supplies	3.5	47
58	(263000) (SF6 DC) DC ELECTRICAL DISTRIBUTION											X	2.2.39 Knowledge of less than or equal to 1 hour technical specification action statements (This K/A does not include action statements of 1 hour or less that follow the expiration of a completion time for a technical specification condition for which an action statement has already been entered.)	3.9	48
59	(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET)										X		A4.07 Ability to manually operate and/or monitor the EMERGENCY GENERATORS (DIESEL/JET) in the control room: Transfer of control to local/remote panels	3.4	49
60	(400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM							X					A1.01 Ability to predict and/or monitor changes in parameters associated with operation of the COMPONENT COOLING WATER SYSTEM including: CCW flow rate	3.0	50
61	(400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM								X				A2.03 Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High/low CCW temperature	3.8	51
62	(510000) (SF4 SWS*) SERVICE WATER SYSTEM								X				A2.06 Ability to (a) predict the impacts of the following on the SERVICE WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Discharge strainer failure	3.3	52

63	(209002) (SF2, SF4 HPCS) HIGH PRESSURE CORE SPRAY SYSTEM											X	2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator	4.1	86
64	(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF								X				A2.03 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System logic failures	4.0	87
65	(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)											X	2.2.45 - Ability to determine or interpret technical specifications with action statements of greater than 1 hour	4.7	88
66	(263000) (SF6 DC) DC ELECTRICAL DISTRIBUTION								X				A2.03 Ability to (a) predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal battery parameters	3.2	89
67	(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET)											X	2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator	3.8	90
	(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM														
K/A Category Totals:		2	3	3	2	2	2	3	2/2	3	2	2/3	Group Point Total:		26/5

ES-4.1-BWR	BWR Examination Outline (Nine Mile Point 2) Plant Systems—Tier 2/Group 2 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
68	(202001) (SF1, SF4 RS) RECIRCULATION SYSTEM							X					A1.10 Ability to predict and/or monitor changes in parameters associated with operation of the RECIRCULATION SYSTEM including: Recirculation seal purge flows	2.9	53
69	(202002) (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM										X		A4.10 Ability to manually operate and/or monitor the RECIRCULATION FLOW CONTROL SYSTEM in the control room: Flow control valve	4.0	54
70	(215002) (SF7 RBMS) ROD BLOCK MONITOR SYSTEM			X									K3.01 Knowledge of the effect that a loss or malfunction of the ROD BLOCK MONITOR SYSTEM will have on the following systems or system parameters: Reactor manual control system	3.9	55
71	(223001) (SF5 PCS) PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES								X				A2.03 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Safety/relief valve leaking or stuck-open	4.3	56
72	(230000) (SF5 RHR SPS) RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE						X						K6.10 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE: Component cooling water systems	2.9	57
73	(239001) (SF3, SF4 MRSS) MAIN AND REHEAT STEAM SYSTEM		X										K2.02 MAIN AND REHEAT STEAM SYSTEM Knowledge of electrical power supplies to the following: Main steam line shutoff valves	3.3	58
74	(241000) (SF3 RTPRS) REACTOR/TURBINE PRESSURE REGULATING SYSTEM	X											K1.11 Knowledge of the physical connections and/or cause and effect relationships between the REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following systems: RPS	4.1	59

75	(245000) (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS					X							K5.08 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Generator cooling	3.1	60
76	(256000) (SF2 CDS) CONDENSATE SYSTEM											X	2.1.25 Ability to interpret reference materials, such as graphs, curves, and tables (reference potential)	3.9	61
77	(272000) (SF7, SF9 RMS) RADIATION MONITORING SYSTEM				X								K4.02 Knowledge of RADIATION MONITORING SYSTEM design features and/or interlocks that provide for the following: System isolations/initiations	4.1	62
78	(290001) (SF5 SC) SECONDARY CONTAINMENT										X		A4.13 Ability to manually operate and/or monitor the SECONDARY CONTAINMENT in the control room: Secondary containment ventilation systems	3.6	63
79	**(201002) (SF1 RMCS) REACTOR MANUAL CONTROL SYSTEM**											**X**	**2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation**	**4.2**	**91**
80	**(214000) (SF7 RPIS) ROD POSITION INFORMATION SYSTEM**								**X**				**A2.04 Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Power supply loss**	**3.3**	**92**
81	**(290002) (SF4 RVI) REACTOR VESSEL INTERNALS**											**X**	**2.1.23 Ability to perform general or normal operating procedures during any plant condition**	**4.4**	**93**
	(201001) (SF1 CRDH) CRD HYDRAULIC SYSTEM														
	(201003) (SF1 CRDM) CONTROL ROD AND DRIVE MECHANISM														
	(201004) (SF7 RSCS) ROD SEQUENCE CONTROL SYSTEM														
	(201005) (SF1, SF7 RCIS) ROD CONTROL AND INFORMATION SYSTEM														
	(201006) (SF7 RWMS) ROD WORTH MINIMIZER SYSTEM														
	(204000) (SF2 RWCU) REACTOR WATER CLEANUP SYSTEM														
	(215001) (SF7 TIP) TRAVERSING IN CORE PROBE														
	(216000) (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION														

	(219000) (SF5 RHR SPC) RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE														
	(226001) (SF5 RHR CSS) RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE														
	(233000) (SF9 FPCCU) FUEL POOL COOLING/CLEANUP														
	(234000) (SF8 FH) FUEL HANDLING														
	(239003) (SF9 MSVLCS) MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM														
	(259001) (SF2 FWS) FEEDWATER SYSTEM														
	(268000) (SF9 RW) RADWASTE SYSTEM														
	(271000) (SF9 OG) OFFGAS SYSTEM														
	(286000) (SF8 FPS) FIRE PROTECTION SYSTEM														
	(288000) (SF9 PVS) PLANT VENTILATION SYSTEMS														
	(290003) (SF9 CRV) CONTROL ROOM VENTILATION														
	(510001) (SF8 CWS*) CIRCULATING WATER SYSTEM														
K/A Category Totals:		1	1	1	1	1	1	1	1/1	0	2	1/2	Group Point Total:		11/3

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Nine Mile Point 2)					
Facility: Nine Mile Point 2					Date of Exam: 11/06/2023	

Generic Knowledge and Abilities Outline (Tier 3) (RO/SRO)

Category	K/A #	Topic	Item #	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations	2.1.14	Knowledge of criteria or conditions that require plantwide announcements, such as pump starts, reactor trips, and mode changes	82	3.1	64		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management	83	4.3	65		
	2.1.36	**Knowledge of procedures and limitations involved in core alterations**	**84**			**4.1**	**94**
	2.1.38	**Knowledge of the station's requirements for verbal communications when implementing procedures**	**85**			**3.8**	**95**
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	2.2.6	Knowledge of the process for making changes to procedures	86	3.0	66		
	2.2.43	Knowledge of the process used to track inoperable alarms	87	3.0	67		
	2.2.37	**Ability to determine operability or availability of safety-related equipment (SRO Only)**	**88**			**4.6**	**96**
	2.2.38	**Knowledge of conditions and limitations in the facility license**	**89**			**4.5**	**97**
	Subtotal			N/A	2	N/A	**2**
3. Radiation Control	2.3.12	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters	90	3.2	68		
	2.3.14	**Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)**	**91**			**3.8**	**98**
	Subtotal			N/A	1	N/A	**1**
4. Emergency Procedures / Plan	2.4.6	Knowledge of emergency and abnormal operating procedures major action categories	92	3.7	69		
	2.4.23	**Knowledge of the bases for prioritizing emergency operating procedures implementation**	**93**			**4.4**	**99**
	2.4.29	**Knowledge of the emergency plan implementing procedures**	**94**			**4.4**	**100**
	Subtotal			N/A	1	N/A	**2**
Tier 3 Point Total				N/A	6	N/A	**7**

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Nine Mile Point 2)				
Facility: Nine Mile Point 2				Date of Exam:	11/06/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	292008	K1.02 Reactor Operational Physics: List reactivity control mechanisms that exist for plant conditions during the approach to criticality	95	3.8	70
	292003	K1.05 Define reactor period	96	3.7	71
	292007	K1.03 FUEL DEPLETION AND BURNABLE POISONS: Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points	97	2.7	72
	Subtotal				3
Thermodynamics	293005	K1.05 THERMODYNAMIC CYCLES: State the advantages of moisture separators/reheaters and feedwater heaters for a typical steam cycle	98	2.8	73
	293008	K1.11 THERMAL HYDRAULICS (POOL BOILING CURVE (TEMPERATURE VS. HEAT FLUX)): Describe transition (partial film) boiling	99	2.8	74
	293010	K1.03 BRITTLE FRACTURE AND VESSEL THERMAL STRESS: Define reference temperature	100	2.5	75
	Subtotal				3
Tier 4 Point Total				N/A	6

Form 4.1-1 Record of Rejected Knowledge and Abilities

Refer to Examination Standard (ES)-4.2, “Developing Written Examinations,” Section B.3, for deviations from the approved written examination outline.

Tier/Group	Randomly Selected K/A	Reason for Rejection
1 / 1	Question 6 295016 CONTROL ROOM ABANDONMENT AK2.05 - Knowledge of the relationship between the CONTROL ROOM ABANDONMENT and the following systems or components: DC electrical distribution	An acceptable question could not be developed for the randomly sampled K/A due to lack of relationship between Control Room abandonment and DC electrical distribution. Randomly reselected K/A 295016 CONTROL ROOM ABANDONMENT AK2.09 - Knowledge of the relationship between the CONTROL ROOM ABANDONMENT and the following systems or components: RHR/LPCI: torus/suppression pool cooling mode.
2 / 1	Question 31 211000 STANDBY LIQUID CONTROL SYSTEM K2.03 - Knowledge of electrical power supplies to the following: Heater power	An acceptable question could not be developed for the randomly sampled K/A without testing minutia. Additionally, T2G1 system 300000 was inadvertently not sample during the original sampling process. Systematically and randomly reselected K/A 300000 Instrument Air System K2.01 - Knowledge of electrical power supplies to the following: Instrument air compressor.
2 / 1	Question 33 212000 REACTOR PROTECTION SYSTEM 291008 BREAKERS, RELAYS, AND DISCONNECTS K1.10 - Function, control, and precautions associated with disconnects	An acceptable question could not be developed for the randomly sampled K/A due to lack of relevant disconnects in the Reactor Protection system. Randomly reselected K/A 212000 REACTOR PROTECTION SYSTEM 291008 BREAKERS, RELAYS, AND DISCONNECTS K1.06 - Interpreting one-line diagram of control circuitry.
2 / 1	Question 43 259002 REACTOR WATER LEVEL CONTROL SYSTEM K6.06 - Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor pressure/ temperature input (for water level input compensation)	An acceptable question could not be developed for the randomly sampled K/A due to lack of pressure/temperature compensation of Reactor water level instruments used for Reactor water level control. Randomly reselected K/A 259002 REACTOR WATER LEVEL CONTROL SYSTEM K6.05 - Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor water level input.

2 / 1	Question 45 261000 STANDBY GAS TREATMENT SYSTEM K5.05 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the STANDBY GAS TREATMENT SYSTEM: Fuel handling building pressure changes	An acceptable question could not be developed for the randomly sampled K/A because the facility does not have a fuel handling building. Randomly reselected K/A 261000 STANDBY GAS TREATMENT SYSTEM K5.04 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the STANDBY GAS TREATMENT SYSTEM: Secondary containment pressure changes.
2 / 2	Question 62 272000 RADIATION MONITORING SYSTEM K4.04 - Knowledge of RADIATION MONITORING SYSTEM design features and/or interlocks that provide for the following: Process radiation monitoring surveillance testing	An acceptable question could not be developed for the randomly sampled K/A because surveillance testing of process radiation monitoring is largely not the responsibility of Operations at the facility (operational validity). Randomly reselected K/A 272000 RADIATION MONITORING SYSTEM K4.02 - Knowledge of RADIATION MONITORING SYSTEM design features and/or interlocks that provide for the following: System isolations/initiations.
2 / 2	Question 63 290001 SECONDARY CONTAINMENT A4.12 - Ability to manually operate and/or monitor the SECONDARY CONTAINMENT in the control room: Surveillance testing	An acceptable question could not be developed for the randomly sampled K/A without testing minutia. Randomly reselected K/A 290001 SECONDARY CONTAINMENT A4.13 - Ability to manually operate and/or monitor the SECONDARY CONTAINMENT in the control room: Secondary containment ventilation systems.
4	Question 70 292001 NEUTRONS K1.02 - Define prompt and delayed neutrons	An acceptable question could not be developed for the randomly sampled K/A without being too generic in nature. Randomly reselected K/A 292008 Reactor Operational Physics K1.02 - List reactivity control mechanisms that exist for plant conditions during the approach to criticality.

1 / 1	Question 76 295004 PARTIAL OR COMPLETE LOSS OF DC POWER AA2.04 - Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: System lineups	An acceptable question could not be developed for the randomly sampled K/A without overlapping concepts tested in Question 89. Randomly reselected K/A 295004 PARTIAL OR COMPLETE LOSS OF DC POWER AA2.01 - Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Partial or complete loss of DC power.
1 / 1	Question 81 295028 HIGH DRYWELL TEMPERATURE 2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms or personnel monitoring equipment	An acceptable question could not be developed for the randomly sampled K/A due to lack of relationship between the evolution and the generic topic. Randomly reselected K/A 295028 HIGH DRYWELL TEMPERATURE 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.
1 / 1	Question 82 700000 GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES AA2.06 - Ability to determine or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator frequency limitations	An acceptable question could not be developed for the randomly sampled K/A due to lack of procedural limitations on Generator frequency. Randomly reselected K/A 700000 GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES AA2.10 - Ability to determine or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions.
2 / 1	Question 88 262002 UNINTERRUPTABLE POWER SUPPLY (AC/DC) 2.4.16 - Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines	An acceptable question could not be developed for the randomly sampled K/A without testing similar concepts as in Question 83. Randomly reselected K/A 262002 UNINTERRUPTABLE POWER SUPPLY (AC/DC) 2.2.45 - Ability to determine or interpret technical specifications with action statements of greater than 1 hour.

2 / 1	Question 90 264000 EMERGENCY GENERATORS (DIESEL/JET) 2.4.49 - Ability to perform without reference to procedures those actions that require immediate operation of system components and controls	An acceptable question could not be developed for the randomly sampled K/A at the SRO license level. Randomly reselected K/A 264000 EMERGENCY GENERATORS (DIESEL/JET) 2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.
2 / 2	Question 61 256000 CONDENSATE SYSTEM 2.1.8 - Ability to coordinate personnel activities outside the control room	An acceptable question could not be developed for the randomly sampled K/A without testing minutia. Randomly reselected K/A 256000 CONDENSATE SYSTEM 2.1.25 - Ability to interpret reference materials, such as graphs, curves, and tables (reference potential).

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK3.02
	Importance Rating	4.0

PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION

Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reactor power response

Proposed Question: #1

The plant is operating at 100% power.

Which one of the following identifies a transient in which core oscillations would be a significant concern and the required action if core oscillations are observed, in accordance with plant procedures?

	Transient	Required Action If Core Oscillations Are Observed
A.	Recirc pumps shift to slow speed	Insert CRAM rods and raise Recirc flow
B.	Recirc pumps shift to slow speed	Scram the Reactor
C.	Isolation of a Feedwater heater string	Isolate the other Feedwater heater strings
D.	Isolation of a Feedwater heater string	Scram the Reactor

Proposed Answer: B

Explanation: Core oscillations due to thermal hydraulic instability are a significant concern following a sudden reduction in core flow, such as would occur with Recirc pumps shifting to slow speed. An override in N2-SOP-29, Sudden Reduction in Core Flow, requires inserting a manual Reactor scram if core oscillations are observed.

A. Incorrect – An override in N2-SOP-29, Sudden Reduction in Core Flow, requires inserting a manual Reactor scram if core oscillations are observed. Plausible because N2-SOP-29 does require inserting CRAM rods in other situations and raising core flow would lower the risk of core oscillations.
C. Incorrect – Core oscillations due to thermal hydraulic instability are a significant concern following a sudden reduction in core flow, not a loss of Feedwater heating. Plausible because loss of a single string of Feedwater heating does create an imbalance in temperatures exiting the Feedwater heating strings and does create a Reactor power change. An override in N2-SOP-29, Sudden Reduction in Core Flow, requires inserting a manual Reactor scram if core oscillations are observed. Plausible because isolating the other Feedwater heater strings would get rid of the imbalance in temperatures between the Feedwater heater strings.
D. Incorrect – Core oscillations due to thermal hydraulic instability are a significant concern following a sudden reduction in core flow, not a loss of Feedwater heating. Plausible because loss of a single string of Feedwater heating does create an imbalance in temperatures exiting the Feedwater heating strings and does create a Reactor power change.

Technical Reference(s): N2-SOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP29C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AK1.07
	Importance Rating	3.8

PARTIAL OR COMPLETE LOSS OF AC POWER

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PARTIAL OR COMPLETE LOSS OF AC POWER: Load sequencing

Proposed Question: #2

The plant is operating at 100% power with the following:

- A loss of ALL offsite power has just occurred.
- Concurrently, a steam leak in the Drywell has resulted in a Drywell pressure of 2.5 psig and rising rapidly.
- All Emergency Diesel Generators (EDGs) automatically start.
- All EDG output breakers close simultaneously.

Which one of the following describes the approximate time for the sequencing of the low pressure ECCS pumps onto the 4.16KV emergency switchgears after the EDG output breakers are closed?

	RHR A & B	CSL & RHR C
A.	1 second	6 seconds
B.	1 second	10 seconds
C.	5 seconds	6 seconds
D.	5 seconds	10 seconds

Proposed Answer: A

Explanation: Due to the presence of combined LOOP and LOCA (Drywell pressure > 1.68 psig) signals, RHS*P1A and RHS*P1B start 1 second after the EDG output breakers are closed. CSL*P1 and RHS*P1C start 5 seconds later (6 seconds from closure of the EDG output breaker).

B. Incorrect – CSL*P1 and RHS*P1C start 6 seconds after EDG output breaker closure. Plausible because 10 seconds is the normal start time with no LOOP.
C. Incorrect – RHS*P1A and RHS*P1B start 1 second after EDG output breaker closure. Plausible because 5 seconds is the normal start time with no LOOP.
D. Incorrect – RHS*P1A and RHS*P1B start 1 second after EDG output breaker closure. Plausible because 5 seconds is the normal start time with no LOOP. CSL*P1 and RHS*P1C start 6 seconds after EDG output breaker closure. Plausible because 10 seconds is the normal start time with no LOOP.

Technical Reference(s): UFSAR Tables 8.3-1 and 8.3-2, ESK-5ENS21, Sheet 2, ESK-5ENS22 (Sheet 2)

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-5

Question Source: Bank - 2019 NRC #28

Question History: 2019 NRC #28

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 2.4.31
	Importance Rating	4.2

PARTIAL OR COMPLETE LOSS OF DC POWER

Knowledge of annunciator alarms, indications, or response procedures

Proposed Question: #3

The plant is operating at 100% power with the following:

- Annunciator 852108, DIV I EMER BUS BYS 002A 125VDC SYSTEM TROUBLE, is in alarm.
- The following page shows related indications that are present at the rear of 2CEC*PNL852.

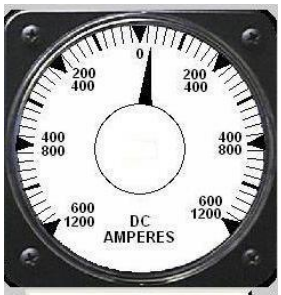
Which one of the following describes these indications?

A. A loss of battery has occurred.

B. Battery cell reversal is occurring.

C. A loss of battery charger has occurred.

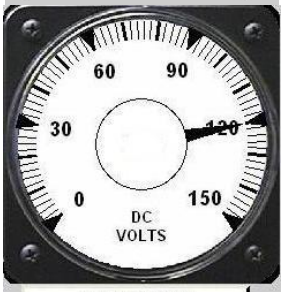
D. A loss of 2BYS*SWG002A has occurred.



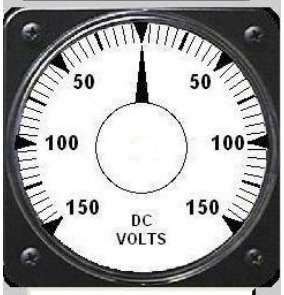
BATTERY 2A
DC AMPERES



BATTERY CHARGER 2A1
D.C. MA



BATTERY BUS
BYS 002A
D.C. VOLTS



BUS 2BYS*SWG002A
D.C. VOLTS





Proposed Answer: C

Explanation: The given indications show that battery charger output is zero amps and the battery is discharging. This indicates that a loss of the battery charger has occurred.

A. Incorrect – A loss of battery would be indicated by 0 amps on the Battery Amp meter. Plausible because the 4th meter indicates 0 VDC and this meter indication is commonly misunderstood.
B. Incorrect – Cell reversal would occur at <105 VDC per P&L 12.0. Plausible because DC volts are lower than normal.
D. Incorrect – The 3rd meter indicates that 2BYS*SWG002A has 120 VDC. Plausible because the 4th meter indicates 0 VDC and this meter indication is commonly misunderstood.

Technical Reference(s): ARP 852108

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: N2-263000-RBO-5

Question Source: Bank – 2019 NRC #48

Question History: 2019 NRC #48

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AA1.01
	Importance Rating	3.3

MAIN TURBINE GENERATOR TRIP

Ability to operate or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system

Proposed Question: #4

The plant is operating at 70% power with the following:

- A Main Turbine trip occurs.
- Reactor water level reaches a low of 168" on the transient.
- Reactor water level is now 180" and rising slowly.
- Reactor pressure reaches a high of 1030 psig on the transient.
- Reactor pressure is now 920 psig and stable.

Which one of the following describes the resulting status of the Reactor Recirculation system?

Reactor Recirculation pumps…

A. remain running in fast speed with flow control valve position unchanged

B. remain running in fast speed and flow control valves runback

C. are running in slow speed

D. are tripped

Proposed Answer: C

Explanation: Main Turbine Stop Valves rapidly close on the Main Turbine trip. This causes an End of Cycle Recirc Pump Trip (EOC-RPT) signal. EOC-RPT shifts the Recirc pumps from high speed to low speed.

A. Incorrect – Recirc pumps are operating in slow speed. Plausible because other trip/runback signals are not received because Reactor water level remains above 108.8" and 159.3" and Reactor pressure remains below 1065 psig.
B. Incorrect – Recirc pumps are operating in slow speed. Plausible because there is a Recirculation flow control valve runback that occurs in other conditions based on Condensate/Feedwater alignment.
D. Incorrect – Recirc pumps are operating in slow speed. Plausible because this would be correct if the Reactor level or pressure response was more severe.

Technical Reference(s): N2-OP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AA2.06
	Importance Rating	3.6

SCRAM

Ability to determine or interpret the following as they apply to SCRAM: Cause of reactor SCRAM

Proposed Question: #5

A plant startup is in progress with the following:

- The Reactor Mode Switch is in START / HOT STBY.
- All IRMs indicate 25 out of 125 on range 8.
- APRMs indicate 2%.

Then, the MSIVs close on a spurious isolation signal and the Reactor scrams, resulting in the following:

- All IRMs spike to 122 out of 125 on range 8.
- APRMs spike to 5%.

Given the following possible RPS signals:

(1) MSIV closure
(2) IRM upscale
(3) APRM upscale

Which one of the following identifies which of these provided an RPS scram signal during this transient?

A. (1) only

B. (2) only

C. (3) only

D. (1), (2), and (3)

Proposed Answer: B

Explanation: IRMs have exceeded the trip setpoint of 120 out of 125. APRMs have not exceeded 15%, which with the Reactor Mode Switch out of RUN would cause a Reactor scram. The MSIV closure scram is bypassed with the Reactor Mode Switch out of RUN.

A. Incorrect – The MSIV closure scram is bypassed with the Reactor Mode Switch out of RUN. Plausible because the MSIV closure happened first and would have resulted in a scram if the Mode Switch was in RUN.
C. Incorrect – APRMs have not exceeded 15%. Plausible because APRMs have risen by a large factor (2.5x) and they would cause a scram if they exceeded 15%.
D. Incorrect – The MSIV closure scram is bypassed with the Reactor Mode Switch out of RUN. Plausible because the MSIV closure happened first and would have resulted in a scram if the Mode Switch was in RUN. APRMs have not exceeded 15%. Plausible because APRMs have risen by a large factor (2.5x) and they would cause a scram if they exceeded 15%.

Technical Reference(s): ARPs 603102, 603108, 603127

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-5

Question Source: Bank – 2019 NRC #5

Question History: 2019 NRC #5

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK2.09
	Importance Rating	3.7

CONTROL ROOM ABANDONMENT

Knowledge of the relationship between the CONTROL ROOM ABANDONMENT and the following systems or components: RHR/LPCI: torus/suppression pool cooling mode

Proposed Question: #6

The plant is operating at 50% power with the following:

- A fire starts in the Main Control Room.
- N2-SOP-78, Control Room Evacuation, is entered.
- All required actions are completed prior to exiting the Main Control Room.
- All operators have just exited the Main Control Room.

Which one of the following identifies the current status of Suppression Pool Cooling, in accordance with N2-SOP-78?

Suppression Pool Cooling is currently…

A. in service and the lineup can be adjusted at the Remote Shutdown Panel.

B. in service, but the lineup CANNOT be adjusted at the Remote Shutdown Panel.

C. NOT in service, but can be placed in service at the Remote Shutdown Panel.

D. NOT in service and CANNOT be placed in service at the Remote Shutdown Panel.

Proposed Answer: C

Explanation: N2-SOP-78 has the Operators place RHR pumps A and B in PTL in the Control Room before exiting, but does not have the Operators place RHR in Suppression Pool Cooling at this time. N2-SOP-78 has the Operators place RHR in Suppression Pool Cooling later from the Remote Shutdown Panel.

A. Incorrect – Suppression Pool Cooling is not yet in service. Plausible because N2-SOP-78 has the Operators place RHR pumps A and B in PTL in the Control Room before exiting. Also plausible because N2-SOP-78 does eventually direct the Operators to place RHR in Suppression Pool Cooling, since heat is being added to the Suppression Pool from RCIC and SRVs.
B. Incorrect – Suppression Pool Cooling is not yet in service. Plausible because N2-SOP-78 has the Operators place RHR pumps A and B in PTL in the Control Room before exiting. Also plausible because N2-SOP-78 does eventually direct the Operators to place RHR in Suppression Pool Cooling, since heat is being added to the Suppression Pool from RCIC and SRVs.
D. Incorrect – RHR can be placed in Suppression Pool Cooling at the Remote Shutdown Panel. Plausible because the Remote Shutdown Panel only has limited controls and other modes of RHR operation are not possible at the Remote Shutdown Panel.

Technical Reference(s): N2-SOP-78, N2-OP-78

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP78C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.03
	Importance Rating	3.7

PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)

Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Partial or complete loss

Proposed Question: #7

A small-break loss of coolant accident has resulted in the following:

- Reactor water level is 170" and lowering slowly.
- Drywell pressure is 2.1 psig and rising slowly.
- Drywell temperature is 170°F and rising slowly.

Which one of the following describes the status of CCP flow to the Drywell Unit Coolers?

CCP flow to the Drywell Unit Coolers is…

A. NOT isolated and will only isolate if offsite power is also lost.

B. NOT isolated and will only isolate if Reactor water level lowers further.

C. isolated, but can be restored using LOCA OVERRIDE switches.

D. isolated and CANNOT be restored until Drywell pressure is lowered.

Proposed Answer: C

Explanation: CCP flow to the Drywell Unit Coolers has isolated because Drywell pressure is above 1.68 psig. N2-EOP-6.24 provides procedural guidance to restore CCP flow to the Drywell Unit Coolers using LOCA OVERRIDE switches.

A. Incorrect – CCP flow to the Drywell Unit Coolers has isolated because Drywell pressure is above 1.68 psig. Plausible that it would still be unisolated since cooling would assist in Drywell pressure/temperature control. Also plausible that offsite power status would affect the logic since then EDGs would be powering buses.
B. Incorrect – CCP flow to the Drywell Unit Coolers has isolated because Drywell pressure is above 1.68 psig. Plausible that it would still be unisolated since cooling would assist in Drywell pressure/temperature control. Also plausible that a Reactor water level signal would be required to complete the LOCA isolation logic.
D. Incorrect – N2-EOP-6.24 provides procedural guidance to restore CCP flow to the Drywell Unit Coolers using LOCA OVERRIDE switches. Plausible because this procedure does not allow restoring flow if Drywell temperature is ≥250^{o}F.

Technical Reference(s): N2-EOP-6 Attachment 24

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-5

Question Source: Bank - 2010 NRC #52

Question History: 2010 NRC #52

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 10/10/23 – Edited lead-in sentence to specify small break LOCA, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.01
	Importance Rating	4.1

PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air pressure

Proposed Question: #8

The plant is operating at 100% power with the following:

- Instrument Air compressor 2IAS-C3C is running.
- The Instrument Air Compressor Selector switch is in CAB.
- Instrument Air compressors 2IAS-C3A and 2IAS-C3B are NOT running but are available.
- Instrument Air pressure is 120 psig.

Then, the following occurs:

- A large leak develops in the Instrument Air system.
- Instrument Air pressure is 97 psig and lowering slowly.

Which one of the following describes the resulting status of Instrument Air compressors 2IAS-C3A and 2IAS-C3B?

	2IAS-C3A	2IAS-C3B
A.	Running	Running
B.	Running	NOT running
C.	NOT running	Running
D.	NOT running	NOT running

Proposed Answer: B

Explanation: With Instrument Air compressor 2IAS-C3C selected as the lead compressor, the Instrument Air compressor selector is selected to CAB. This makes 2IAS-C3A the lag compressor and 2IAS-C3B the backup compressor. The lag compressor starts when Instrument Air pressure lowers to 100 psig and the backup compressor starts when Instrument Air pressure lowers to 85 psig. With pressure currently at 97 psig, 2IAS-C3A is running and 2IAS-C3B is NOT running.

A. Incorrect – 2IAS-C3B is NOT running. Plausible because this would be correct if air pressure was below 85 psig.
C. Incorrect – 2IAS-C3A is running and 2IAS-C3B is NOT running. Plausible because this would be correct if the lag/backup arrangement was different.
D. Incorrect – 2IAS-C3A is running. Plausible because this would be correct if air pressure was above 100 psig.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AK3.01
	Importance Rating	4.0

LOSS OF SHUTDOWN COOLING

Knowledge of the reasons for the following responses or actions as they apply to LOSS OF SHUTDOWN COOLING: Raising reactor water level

Proposed Question: #9

The plant is shutdown with the following:

- RHR is operating in the Shutdown Cooling (SDC) lineup.
- NO Reactor Recirc pumps are running.
- Reactor water level is 183" and stable.

Then, the running RHR pump trips.

- N2-SOP-31, Loss of Shutdown Cooling, is entered.
- The procedure requires raising Reactor water level to at least 227".

Which one of the following describes the reason for raising Reactor water level, in accordance with N2-SOP-31 and N2-OP-31, Residual Heat Removal System?

A. Promote natural circulation

B. Ensure accurate Reactor water level indication

C. Commence feed and bleed of the Reactor vessel

D. Provide proper conditions for Reactor Recirc pump restart

Proposed Answer: A

Explanation: With no Reactor Recirc pump running and a loss of the running SDC pump, N2-SOP-31 requires raising level to at least 227". The reason for this is to raise level into the steam separators to promote natural circulation in the Reactor vessel.

B. Incorrect – The reason is to promote natural circulation. Plausible because raising level into the steam separators connects the water inside and outside of the shroud, which provides another means of communication to ensure level instruments measuring outside the shroud reflect level inside the shroud.
C. Incorrect – The reason is to promote natural circulation. Plausible because one possible method of decay heat removal in N2-SOP-31 is to perform a feed and bleed, and the first step in a feed and bleed would be to add water to the Reactor.
D. Incorrect – The reason is to promote natural circulation. Plausible because raising level would provide more NPSH for Reactor Recirc pumps.

Technical Reference(s): N2-SOP-31, N2-OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP31C01 EO-2

Question Source: Bank – 2019 NRC #9

Question History: 2019 NRC #9

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 2.2.35
	Importance Rating	3.6

REFUELING ACCIDENTS

Ability to determine technical specification mode of operation

Proposed Question: #10

Preparations are underway for a refueling outage with the following:

- All control rods are inserted.
- The Reactor Mode Switch is in SHUTDOWN.
- Reactor coolant temperature is 180ºF and lowering slowly.
- The Spent Fuel Pool gates are installed.
- All Reactor cavity shield plugs have been removed.
- The Reactor head remains fully tensioned.
- Fuel movements are being performed in the Spent Fuel Pool to prepare for core shuffle.

Then, an irradiated fuel bundle is dropped in the Spent Fuel Pool, resulting in the following:

- Annunciator 851254, PROCESS AIRBORNE RADN MON ACTIVATED, is in alarm.
- 2HVR*CAB14A(B), Reactor Building Above Refueling Floor Exhaust Rad Monitors, indicate 7.2×10^{-2} µCi/cc and rising slowly.
- 2HVR*CAB32A(B), Reactor Building Below Refueling Floor Exhaust Rad Monitors, indicate 5.1×10^{-7} µCi/cc and stable.
- Annunciator 851244, REACTOR BLDG AREA RADN MON ACTIVATED, is in alarm.
- Multiple Refuel Floor Area Radiation monitors are in alarm with dose rates rising.

Which one of the following describes the required Emergency Operating Procedure (EOP) entry, if any?

A. NO EOP entry is required

B. Enter N2-EOP-SC (1-3), Secondary Containment Control for Modes 1-3

C. Enter N2-EOP-CONT & RR (4), Containment and Radioactivity Release Control for Mode 4

D. Enter N2-EOP-CONT & RR (5), Containment and Radioactivity Release Control for Mode 5

Proposed Answer: C

Explanation: With the Reactor Mode Switch in SHUTDOWN, Reactor coolant temperature <200°F, and the Reactor head fully tensioned, the plant is in Mode 4. This makes N2-EOP-CONT & RR (4) the applicable EOP for high Reactor Building area radiation and high HVR exhaust radiation. The given conditions show that the EOP entry conditions for these parameters are exceeded (ARM above an alarm setpoint, Above Refuel Floor exhaust rad monitor >1.7 x 10^{-3} µCi/cc). Therefore, entry into N2-EOP-CONT & RR (4) is required.

A. Incorrect – Entry into N2-EOP-CONT & RR (4) is required. Plausible because this would be correct if the given radiation information remained below the associated EOP entry conditions.
B. Incorrect – Entry into N2-EOP-CONT & RR (4) is required. Plausible because this would be correct if the plant was in Mode 3, which would be true if Reactor coolant temperature was slightly higher.
D. Incorrect – Entry into N2-EOP-CONT & RR (4) is required. Plausible because this would be correct if the plant was in Mode 5, which would be true if the Reactor head was not still fully tensioned.

Technical Reference(s): Technical Specification Table 1.1-1, N2-EOP-CONT & RR (4)

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/10/23 – Added bullet to explain reason why fuel moves are being performed while in Mode 4, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK1.01
	Importance Rating	4.3

HIGH DRYWELL PRESSURE

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the HIGH DRYWELL PRESSURE: Drywell integrity

Proposed Question: #11

The plant has experienced a loss of coolant accident with the following:

- Drywell spray has been initiated due to high Drywell pressure per N2-EOP-PC, Primary Containment Control.
- Drywell pressure is 6 psig and slowly lowering.
- Drywell temperature is 200°F and slowly lowering.

Which one of the following identifies when Drywell spray is required to be secured, in accordance with N2-EOP-PC?

Secure Drywell spray before Drywell…

A. pressure drops to the threshold value of 1.68 psig.

B. pressure drops to the threshold value of 0 psig.

C. temperature drops to the threshold value of 150°F.

D. temperature drops to the threshold value of 125°F.

Proposed Answer: B

Explanation: N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig.

Note: The question meets the K/A by testing a procedural requirement that is exercised due to a high Drywell pressure condition (how to control sprays once initiated). The bases for this procedural requirement is the negative design rating of the Drywell, and therefore directly relates to Drywell integrity.

A. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell pressure of 1.68 psig is the N2-EOP-PC entry condition.
C. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell temperature is also elevated and 150°F is the N2-EOP-PC entry condition.
D. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell temperature is also elevated and 125°F is in the vicinity of normal Drywell temperature at power.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – 2021 NRC #11

Question History: 2021 NRC #11

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA1.09
	Importance Rating	4.2

HIGH REACTOR PRESSURE

Ability to operate or monitor the following as they apply to HIGH REACTOR PRESSURE: RPS

Proposed Question: #12

The plant is operating at 100% power with the following:

- A Main Turbine pressure regulator malfunction causes Reactor pressure to rise.
- Reactor pressure reaches a peak of 1083 psig.

Which one of the following describes the resulting status of scram solenoids and ARI solenoids?

	Scram Solenoids	ARI Solenoids
A.	Energized	Energized
B.	Energized	De-energized
C.	De-energized	Energized
D.	De-energized	De-energized

Proposed Answer: C

Explanation: The high Reactor pressure scram setpoint is 1052 psig (must be ≤1072 per Technical Specifications). Peak Reactor pressure exceeds this value, therefore RPS processes a full scram and de-energizes the scram solenoids (they are de-energize to function). The high Reactor pressure ARI setpoint is 1065 psig. Peak Reactor pressure exceeds this value, therefore RRCS processes an ARI signal and energizes the ARI solenoids (they are energize to function).

A. Incorrect – Scram solenoids are de-energized. Plausible because this would be correct if Reactor pressure remained below 1052 psig or if these solenoids were energize to function.
B. Incorrect – Scram solenoids are de-energized. Plausible because this would be correct if Reactor pressure remained below 1052 psig or if these solenoids were energize to function. ARI solenoids are energized. Plausible because this would be correct if Reactor pressure remained below 1065 psig or if these solenoids were de-energize to function.
D. Incorrect – ARI solenoids are energized. Plausible because this would be correct if Reactor pressure remained below 1065 psig or if these solenoids were de-energize to function.

Technical Reference(s): ARPs 603103, 603403, 603422, Technical Specification Table 3.3.1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 2.1.29
	Importance Rating	4.1

SUPPRESSION POOL HIGH WATER TEMPERATURE

Knowledge of how to conduct system lineups, such as valves, breakers, or switches

Proposed Question: #13

The plant is operating at 85% power with the following:

- An SRV inadvertently opened and has now been closed.
- Suppression Pool water temperature is 92°F and rising slowly.
- Suppression Pool Cooling is being placed in service per N2-EOP-HC, EOP Hard Cards Procedure, Attachment 5, Placing RHR A in Suppression Pool Cooling.
- The Unit Supervisor has directed maximizing cooling with this system.

Which one of the following describes the required valve positioning to achieve this lineup, in accordance with N2-EOP-HC Attachment 5?

	2RHS*MOV8A, HEAT EXCHANGER 1A INLET BYPASS VLV THROTTLE	2SWP*MOV33A, HEAT EXCHANGER 1A SVCE WTR OUTLET VLV
A.	Fully open	Fully open
B.	Fully open	Throttled open
C.	Fully closed	Fully open
D.	Fully closed	Throttled open

Proposed Answer: D

Explanation: N2-EOP-HC Attachment 5 directs fully closing 2RHS*MOV8A. N2-EOP-HC Attachment 5 directs a maximum flow of 7400 gpm through 2SWP*MOV33A. With the valve fully open, this flow would be exceeded. Therefore, 2SWP*MOV33A is throttled open.

A. Incorrect – 2RHS*MOV8A is fully closed. Plausible because this valve is normally fully open and closing it reduces total system flow. 2SWP*MOV33A is throttled open to establish a maximum of 7400 gpm. Plausible because other valves in the lineup are fully open.
B. Incorrect – 2RHS*MOV8A is fully closed. Plausible because this valve is normally fully open and closing it reduces total system flow.
C. Incorrect – 2SWP*MOV33A is throttled open to establish a maximum of 7400 gpm. Plausible because other valves in the lineup are fully open.

Technical Reference(s): N2-EOP-HC Attachment 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 10/10/23 – Fixed typo in last bullet (directly -> directed), based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EK3.03
	Importance Rating	3.8

HIGH DRYWELL TEMPERATURE

Knowledge of the reasons for the following responses or actions as they apply to HIGH DRYWELL TEMPERATURE: Drywell spray

Proposed Question: #14

Which one of the following describes:

- the threshold temperature for initiating Drywell spray due to high Drywell temperature

  and

- the basis for this temperature,

in accordance with N2-EOP-PC (1-3), Primary Containment Control For Modes 1-3?

	Threshold Temperature	Basis
A.	340°F	ADS qualification temperature
B.	340°F	Recirc pump seal qualification temperature
C.	400°F	ADS qualification temperature
D.	400°F	Recirc pump seal qualification temperature

Proposed Answer: A

Explanation: N2-EOP-PC (1-3) requires initiating Drywell spray based on a Drywell temperature threshold of 340°F. The basis for this temperature is the ADS qualification temperature.

B. Incorrect – The basis for this temperature is the ADS qualification temperature. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.
C. Incorrect – N2-EOP-PC (1-3) requires initiating Drywell spray based on a Drywell temperature threshold of 340°F. Plausible because 400°F is the upper limit on the Drywell temperature indication and is part of the basis for multiple EOP limits. The basis for this temperature is the ADS qualification temperature. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.
D. Incorrect – N2-EOP-PC (1-3) requires initiating Drywell spray based on a Drywell temperature threshold of 340°F. Plausible because 400°F is the upper limit on the Drywell temperature indication and is part of the basis for multiple EOP limits.

Technical Reference(s): N2-EOP-PC, N2-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Modified Bank – NMP1 2018 NRC #40

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/11/23 – Added parent question based on NRC comment.

High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: ADS

Proposed Question: #40

Which one of the following describes the requirement in N1-EOP-4, Primary Containment Control, for performing an RPV Blowdown due to high Drywell temperature and the basis of this requirement?

N1-EOP-4 requires RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of…

A. 300°F. This requirement is based on ADS qualification temperatures.

B. 300°F. This requirement is based on Recirc pump seal qualification temperatures.

C. 400°F. This requirement is based on ADS qualification temperatures.

D. 400°F. This requirement is based on Recirc pump seal qualification temperatures.

Proposed Answer: A

Explanation: N1-EOP-4 requires an RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of 300°F. This temperature limit is based on the component qualification temperature for ERV solenoids (rounded down from 301°F).

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK2.05
	Importance Rating	3.8

LOW SUPPRESSION POOL WATER LEVEL

Knowledge of the relationship between the LOW SUPPRESSION POOL WATER LEVEL and the following systems or components: HPCS

Proposed Question: #15

The plant is operating at 100% power with the following:

- A leak has developed from the Suppression Pool.
- Suppression Pool water level is 199' and lowering slowly.
- N2-EOP-PC, Primary Containment Control, has been entered.

Which one of the following describes the direction on how to operate HPCS in this situation, in accordance with N2-EOP-PC?

A. Swap HPCS suction to prevent damage to HPCS

B. Swap HPCS suction to preserve Suppression Pool inventory

C. Use HPCS to add water to the Suppression Pool from the Hotwell

D. Use HPCS to add water to the Suppression Pool from the Condensate Storage Tanks

Proposed Answer: D

Explanation: N2-EOP-PC provides direction for using N2-OP-33 to use HPCS to add water to the Suppression Pool. The added water comes from the Condensate Storage Tanks.

A. Incorrect – HPCS suction is not swapped due to low Suppression Pool water level. Plausible because the Suppression Pool is the alternate suction source for HPCS and the primary suction source for LPCS. Also plausible because HPCS suction does swap on high Suppression Pool water level. Also plausible because if HPCS was taking suction from the Suppression Pool and water level was low, then it would lower NPSH and raise the risk on damage to HPCS.
B. Incorrect – HPCS suction is not swapped due to low Suppression Pool water level. Plausible because the Suppression Pool is the alternate suction source for HPCS and the primary suction source for LPCS. Also plausible because HPCS suction does swap on high Suppression Pool water level.
C. Incorrect – The added water is from the Condensate Storage Tanks. Plausible because the Hotwell is another large source of water similar to the Condensate Storage Tanks and Condensate pumps do take suction from the Hotwell.

Technical Reference(s): N2-EOP-PC, N2-OP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK2.14
	Importance Rating	4.1

REACTOR LOW WATER LEVEL

Knowledge of the relationship between the REACTOR LOW WATER LEVEL and the following systems or components: Emergency generators

Proposed Question: #16

The plant is operating at 100% power with the following:

- A loss of Feedwater injection occurs.
- The Reactor scrams on low Reactor water level.
- Reactor water level reaches a low of 100” during the transient.

Which one of the following describes the resulting status of Emergency Diesel Generators (EDGs)?

	Div I EDG	Div II EDG	Div III EDG
A.	Remains in standby	Remains in standby	Remains in standby
B.	Remains in standby	Remains in standby	Starts
C.	Starts	Starts	Remains in standby
D.	Starts	Starts	Starts

Proposed Answer: B

Explanation: The Div I and II EDGs remain in standby because Reactor water level stays above 17.8". The Div III EDG starts because Reactor water level lowers below 108.8".

A. Incorrect – The Div III EDG starts because Reactor water level lowers below 108.8". Plausible because this would be correct if Reactor water level didn't lower quite so far. Also plausible because there is no high Drywell pressure or loss of offsite power.
C. Incorrect – The Div I and II EDGs remain in standby because Reactor water level stays above 17.8". Plausible because the Div III EDG starts. The Div III EDG starts because Reactor water level lowers below 108.8". Plausible because the Div I and Div II EDGs remain in standby.
D. Incorrect – The Div I and II EDGs remain in standby because Reactor water level stays above 17.8". Plausible because this would be correct if Reactor water level lowered below 17.8".

Technical Reference(s): N2-EOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EA2.01
	Importance Rating	4.3

SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN

Ability to determine or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor power

Proposed Question: #17

The plant is operating at 100% power with the following:

- A spurious Group 1 isolation occurs.
- Multiple control rods fail to fully insert.
- RRCS failed to initiate.
- APRMs are unavailable.
- Six SRVs are open.
- Reactor pressure is 1120 psig and stable.
- Reactor water level is 180” and stable.
- RCIC is NOT running.

Which one of the following ranges contains the approximate value of Reactor power?

A. 5-10%

B. 15-20%

C. 30-35%

D. 55-60%

Proposed Answer: C

Explanation: A Group 1 isolation signal causes the MSIVs to close. This results in Turbine Bypass Valves being unavailable. With RCIC also not operating, the only steam flow from the Reactor is passing through the SRVs. Each SRV passes steam flow equivalent to approximately 5% Reactor power (~900,000 lbm/hr divided by 100% steam flow of 17.6×10^6 lbm/hr = approximately 5%). With 6 SRVs open and Reactor pressure stable, Reactor power is approximately 30%.

A. Incorrect – Reactor power is approximately 30%. Plausible because this would be correct with 1 or 2 SRVs open.
B. Incorrect – Reactor power is approximately 30%. Plausible because this would be correct with 3 or 4 SRVs open or if other systems were drawing steam flow too.
D. Incorrect – Reactor power is approximately 30%. Plausible because this would be correct with 11 or 12 SRVs open or if other systems were drawing steam flow too.

Technical Reference(s): N2-OP-34, EM-900A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Bank - JAF 16-1 NRC #63

Question History: JAF 16-1 NRC #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK2.03
	Importance Rating	3.8

HIGH OFFSITE RADIOACTIVITY RELEASE RATE

Knowledge of the relationship between the HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components: Plant ventilation systems

Proposed Question: #18

The plant is operating at 100% power with the following:

- Stack WRGMS radiation reading is normal.
- Vent WRGMS radiation reading is higher than normal.

Which one of the following identifies the possible release source?

A. Standby Gas Treatment fan exhaust

B. Main Steam Tunnel Ventilation exhaust

C. Above Refuel Floor Ventilation exhaust

D. Turbine Building Ventilation exhaust

Proposed Answer: C

Explanation: The Above Refuel Floor Ventilation exhaust is routed to the Reactor Building Vent Stack, which is monitored by Vent WRGMS, not Stack WRGMS.

A. Incorrect – Standby Gas Treatment fan exhaust is monitored by Stack WRGMS, not Vent WRGMS. Plausible because some ventilation exhausts go to each of these detectors.
B. Incorrect – Main Steam Tunnel Ventilation exhaust is monitored by Stack WRGMS, not Vent WRGMS. Plausible because some ventilation exhausts go to each of these detectors.
D. Incorrect – Turbine Building Ventilation exhaust is monitored by Stack WRGMS, not Vent WRGMS. Plausible because some ventilation exhausts go to each of these detectors.

Technical Reference(s): PID-052E & G

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-8

Question Source: Bank – 2017 Cert #63

Question History: 2012 NRC #53

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK1.02
	Importance Rating	3.4

PLANT FIRE ON SITE

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PLANT FIRE ON SITE: Firefighting methods for each type of fire

Proposed Question: #19

The plant is operating at 100% power with the following:

- A fire has developed in the Standby Gas Treatment train A charcoal filter.
- Annunciator 870109, SBGTS TRAIN A SYSTEM TROUBLE, is in alarm with the following computer points alarming:
  - GTSTC03, GTS FLTR A ACT CHAR TEMP
  - GTSTC12, GTS FLTR A ACT CHAR TEMP

Which one of the following describes the means of fire suppression for this charcoal filter?

A. CO_2 flooding that automatically initiates.

B. Water deluge that automatically initiates.

C. CO_2 flooding that must be manually initiated.

D. Water deluge that must be manually initiated.

Proposed Answer: D

Explanation: The Standby Gas Treatment charcoal filters are provided with a water deluge fire suppression system. This suppression system is normally blocked by closure of manual valves 2FPW-V430 and 2FPW-V817, therefore it must be manually initiated.

Note: The question meets the K/A by presenting a situation with a fire in the plant and testing the operational implications of the firefighting method for this fire (what suppression system will be used and how it is initiated).

A. Incorrect – Suppression is from water deluge, not CO_2. Plausible because the presence of water in a charcoal filter is undesirable due to an exothermic reaction, which would not be present with CO_2 usage, and CO_2 is used in other fire suppression applications. Suppression is manual, not automatic. Plausible because temperature detectors are in alarm, other systems automatically initiate, and GTS decay heat removal valves have automatic response.
B. Incorrect – Suppression is manual, not automatic. Plausible because temperature detectors are in alarm, other systems automatically initiate, and GTS decay heat removal valves have automatic response.
C. Incorrect – Suppression is from water deluge, not CO2. Plausible because the presence of water in a charcoal filter is undesirable due to an exothermic reaction, which would not be present with CO2 usage, and CO2 is used in other fire suppression applications.

Technical Reference(s): N2-OP-61B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-286000-RBO-5

Question Source: Bank – 2017 NRC #41

Question History: 2017 NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AA1.06
	Importance Rating	3.5

GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES

Ability to operate or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AC electrical distribution system

Proposed Question: #20

The plant is operating at 100% power with the following:

- Strong thunderstorms are causing unstable grid conditions.
- N2-SOP-70, Major Grid Disturbances, has been entered.
- Power Control reports that loss of 115KV off-site power is imminent.

Which one of the following describes the required control of the Division I and II Emergency Diesel Generators (EDGs), in accordance with N2-SOP-70?

A. Start and load the EDGs

B. Start the EDGs and run them unloaded

C. Place the EDGs in the Maintenance Mode

D. Maintain the EDGs in standby and verify fuel oil day tank levels are maximized

Proposed Answer: B

Explanation: With loss of off-site power imminent, N2-SOP-70 requires starting the Division I and II EDGs and running them unloaded. A caution in the procedure states, “Operating an EDG in parallel with off-site power while unstable grid conditions exist can result in the loss of the EDG and the bus during a grid transient.”

A. Incorrect – The EDGs are maintained unloaded. Plausible because the EDGs are started. Also plausible that they would be placed on their respective buses to maintain continuity of power if/when off-site power is lost.
C. Incorrect – The EDGs are started. Plausible because N2-OP-100A does have a section for placing the EDGs in Maintenance Mode and this could be misunderstood to be a mode that readies them for a surveillance test run or similar. Also plausible that Maintenance Mode would be used to prevent spurious start due to fluctuating voltages.
D. Incorrect – The EDGs are started. Plausible that they would be maintained in standby to preserve fuel.

Technical Reference(s): N2-SOP-70

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP70C01 EO-2

Question Source: Bank – 4/23 NRC #20

Question History: 4/23 NRC #20

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AA1.07
	Importance Rating	3.4

LOSS OF MAIN CONDENSER VACUUM

Ability to operate or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Circulating water system

Proposed Question: #21

The plant is operating at 100% power with the following:

- Annunciator 851301, CIRCULATING WATER PUMP 1A THRU 1F AUTO TRIP, alarms.
- Circulating Water pump 1D has tripped.
- Annunciator 851358, TURBINE CNSR A/B/C VACUUM LOW, alarms.
- Condenser 1A vacuum indicates 26.9” Hg.
- Condenser 1B vacuum indicates 23.8” Hg.
- Condenser 1C vacuum indicates 26.8” Hg.

Which one of the following describes the need to enter N2-SOP-09, Loss of Main Condenser Vacuum, and N2-SOP-21, Turbine Trip, in accordance with plant procedures?

	N2-SOP-09 Entry	N2-SOP-21 Entry
A.	Required	NOT required
B.	Required	Required
C.	NOT required	NOT required
D.	NOT required	Required

Proposed Answer: B

Explanation: N2-SOP-09 entry is required based on both ARP 851301 and 851358. ARP 851301 and N2-OP-10A P&L D.14 require vacuum differential between any two condenser sections to be less than 2" Hg. The given indications show a differential of greater than 2" Hg., therefore it is UNSAT. This requires entry into N2-SOP-21.

A. Incorrect – Vacuum differential is UNSAT, therefore N2-SOP-21 entry is required. Plausible because all Main Condenser section vacuum indications remain relatively close and above the Main Turbine trip criteria (22.1" Hg).
C. Incorrect – N2-SOP-09 entry is required. Plausible because 5 of the 6 Circulating Water pumps are still operating, Main Condenser vacuum is above limits in 2 of the 3 Main Condenser sections, and all Main Condenser vacuum indications remain above the Main Turbine trip criteria (22.1" Hg). Vacuum differential is UNSAT, therefore N2-SOP-21 entry is required. Plausible because all Main Condenser section vacuum indications remain relatively close and above the Main Turbine trip criteria (22.1" Hg).
D. Incorrect – N2-SOP-09 entry is required. Plausible because 5 of the 6 Circulating Water pumps are still operating, Main Condenser vacuum is above limits in 2 of the 3 Main Condenser sections, and all Main Condenser vacuum indications remain above the Main Turbine trip criteria (22.1" Hg).

Technical Reference(s): ARPs 851301 and 851358, N2-SOP-09, N2-OP-10A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP09C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AK3.01
	Importance Rating	3.7

LOW REACTOR WATER LEVEL

Knowledge of the reasons for the following responses or actions as they apply to LOW REACTOR WATER LEVEL: Recirculation runback

Proposed Question: #22

Which one of the following describes a reason for the Recirc FCV runback?

A. Prevent cavitation of Recirc pumps

B. Add negative reactivity during an ATWS

C. Lower Reactor power following the trip of a Feedwater pump

D. Reduce velocity head effects on Wide Range level instrumentation

Proposed Answer: C

Explanation: The reason for the Recirc FCV runback is to lower Reactor power following a trip of a Feedwater pump. This is designed to anticipate low Reactor water level and reduce the severity of the low Reactor water level transient.

A. Incorrect – The reason for the Recirc FCV runback is to lower Reactor power following a trip of a Feedwater pump. Plausible because there is a Feedwater flow interlock in the Recirc pump speed circuitry that is designed to prevent cavitation. Also plausible because lowering Recirc flow through a FCV runback does reduce the chances of cavitation.
B. Incorrect – The reason for the Recirc FCV runback is to lower Reactor power following a trip of a Feedwater pump. Plausible because there is a Recirc pump trip designed to add negative reactivity during an ATWS. Also plausible because lowering Recirc flow through a FCV runback also does add negative reactivity.
D. Incorrect – The reason for the Recirc FCV runback is to lower Reactor power following a trip of a Feedwater pump. Plausible because this is the purpose of another interlock in the Recirc pump speed circuitry. Also plausible because lowering Recirc flow through a FCV runback also does help reduce velocity head effects on Wide Range level instrumentation.

Technical Reference(s): ARPs 602210(222)

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202002-RBO-5

Question Source: Bank – 2019 NRC #22

Question History: 2019 NRC #22

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 2.1.32
	Importance Rating	3.8

HIGH DRYWELL PRESSURE

Ability to explain and apply system precautions, limitations, notes, or cautions

Proposed Question: #23

The plant is operating at 100% power with the following:

- Degraded performance of the Drywell cooling system is causing rising Drywell temperature and pressure.
- Average Drywell temperature is 140°F and rising slowly.
- Drywell pressure is 0.70 psig and rising slowly.
- Drywell cooling is in a normal lineup with 2DRS-UC3B in standby.

Which one of the following describes the ability to place 2DRS-UC3B in operation with 2DRS-UC3A, in accordance with N2-OP-60, Drywell Cooling?

Placing 2DRS-UC3B in operation with 2DRS-UC3A is…

A. allowed without restriction

B. allowed for a limited period of time

C. NOT allowed due to an installed interlock

D. NOT allowed due to a procedure limitation

Proposed Answer: B

Explanation: N2-OP-60 does allow placing 2DRS-UC3B in operation with 2DRS-UC3A, although operation of both Drywell coolers in parallel is abnormal and a procedure note states that it should not be extended beyond 8 hours.

A. Incorrect – A procedure note states that it should not be extended beyond 8 hours. Plausible because there are no similar restrictions on any of the other Drywell coolers.
C. Incorrect – N2-OP-60 does allow placing 2DRS-UC3B in operation with 2DRS-UC3A. Plausible because this is abnormal and should not be extended beyond 8 hours.
D. Incorrect – N2-OP-60 does allow placing 2DRS-UC3B in operation with 2DRS-UC3A. Plausible because this is abnormal and should not be extended beyond 8 hours.

Technical Reference(s): N2-OP-60

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-234004-RBO-10

Question Source: Modified Bank – 2019 NRC #23

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 1
Group # 2
K/A # 295012 AK2.02
Importance Rating 3.6

High Drywell Temperature

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling

Proposed Question: #23

The plant is operating at 100% power with the following:

- Degraded performance of the Drywell cooling system is causing rising Drywell temperatures.
- Average Drywell temperature is 140°F and slowly rising.
- Drywell cooling is in a normal lineup with 2DRS-UC3B in standby.

Which one of the following describes:

- the ability to place 2DRS-UC3B in operation with 2DRS-UC3A, in accordance with N2-OP-60, Drywell Cooling,

and

- the status of average Drywell temperature with respect to the limit in Technical Specification 3.6.1.5, Drywell Air Temperature?

Placing 2DRS-UC3B in operation with 2DRS-UC3A is ___(1)___.

Average Drywell air temperature is ___(2)___ the limit in Technical Specification 3.6.1.5.

	(1)		(2)
A.	allowed		below
B.	allowed		above
C.	NOT allowed		below
D.	NOT allowed		above

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295013 AA2.02
	Importance Rating	3.5

HIGH SUPPRESSION POOL TEMPERATURE

Ability to determine or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Localized heating/stratification

Proposed Question: #24

The Reactor has scrammed with the following:

- MSIVs are closed.
- The Unit Supervisor has directed Reactor pressure controlled 800-1000 psig using SRVs.

Which one of the following describes how SRVs should be operated and the associated reason, in accordance with the Emergency Operating Procedures?

Cycle…

A. through all SRVs sequentially, as needed, to verify they are all seated properly

B. the same SRV repeatedly, as needed, to prevent thermal cycling of multiple SRV tailpipes

C. through all SRVs sequentially, as needed, to distribute the heat load uniformly throughout the Suppression Pool

D. the same SRV repeatedly, as needed, to maintain Suppression Pool heating as close as possible to the Suppression Pool Cooling injection point

Proposed Answer: C

Explanation: SRV discharges are designed to be spaced at points around the Suppression Pool. N2-EOP-RPV contains direction to use SRVs in a preferred sequence. The basis for this direction is to distribute heat uniformly throughout the suppression pool to avoid high local pool temperatures which may result in inefficient pool cooling.

A. Incorrect – The basis for this direction is to distribute heat uniformly throughout the suppression pool to avoid high local pool temperatures which may result in inefficient pool cooling. Plausible because cycling a leaking SRV is a method used to attempt to reseat the valve.
B. Incorrect – N2-EOP-RPV contains direction to use SRVs in a sequence, not a single SRV repeatedly. Plausible because opening a new SRV will cause a larger thermal transient since its tailpipe is colder.
D. Incorrect – N2-EOP-RPV contains direction to use SRVs in a sequence, not a single SRV repeatedly. Plausible because Suppression Pool cooling does discharge to the Suppression Pool through a single line.

Technical Reference(s): N2-EOP-RPV, N2=EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2

Question Source: Bank - NMP1 2023 NRC #23

Question History: NMP1 2023 NRC #23

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 10/11/23 – Revised to more closely match the format of the most recent NRC exam usage (NMP1 2023 NRC exam), based on NRC comment.
TRH 10/25/23 – Fixed error in key.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EK1.01
	Importance Rating	3.8

HIGH SUPPRESSION POOL WATER LEVEL

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the HIGH SUPPRESSION POOL WATER LEVEL: Containment integrity

Proposed Question: #25

Which one of the following identifies the threshold for Suppression Pool water level that requires securing Drywell Sprays in accordance with N2-EOP-PC, Primary Containment Control?

A. 201.0’

B. 207.0’

C. 210.0’

D. 217.0’

Proposed Answer: D

Explanation: N2-EOP-PC requires securing Drywell Spray if Suppression Pool water level cannot be maintained below the threshold of 217'.

Note: One operational implication of Containment integrity as it applies to high Suppression Pool water level is when to secure Drywell spray as Suppression Pool water level rises. Drywell spray is secured in this situation to avoid submerging the Suppression Chamber to Drywell vacuum breakers, to ensure proper functioning of the Containment and to avoid damaging the Containment.

A. Incorrect – The threshold is 217', not 201'. Plausible because 201' is higher than normal Suppression Pool water level and an entry condition into N2-EOP-PC.
B. Incorrect – The threshold is 217', not 207'. Plausible because 207' is the approximate value of the SRVTPLL at the initial Reactor pressure (100% power).
C. Incorrect – The threshold is 217', not 210'. Plausible because 210' is the value of the SRVTPLL at the current Reactor pressure of 50 psig.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – 2019 NRC #24

Question History: 2019 NRC #24

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EK2.07
	Importance Rating	4.2

HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS

Knowledge of the relationship between the HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following systems or components: Systems required for adequate core cooling

Proposed Question: #26

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is -10" and stable.
- RCIC is injecting 600 gpm.
- NO other injection systems are available.

Then, a steam leak from RCIC results in the following:

- Annunciator 601332, RCIC EQUIP ROOM TEMPERATURE HIGH, alarms.
- RCIC equipment room temperature is 100°F and rising slowly.
- Annunciator 851244, REACTOR BLDG AREA RADN MON ACTIVATED, alarms.
- Multiple Reactor Building Area Radiation monitors are in alarm.
- The highest Reactor Building area radiation level is 100 mr/hr and rising slowly.

Which one of the following describes the required response, in accordance with the Emergency Operating Procedures?

A. Isolate RCIC based on area temperature

B. Isolate RCIC based on area radiation levels

C. Do NOT isolate RCIC because its use is required for EOP actions

D. Do NOT isolate RCIC because entry is NOT yet required into N2-EOP-SC, Secondary Containment Control

Proposed Answer: C

Explanation: Reactor water level is just above the top of active fuel and stable with RCIC at maximum flow. Additionally, no other injection systems are available to replace RCIC. Therefore RCIC is required by N2-EOP-RPV to maintain adequate core cooling, the main mitigating strategy of the procedure. This is in contrast to the mitigating strategy of N2-EOP-SC, which would lead to eventual isolation of RCIC. The overall EOP strategy places greater importance on adequate core cooling, thus N2-EOP-RPV (and, as a result, ARP 601332) direct defeating the RCIC high area temperature isolation using N2-EOP-6.2 and continuing injection using RCIC.

A. Incorrect – RCIC is not isolated because it is required for adequate core cooling. Plausible because N2-EOP-SC would otherwise require isolating RCIC and area temperature is above an alarm setpoint.
B. Incorrect – RCIC is not isolated because it is required for adequate core cooling. Plausible because N2-EOP-SC would otherwise require isolating RCIC and area radiation levels are above an alarm setpoint.
D. Incorrect – RCIC is not isolated because it is required for adequate core cooling. Plausible because the given temperature and radiation information is not overly severe.

Technical Reference(s): N2-EOP-RPV, N2-EOP-SC, ARP 601332

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Modified Bank – 2014 NRC #22

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 2.4.6
	Importance Rating	3.7

PCIS/Nuclear Steam Supply Shutoff

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: #22

The plant has experienced a LOCA. Conditions are as follows:

- Reactor water level is -10 inches and stable.
- RCIC is injecting at 600 gpm.
- No other injection systems are available.

Then…

- Annunciator 601332, RCIC EQUIP ROOM TEMPERATURE HIGH, alarms.
- RCIC equipment room temperature is 100°F and slowly rising.
- An Operator reports a steam leak is present in the RCIC equipment room.

Which one of the following describes the required response, in accordance with the Emergency Operating Procedures?

A. Shutdown and isolate RCIC immediately.

B. Defeat the RCIC high area temperature isolation and continue use of RCIC for Reactor injection.

C. Continue use of RCIC for Reactor injection until a high-high room temperature isolation is reached. Then if Reactor water level cannot be maintained above -14 inches, perform Steam Cooling.

D. Continue use of RCIC for Reactor injection until a high-high room temperature isolation is reached. Then if Reactor water level cannot be maintained above -14 inches, perform an RPV Blowdown.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K4.03
	Importance Rating	3.2

RHR/LPCI: INJECTION MODE

Knowledge of RHR/LPCI: INJECTION MODE design features and/or interlocks that provide for the following: Pump minimum flow protection

Proposed Question: #27

A plant startup is in progress with the following:

- A small loss of coolant accident occurs.
- The Reactor scrams.
- Reactor water level is 150” and stable.
- Drywell pressure is 2 psig and stable.
- Reactor pressure is 400 psig and stable.

Which one of the following describes the resulting status of RHR loop C?

A. Remains in standby

B. Automatically starts and injects to the Reactor

C. Automatically starts and runs on minimum flow to the CSTs.

D. Automatically starts and runs on minimum flow to the Suppression Pool

Proposed Answer: D

Explanation: Reactor water level is above the LPCI initiation setpoint of 17.8", but Drywell pressure is high enough to initiate LPCI. This automatically starts RHR loop C from standby. With Reactor pressure at 400 psig, RHR loop C is NOT capable of injecting to the Reactor. Therefore, RHR loop C is running on minimum flow. Minimum flow is routed to the Suppression Pool, not the CSTs.

A. Incorrect – LPCI initiates on high Drywell pressure. Plausible because Reactor water level has not reached the initiation setpoint.
B. Incorrect – With Reactor pressure at 400 psig, RHR loop C is NOT capable of injecting to the Reactor. Plausible because this would be correct if Reactor pressure was lower.
C. Incorrect – Minimum flow is routed to the Suppression Pool, not the CSTs. Plausible that the minimum flow would be to the CSTs since they are another large water source that receive various pump discharges (ex. RCIC full flow test line).

Technical Reference(s): N2-OP-31, N2-EOP-RPV, PID 31G, PID 31B, PID 31C

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-5

Question Source: Modified Bank – 2017 Cert #21

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 1
K/A # 203000 2.4.2
Importance Rating 4.5

RHR/LPCI: Injection Mode

Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: #21

The plant is shutdown with the following:

- RHR loop B is operating in Suppression Pool cooling.
- A small loss of coolant accident occurs.
- Reactor water level lowers to the N2-EOP-RPV entry condition value and stabilizes.
- Drywell pressure rises to the N2-EOP-RPV entry condition value and stabilizes.
- Reactor pressure is 200 psig and slowly lowering.

Which one of the following describes the status of RHR loops A and B?

	RHR Loop A		RHR Loop B
A.	Remains in standby		Remains in Suppression Pool cooling
B.	Automatically starts and runs on min flow		Remains in Suppression Pool cooling
C.	Automatically starts and runs on min flow		Suppression Pool cooling lineup isolates and runs on min flow
D.	Automatically starts and injects to the Reactor		Suppression Pool cooling lineup isolates and injects to the Reactor

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K5.03
	Importance Rating	3.9

SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Decay heat removal

Proposed Question: #28

The plant is shutdown in Mode 4 with the following:

- RHR loop A is aligned for Shutdown Cooling.
- All other RHR pumps are available and in standby.

Then, the following occurs:

- A leak develops between 2RHS*MOV112, SDC SUCT INBD ISOL VLV, and 2RHS*MOV113, SDC SUCT OUTBD ISOL VLV THROTTLE.
- The Shift Manager directs securing RHR pump A and isolating the leak by closing 2RHS*MOV112 and 2RHS*MOV113.

Which one of the following describes the ability to re-establish Reactor decay heat removal using RHR in either the normal or alternate Shutdown Cooling modes while 2RHS*MOV112 and 2RHS*MOV113 are closed?

	Normal Shutdown Cooling Mode	Alternate Shutdown Cooling Mode
A.	Available	Available
B.	Available	NOT available
C.	NOT available	Available
D.	NOT available	NOT available

Proposed Answer: C

Explanation: The normal Shutdown Cooling mode cannot be re-established for Reactor decay heat removal because 2RHS*MOV112 and 2RHS*MOV113 must be open to use either loop of RHR in this mode. The alternate Shutdown Cooling mode can be established because it does not required 2RHS*MOV112 and 2RHS*MOV113 to be open. Instead, this flowpath circulates water from the Suppression Pool through the Reactor and out the SRVs using an RHR pump.

A. Incorrect – The normal Shutdown Cooling mode cannot be re-established for Reactor decay heat removal while 2RHS*MOV112 and 2RHS*MOV113 are closed. Plausible that RHR pump B would use a different suction path than RHR pump A, such as it does for the LPCI mode.
B. Incorrect – The normal Shutdown Cooling mode cannot be re-established for Reactor decay heat removal while 2RHS*MOV112 and 2RHS*MOV113 are closed. Plausible that RHR pump B would use a different suction path than RHR pump A, such as it does for the LPCI mode. The alternate Shutdown Cooling mode can be established because it does not required 2RHS*MOV112 and 2RHS*MOV113 to be open. Plausible that it would not be available because the normal Shutdown Cooling is not available and because RHR pumps cannot be aligned with their suction from the Reactor.
D. Incorrect – The alternate Shutdown Cooling mode can be established because it does not required 2RHS*MOV112 and 2RHS*MOV113 to be open. Plausible that it would not be available because the normal Shutdown Cooling is not available and because RHR pumps cannot be aligned with their suction from the Reactor.

Technical Reference(s): N2-SOP-31, N2-OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-11

Question Source: Modified Bank - JAF 14-1 NRC #43

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AK2.03
	Importance Rating	3.6

Loss of Shutdown Cooling

Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: RHR/shutdown cooling

Proposed Question: #43

The plant is shutdown during a refueling outage with the following:

- New fuel is being loaded into the Reactor core.
- RHR pump A is running for Shutdown Cooling.
- All other RHR pumps are available and in standby.
- Then, a leak develops from a through-wall crack between 10MOV-18, RHR Shutdown Cooling Outboard Isolation Valve, and 10MOV-17, RHR Shutdown Cooling Inboard Isolation Valve.
- The Shift Manager has directed securing RHR pump A and isolating the leak by closing 10MOV-18 and 10MOV-17.

Which one of the following describes the impact of this evolution on Shutdown Cooling?

A. Shutdown Cooling will be secured temporarily during RHR pump A shutdown, but can be re-established with any of the other three RHR pumps once the leak is isolated.

B. Shutdown Cooling will be secured temporarily during RHR pump A shutdown, but can be re-established with either RHR pump B or D once the leak is isolated. RHR pump C will be unavailable for Shutdown Cooling.

C. The RHR Shutdown Cooling mode will be unavailable as long as 10MOV-18 and 10MOV-17 are closed. Alternate decay heat removal using RHR in the Fuel Pool Cooling Assist mode may be manually aligned.

D. The RHR Shutdown Cooling mode will be unavailable as long as 10MOV-18 and 10MOV-17 are closed. Alternate decay heat removal using RHR in the Fuel Pool Cooling Assist mode is NOT available to be manually aligned.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K2.02
	Importance Rating	3.8

LOW PRESSURE CORE SPRAY SYSTEM

Knowledge of electrical power supplies to the following: Valves

Proposed Question: #29

The plant is operating at 100% power with the following:

- LPCS is in a normal standby lineup.
- 600 VAC 2EJS*US1 de-energizes due to a sustained electrical fault.

Then, a loss of coolant accident results in a condition requiring automatic LPCS injection.

Which one of the following describes the response of LPCS?

2CSL*P1…

A. starts and runs on minimum flow

B. starts and injects to the Reactor vessel

C. does NOT start, but can be manually started

D. does NOT start and CANNOT be manually started

Proposed Answer: A

Explanation: Loss of 2EJS*US1 results in loss of motive power for multiple LPCS MOVs, including 2CSL*MOV104, LPCS Injection Valve, and 2CSL*MOV107, LPCS Min Flow to Suppression Pool. The normal standby lineup has MOV104 initially closed and MOV107 initially open. With loss of power to these valves, they remain in these positions even upon receipt of the valid injection signal. 2CSL*P1 still has power from the upstream 4160 VAC 2ENS*SWG101 and initiation logic still functions with power from 2BYS*PNL201A. Therefore, 2CSL*P1 starts and runs on minimum flow through the open 2CSL*MOV107, but does not inject to the Reactor vessel.

B. Incorrect – The pump runs on min flow and does not inject to the Reactor vessel. Plausible that the standby lineup would have injection aligned or a UPS would still power the injection valve.
C. Incorrect – 2CSL*P1 still starts. Plausible that the logic would be affected by this power loss and prevent automatic pump start, but still allow manual start.
D. Incorrect – 2CSL*P1 still starts. Plausible that the logic and control would be affected by this power loss and prevent pump start.

Technical Reference(s): Emergency AC Big Note, N2-OP-32, N2-OP-32-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209001-RBO-8

Question Source: Bank - 2019 NRC #30

Question History: 2019 NRC #30

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 A3.01
	Importance Rating	4.2

HIGH PRESSURE CORE SPRAY SYSTEM

Ability to monitor automatic operation of the HIGH PRESSURE CORE SPRAY SYSTEM including: Valve operation

Proposed Question: #30

A plant transient results in the following:

Time (min)	Condition
0	Reactor water level is 108.8” and lowering
5	Reactor water level is 202.3” and rising
10	Reactor water level is 202.3” and lowering
15	Reactor water level is 108.8” and lowering

Which one of the following describes the operation of HPCS at times 6 minutes and 16 minutes?

	Operation at 6 Minutes 2CSH*P1 is…	Operation at 16 Minutes 2CSH*P1 is..
A.	tripped	tripped
B.	tripped	running with the injection valve open
C.	running with the injection valve closed	running with the injection valve open
D.	running with the injection valve closed	running with the injection valve closed

Proposed Answer: C

Explanation: At time 0 minutes, HPCS receives a start/injection signal due to low Reactor water level (108.8”). This starts 2CSH*P1 and opens the injection valve. At time 5 minutes, HPCS receives a signal to close the injection valve on high Reactor water level (202.3”). At time 10 minutes, the high level condition clears, but the injection valve remains closed. At time 15 minutes, HPCS receives another injection signal on low Reactor water level. This causes the injection valve to re-open.

A. Incorrect – At 6 minutes, 2CSH*P1 is running with the injection valve closed. Plausible that the pump would receive a trip signal, since it receives a start signal on low level and other pumps (Feedwater) receive trip signals on high level. At 16 minutes, 2CSH*P1 is running with the injection valve open. Plausible that, if the pump initially tripped, it would remain tripped until operator action was taken to reset the trip (there is a HI WTR LEVEL SEAL IN RESET pushbutton).
B. Incorrect – At 6 minutes, 2CSH*P1 is running with the injection valve closed. Plausible that the pump would receive a trip signal, since it receives a start signal on low level and other pumps (Feedwater) receive trip signals on high level.
D. Incorrect – At 16 minutes, 2CSH*P1 is running with the injection valve open. Plausible that, the injection valve would remain closed until operator action was taken to reset the earlier closure (there is a HI WTR LEVEL SEAL IN RESET pushbutton).

Technical Reference(s): N2-OP-33, ARP 601718, HPCS electrical prints

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-5

Question Source: Modified Bank – JAF 17-2 NRC #30

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K5.01
	Importance Rating	3.3

High Pressure Coolant Injection

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM: Turbine operation: BWR-2,3,4

Proposed Question: #30

A plant transient results in the following:

Time (min)	Condition
0	Reactor water level is 126.5" and lowering.
5	Reactor water level is 222.5" and rising.
10	Reactor water level is 222.5" and lowering.
15	Reactor water level is 126.5" and lowering.

Which one of the following describes the operation of HPCI at times 6 minutes and 16 minutes?

	Operation at 6 Minutes	Operation at 16 Minutes
A.	Tripped	Tripped
B.	Tripped	Running with injection valve open
C.	Running with injection valve closed	Running with injection valve open
D.	Running with injection valve closed	Running with injection valve closed

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K2.01
	Importance Rating	3.3

Instrument Air System

Knowledge of electrical power supplies to the following: Instrument air compressor

Proposed Question: #31

Which one of the following is the power supply to 2IAS-C3B, Instrument Air Compressor B?

A. 2EJS*US1

B. 2EJS*US3

C. 2NJS-US5

D. 2NJS-US6

Proposed Answer: D

Explanation: The power supply to 2IAS-C3B, Instrument Air Compressor B, is 2NJS-US6.

A. Incorrect – The power supply to 2IAS-C3B, Instrument Air Compressor B, is 2NJS-US6. Plausible because this is a similar bus, but within the emergency AC distribution.
B. Incorrect – The power supply to 2IAS-C3B, Instrument Air Compressor B, is 2NJS-US6. Plausible because this is a similar bus, but within the emergency AC distribution.
C. Incorrect – The power supply to 2IAS-C3B, Instrument Air Compressor B, is 2NJS-US6. Plausible because this is the power supply to 2IAS-C3A, Instrument Air Compressor A.

Technical Reference(s): N2-OP-19-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K3.01
	Importance Rating	4.0

STANDBY LIQUID CONTROL SYSTEM

Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the following systems or system parameters: Reactor power

Proposed Question: #32

A failure to scram has occurred with the following:

- RRCS failed to actuate.
- Standby Liquid Control (SLC) pump 2SLS*P1A, PMP 1A, keylock control switch has been placed in the PUMP A RUN position.
- 2SLS*MOV1A, SLC STORAGE TK OUTLET VLV, indicates open.
- 2SLS*P1A red light is illuminated.
- 2SLS*VEX3A ready light is illuminated.
- 2SLS*P1A discharge pressure is approximately 1650 psig.

Which one of the following describes whether SLC will lower Reactor power or NOT and the flow of boron solution?

SLC will…

A. lower Reactor power because boron is being injected through a Feedwater sparger.

B. lower Reactor power because boron is being injected through the HPCS injection line.

C. NOT lower Reactor power because boron is flowing to the equipment drain system.

D. NOT lower Reactor power because boron is flowing back to the suction of 2SLS*P1A.

Proposed Answer: D

Explanation: 2SLS*VEX3A ready light being illuminated indicates that the squib failed to fire. Additionally, the discharge pressure of 1650 psig is abnormally high. This indicates that the explosive valve did not open. Boron solution is therefore flowing through 2SLS*RV2A (nominally set at 1600 psig) back to the suction of the pump. Therefore, boron is not being injected to the Reactor and Reactor power will not lower.

A. Incorrect – Reactor power will not lower because boron solution is flowing back to the suction of the SLS pump. Plausible because the given indications show that the pump is running and developing adequate discharge pressure. Also plausible because the Feedwater sparger is one possible injection points to the Reactor vessel.
B. Incorrect – Reactor power will not lower because boron solution is flowing back to the suction of the SLS pump. Plausible because the given indications show that the pump is running and developing adequate discharge pressure. Also plausible because the HPCS sparger is the normal injection point for SLS into the Reactor.
C. Incorrect – Boron solution is flowing back to the suction of the SLS pump. Plausible because relief valves in the CRD system relieve to the equipment drain system and many other discharge locations would be undesirable for boron solution (Torus, CSTs, Hotwell, etc.).

Technical Reference(s): PID-36A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-211000-RBO-3

Question Source: Bank – 2017 NRC #7

Question History: 2017 NRC #7

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 291008 K1.06
	Importance Rating	3.6

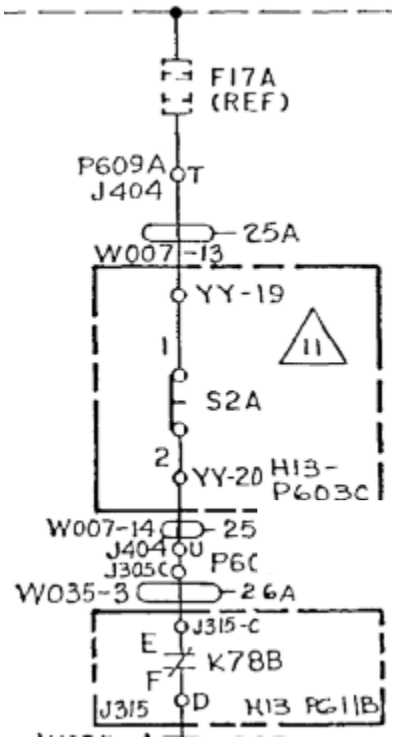
REACTOR PROTECTION SYSTEM

BREAKERS, RELAYS, AND DISCONNECTS

Interpreting one-line diagram of control circuitry

Proposed Question: #33

Given the following excerpt from the RPS control circuitry:



Which one of the following describes the component labeled “S2A”?

A. Pushbutton that energizes this portion of the circuit when depressed

B. Pushbutton that de-energizes this portion of the circuit when depressed

C. Sensor that energizes this portion of the circuit when the setpoint is exceeded

D. Sensor that de-energizes this portion of the circuit when the setpoint is exceeded

Proposed Answer: B

Explanation: The component labeled “S2A” is a pushbutton. Based on how the pushbutton is drawn, it closes the circuit normally (when not depressed) and opens the circuit when depressed. Therefore, when depressed, it de-energizes this portion of the circuit.

A. Incorrect – Based on how the pushbutton is drawn, it energizes the circuit normally (when not depressed) and de-energizes the circuit when depressed. Plausible because this would be correct for some other pushbuttons in the RPS control circuitry.
C. Incorrect – The component is a pushbutton. Plausible because there are many sensor inputs in the RPS control circuitry and the component identifier starts with an “S”.
D. Incorrect – The component is a pushbutton. Plausible because there are many sensor inputs in the RPS control circuitry and the component identifier starts with an “S”.

Technical Reference(s): BC06Ir4_Breakers, 807E166TY Sheets 3 and 5

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: BC06Ir4_Breakers 20

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 10/11/23 – Re-worded answers from open/close to energize/de-energize, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K1.05
	Importance Rating	3.2

REACTOR PROTECTION SYSTEM

Knowledge of the physical connections and/or cause and effect relationships between the REACTOR PROTECTION SYSTEM and the following systems: Process radiation monitoring system

Proposed Question: #34

The plant is operating at 100% power.

Which one of the following identifies radiation monitors that would lead to an automatic Reactor scram if they experienced a sustained high radiation condition and NO operator action was taken?

A. Offgas

B. Vent WRGMs

C. Stack WRGMs

D. Containment Atmosphere

Proposed Answer: A

Explanation: A sustained high radiation condition on Offgas radiation monitors would cause an Offgas isolation. At 100% power with no operator action, this leads to a loss of Main Condenser vacuum, Main Turbine trip, and automatic Reactor scram.

B. Incorrect – A high radiation condition on Vent WRGMs will not cause any automatic response that would lead to an automatic Reactor scram. Plausible because this could be indicative of a condition that would lead operators to an eventual manual Reactor scram.
C. Incorrect – A high radiation condition on Stack WRGMs will not cause any automatic response that would lead to an automatic Reactor scram. Plausible because this could be indicative of a condition that would lead operators to an eventual manual Reactor scram.
D. Incorrect – A high radiation condition on Containment Atmosphere radiation monitors will not cause any automatic response that would lead to an automatic Reactor scram. Plausible because this could be indicative of a condition that would lead operators to an eventual manual Reactor scram.

Technical Reference(s): ARP 851326, N2-SOP-42

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A3.04
	Importance Rating	3.9

INTERMEDIATE RANGE MONITOR SYSTEM

Ability to monitor automatic operation of the INTERMEDIATE RANGE MONITOR SYSTEM including: Control rod block signals

Proposed Question: #35

A plant startup is in progress with the following:

- The Reactor Mode Switch is in START / HOT STBY.
- All SRMs are fully inserted and reading between 5 x 10E4 and 9 x 10E4 cps.
- All IRMs are fully inserted and mid-scale on range 1.

Then, all the SRMs and the IRMs inadvertently begin to withdraw.

Which one of the following will ***first*** initiate a rod block?

A. IRM Downscale

B. SRM Downscale

C. IRM Detector NOT Fully Inserted

D. SRM Detector NOT Fully Inserted

Proposed Answer: C

Explanation: When the withdraw function is selected the first action will be the movement of the SRM and IRM detectors from the core. Immediately the IRMs will be detected to be not fully inserted, which will immediately cause a rod block. This rod block is active whenever the mode switch is not in run.

A. Incorrect – The IRM Downscale rod block is bypassed with IRMs on Range 1. Plausible because IRM indication will lower and eventually be downscale. This would cause a rod block on other ranges.
B. Incorrect – The IRM Detector NOT Fully Inserted rod block will cause a rod block immediately. Plausible because the SRMs will move downscale, but this will take time as the detectors withdraw.
D. Incorrect – The IRM Detector NOT Fully Inserted rod block will cause a rod block immediately. Plausible because SRM detectors not being fully inserted does result in a rod block once the detectors also read < 100 cps, but this will take time as the detectors withdraw.

Technical Reference(s): N2-OP-96 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: Bank - 2015 Cert #8

Question History: 2010 NRC #10

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A4.08
	Importance Rating	3.4

SOURCE RANGE MONITOR SYSTEM

Ability to manually operate and/or monitor the SOURCE RANGE MONITOR SYSTEM in the control room: SRMS channel bypass

Proposed Question: #36

A plant startup is in progress with the following:

- The Reactor is critical with a positive period.
- The non-coincidence shorting links are installed.
- SRM A is fully inserted and indicating 7 x 10^4 cps.
- SRM B is INOP and bypassed.
- SRM C is partially withdrawn and indicating 2 x 10^3 cps.
- SRM D is partially withdrawn and indicating 4 x 10^3 cps.
- All IRMs are indicating at the low end of Range 2.

Then, SRM A fails to withdraw.

Which one of the following describes:

(1) the impact of this failure if Reactor power continues to rise

and

(2) the ability to bypass SRM A while SRM B remains bypassed,

in accordance with N2-OP-92, Neutron Monitoring?

(1) SRM A will cause a…

(2) While SRM B remains bypassed, SRM A…

A. (1) rod block, only
(2) can be bypassed

B. (1) rod block, only
(2) CANNOT be bypassed

C. (1) rod block first, and then a full scram
(2) can be bypassed

D. (1) rod block first, and then a full scram
(2) CANNOT be bypassed

Proposed Answer: B

Explanation: With IRMs on less than range 8, the SRM upscale rod out block is enabled. With the non-coincidence shorting links installed, the SRM scram function is bypassed. When SRM A exceeds 1 x 10^5 cps, a rod out block is generated. Only 1 SRM can be physically bypassed at a time. Therefore, with SRM B currently INOP and bypassed, SRM A cannot be bypassed.

A. Incorrect – SRM A cannot be bypassed because SRM B is already bypassed. Plausible that they could both be bypassed at the same time because A and B equipment are generally in different divisions.
C. Incorrect – A full scram does not occur because the non-coincident shorting links are installed. Plausible if these being installed meant the scram function was enabled, not bypassed. SRM A cannot be bypassed because SRM B is already bypassed. Plausible that they could both be bypassed at the same time because A and B equipment are generally in different divisions.
D. Incorrect – A full scram does not occur because the non-coincident shorting links are installed. Plausible if these being installed meant the scram function was enabled, not bypassed.

Technical Reference(s): N2-OP-92

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-5

Question Source: Bank – 2017 Cert #16

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 10/11/23 – Changed to the bank, parent question, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K6.08
	Importance Rating	3.5

AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR: Uninterruptable power supply (AC/DC)

Proposed Question: #37

The plant is operating at 100% power when power from 2VBB-UPS3A is lost due to a sustained electrical fault.

Which one of the following describes the impact of this UPS loss on the APRMs?

A. APRM Channels 1 and 2 de-energize

B. APRM Channels 3 and 4 de-energize

C. APRM Channels 1 and 2 lose a power source, but remain energized

D. APRM Channels 3 and 4 lose a power source, but remain energized

Proposed Answer: C

Explanation: APRM channels are powered from Quadruple Low Voltage Power Supplies with redundant supply sources. Loss of 2VBB-UPS3A removes one of two inputs of power to QLVPS for APRMs 1 and 2, but APRM channels all remain energized.

A. Incorrect – All APRM channels remain energized. Plausible because 2VBB-UPS3A does supply power to the APRMs.
B. Incorrect – All APRM channels remain energized. Plausible because 2VBB-UPS3A does supply power to the APRMs.
D. Incorrect – APRM Channels 3 and 4 do not lose a power source. Plausible because this would be correct for a loss of 2VBB-UPS3B or if the power supplies were aligned differently.

Technical Reference(s): N2-215003-RBO-8

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-8

Question Source: Bank - 2010 Cert #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K3.06
	Importance Rating	3.2

REACTOR CORE ISOLATION COOLING SYSTEM

Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM will have on the following systems or system parameters: Condensate storage tank level

Proposed Question: #38

The plant is operating at 100% power with the following:

- RCIC is in a normal standby lineup.
- A significant leak develops on the RCIC suction piping.
- Attempts to isolate the leak have failed.

Which one of the following describes the effect of this leak on Condensate Storage Tank (CST) level if it is NOT stopped?

Level lowers below 135,000 gallons in…

A. CST A only.

B. CST B only.

C. both CST A and B.

D. neither CST A nor B.

Proposed Answer: A

Explanation: RCIC is normally lineup up with its suction to the Condensate Storage Tanks. These tanks are cross-connected such that their levels lower at the same time at normal tank levels. However, the cross-connect is located such that the lower 135,000 gallons of CST A is reserved for RCIC only. The lower 135,000 gallons of CST B is reserved for HPCS only. Since the leak is from RCIC, level in CST B will NOT lower below 135,000 gallons but the level in CST A will lower below 135,000 gallons.

B. Incorrect – Level in CST A will lower below 135,000 gallons. Level in CST B will NOT lower below 135,000 gallons. Plausible because this would be correct for a leak from HPCS.
C. Incorrect – Level in CST B will NOT lower below 135,000 gallons. Plausible because this would be correct for simultaneous leaks from HPCS and RCIC.
D. Incorrect – Level in CST A will lower below 135,000 gallons. Plausible because this would be correct for a leak from a system other than RCIC or HPCS.

Technical Reference(s): N2-OP-4, N2-OP-35, PID-35D

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-256001-RBO-8

Question Source: Bank – 4/23 NRC #39

Question History: 4/23 NRC #39

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K2.01
	Importance Rating	4.0

AUTOMATIC DEPRESSURIZATION SYSTEM

Knowledge of electrical power supplies to the following: ADS logic

Proposed Question: #39

Which one of the following describes:

- the power supply to the ADS Div I logic

  and

- the impact on ADS if the power supply to the ADS Div I logic de-energizes?

	Power Supply	Impact if Power Supply De-energizes
A.	125 VDC	ADS is still capable of opening 7 SRVs
B.	125 VDC	ADS is NOT still capable of opening 7 SRVs
C.	120 VAC	ADS is still capable of opening 7 SRVs
D.	120 VAC	ADS is NOT still capable of opening 7 SRVs

Proposed Answer: A

Explanation: Div I ADS logic is power by 125 VDC from BYS*PNL201A. If this power supply de-energizes, ADS is still capable of opening 7 SRVs. BYS*PNL201B continues to power Div II ADS logic and the SRV "B" solenoids, and this is enough to open all 7 ADS SRVs.

B. Incorrect – ADS is still capable of opening 7 SRVs. Plausible because half the ADS logic and half the ADS solenoids are de-energized.
C. Incorrect – Div I ADS logic is power by 125 VDC from BYS*PNL201A. Plausible because 120 VAC powers many logic circuits, such as RPS.
D. Incorrect – Div I ADS logic is power by 125 VDC from BYS*PNL201A. Plausible because 120 VAC powers many logic circuits, such as RPS. ADS is still capable of opening 7 SRVs. Plausible because half the ADS logic and half the ADS solenoids are de-energized.

Technical Reference(s): N2-OP-34-LINEUPS, N2-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K3.14
	Importance Rating	3.1

PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF will have on the following systems or system parameters: Recirculation system

Proposed Question: #40

The plant is operating at 100% power when multiple relay failures cause a spurious Group 2 isolation on Level 2 Reactor water level.

Given the following valves:

(1) 2RHS*SOV35A(B) and 2RHS*SOV36A(B), RHR Sample Line Isolation Valves
(2) 2RCS*SOV104 and 2RCS*SOV105, Recirculation System Sample Isolation Valves
(3) 2MSS*MOV111 and 2MSS*MOV112, Main Steam Line Drain Isolation Valves

Which one of the following identifies which of these valves automatically closes?

A. (1) only

B. (2) only

C. (3) only

D. (1), (2), and (3)

Proposed Answer: B

Explanation: A Group 2 isolation occurs on Level 2 Reactor water level (108.8”) and automatically closes 2RCS*SOV104 and 2RCS*SOV105, Recirculation System Sample Isolation Valves.

A. Incorrect – 2RHS*SOV35A(B) and 2RHS*SOV36A(B), RHR Sample Line Isolation Valves, do not close on this signal. Plausible because they do close on a Group 4 isolation (Level 3 Reactor water level (159.3”) or high Drywell pressure).
C. Incorrect – 2MSS*MOV111 and 2MSS*MOV112, Main Steam Line Drain Isolation Valves, do not close on this signal. Plausible because they do close on a Group 1 isolation signal (including Level 1 Reactor water level (17.8”)).
D. Incorrect – 2RHS*SOV35A(B) and 2RHS*SOV36A(B), RHR Sample Line Isolation Valves, do not close on this signal. Plausible because they do close on a Group 4 isolation (Level 3 Reactor water level (159.3”) or high Drywell pressure). 2MSS*MOV111 and 2MSS*MOV112, Main Steam Line Drain Isolation Valves, do not close on this signal. Plausible because they do close on a Group 1 isolation signal (including Level 1 Reactor water level (17.8”)).

Technical Reference(s): N2-OP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223002-RBO-5

Question Source: Modified Bank - JAF 17-2 NRC #41

Question History: JAF 17-2 NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 10/11/23 – Changed choice (3) from RCIC vacuum breakers to Main Steam Line Drain valves, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K3.14
	Importance Rating	3.0

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Recirculation system: Plant-Specific

Proposed Question: #41

The plant is operating at 100% power with the following:

- Multiple relay failures cause a spurious Group I isolation on Main Steam Line radiation.

Given the following sets of valves:

(1) 20MOV-82 and 20AOV-83 (DW Floor Drain Valves)
(2) 02-2AOV-39 and 02-2AOV-40 (RWR Sample Valves)
(3) 10SOV-263A and 10SOV-263B (RHR Sample Valves)

Which one of the following identifies which of these valves automatically close?

A. (1) only

B. (2) only

C. (3) only

D. (1), (2), and (3)

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A3.02
	Importance Rating	4.2

SAFETY RELIEF VALVES

Ability to monitor automatic operation of the SAFETY RELIEF VALVES including: SRV operation on high reactor pressure

Proposed Question: #41

The plant is operating at 100% power with the following:

- A spurious Main Turbine trip occurs.
- Reactor pressure reaches a high of 1120 psig during the transient.

Which one of the following describes the response of the Safety Relief Valves (SRVs)?

A. All SRVs remain closed.

B. Multiple SRVs open in relief mode. The setpoint for safety mode is NOT exceeded for any SRV.

C. Multiple SRVs open in safety mode. The setpoint for relief mode is NOT exceeded for any SRV.

D. Multiple SRVs open, with the setpoint for both relief and safety mode exceeded for some SRVs.

Proposed Answer: B

Explanation: On rising Reactor pressure, SRVs first receive an open signal in relief mode. The setpoints for relief mode range from 1103 psig to 1143 psig. SRVs then open in safety mode if Reactor pressure continues to rise. The setpoints for safety mode range from 1165 psig to 1205 psig. With Reactor pressure reaching a high of 1120 psig, multiple SRVs open in relief mode, but none exceed the setpoint for safety mode.

A. Incorrect – Multiple SRVs open in relief mode. Plausible because this would be correct if Reactor pressure stayed below 1103 psig.
C. Incorrect – The safety mode setpoint is not exceeded, but the relief mode setpoint is exceeded. Plausible if the functions were reversed.
D. Incorrect – Reactor pressure does not exceed the setpoint for safety mode for any SRV. Plausible because this would be correct if Reactor pressure exceeded 1165 psig.

Technical Reference(s): N2-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K1.04
	Importance Rating	3.9

REACTOR WATER LEVEL CONTROL SYSTEM

Knowledge of the physical connections and/or cause and effect relationships between the REACTOR WATER LEVEL CONTROL SYSTEM and the following systems: Reactor feedwater flow

Proposed Question: #42

A plant startup is in progress with the following:

- Feedwater pump A is operating.
- Digital Feedwater Level Control (DFWLCS) is operating fully in automatic.
- Reactor power is 35% and stable.
- Total Feedwater flow is 6.4 Mlbm/hr and stable.
- Total Steam flow is 6.4 Mlbm/hr and stable.

Which one of the following describes the current status of DFWLCS?

DFLWCS is operating in…

A. Low Power Mode with 2FWS-LV55A controlling

B. Low Power Mode with 2FWS-LV10A controlling

C. High Power Mode with 2FWS-LV55A controlling

D. High Power Mode with 2FWS-LV10A controlling

Proposed Answer: D

Explanation: With total Feedwater flow >20% (about 3.53 Mlbm/hr), DFWLCS has automatically transferred to High Power Mode. With total Steam flow >3.7 Mlbm/hr, DFWLCS has fully closed 2FWS-LV55A and has 2FWS-LV10A in control.

A. Incorrect – With total Feedwater flow >20% (about 3.53 Mlbm/hr), DFWLCS has automatically transferred to High Power Mode. Plausible because Low Power Mode would be in effect at a lower total Feedwater flow. With total Steam flow >3.7 Mlbm/hr, DFWLCS has fully closed 2FWS-LV55A and has 2FWS-LV10A in control. Plausible because 2FWS-LV55A would be in control at a lower total Steam flow.
B. Incorrect – With total Feedwater flow >20% (about 3.53 Mlbm/hr), DFWLCS has automatically transferred to High Power Mode. Plausible because Low Power Mode would be in effect at a lower total Feedwater flow.
C. Incorrect – With total Steam flow >3.7 Mlbm/hr, DFWLCS has fully closed 2FWS-LV55A and has 2FWS-LV10A in control. Plausible because 2FWS-LV55A would be in control at a lower total Steam flow.

Technical Reference(s): N2-OP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K6.05
	Importance Rating	3.8

REACTOR WATER LEVEL CONTROL SYSTEM

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor water level input

Proposed Question: #43

The plant is operating at 100% power with Digital Feedwater Level Control (DFWLCS) operating fully in automatic.

Then, the signal from narrow range Reactor water level transmitter 2ISC-PDT14A suddenly fails upscale.

Which one of the following describes the automatic plant response to this failure?

Feedwater flow will…

A. lower significantly, but Reactor water level will stay above Level 3.

B. lower significantly and Reactor water level will eventually drop below Level 3.

C. remain approximately constant because the Feedwater Level Control valves automatically lock up.

D. remain approximately constant because the Feedwater Level Control system automatically adjusts for the failed level transmitter.

Proposed Answer: D

Explanation: The Feedwater Level Control system uses a Selected Narrow Range Level as the primary feedback during operation in Low Power and High Power modes. The Selected Narrow Range Level is generated from three narrow range Reactor Water Level transmitters (2ISC-PDT14A/B/C). The narrow range signals are validated by an algorithm that compares the three narrow range level transmitter signals. With a sudden failure of the selected transmitter, the system automatically switches to a different level transmitter and Feedwater flow remains approximately constant.

A. Incorrect – Feedwater flow remains approximately constant. Plausible because Feedwater flow would lower if the failed transmitter remained selected.
B. Incorrect – Feedwater flow remains approximately constant. Plausible because Feedwater flow would lower if the failed transmitter remained selected.
C. Incorrect – Feedwater level control valves do not lock up in response to this malfunction. Plausible because they do lock up in response to other malfunctions.

Technical Reference(s): N2-OP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A1.07
	Importance Rating	3.0

STANDBY GAS TREATMENT SYSTEM

Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY GAS TREATMENT SYSTEM including: SGTS train temperature

Proposed Question: #44

The plant has experienced a loss of coolant accident with the following:

- Drywell pressure is 4 psig and rising slowly.
- GTS Train B is running.
- GTS Train A automatically started and was placed in standby with its control switch in AUTO.

Then, annunciator 871109, SBGTS TRAIN B SYSTEM TROUBLE, alarms. Computer point GTSTC13, GTS FLTR B ACT CHAR TEMP, is alarming above the hi-hi setpoint of 300°F.

Which one of the following describes the resulting GTS system status?

	Train A	Train B
A.	Standby	Running
B.	Standby	Tripped
C.	Running	Running
D.	Running	Tripped

Proposed Answer: C

Explanation: This alarm indicates a hi-hi adsorber temperature on Train B. With an initiation signal present (Drywell pressure > 1.68 psig) and the Train A control switch in AUTO, hi-hi adsorber temperature on Train B causes Train A to restart. This condition does not cause Train B to trip.

A. Incorrect – Train A automatically restarts. Plausible for Train A to remain in standby because manual action has been taken previously to place if in standby after an auto-start.
B. Incorrect – Train A automatically restarts. Plausible for Train A to remain in standby because manual action has been taken previously to place if in standby after an auto-start. Train B remains running. Plausible for Train B to trip since it is experiencing an adverse condition.
D. Incorrect – Train B remains running. Plausible for Train B to trip since it is experiencing an adverse condition.

Technical Reference(s): ARP 871109

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-11

Question Source: Bank – 2017 Cert #17

Question History: 2009 NRC #14

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K5.04
	Importance Rating	3.4

STANDBY GAS TREATMENT SYSTEM

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the STANDBY GAS TREATMENT SYSTEM: Secondary containment pressure changes

Proposed Question: #45

The plant is operating at 100% power with the following:

- Standby Gas Treatment (GTS) train A has been manually started and Reactor Building Ventilation has been secured due to turbulent atmospheric conditions.
- Then, a steam leak develops in the Reactor Building.
- Reactor Building Above/Below Refueling Floor Exhaust radiation remains below the alarm setpoint.
- Reactor Building differential pressure drops to -0.10" WG.

Which one of the following describes the response of 2GTS*PV5A, FAN RECIRC VALVE, and GTS fan B?

	2GTS*PV5A	GTS Fan B
A.	Throttles further closed	Automatically starts
B.	Throttles further closed	Remains in standby
C.	Throttles further open	Automatically starts
D.	Throttles further open	Remains in standby

Proposed Answer: B

Explanation: 2GTS*PV5A modulates to maintain Reactor Building differential pressure approximately -0.25” WG. With Reactor Building differential pressure at -0.10” WG, the valve throttles further closed (recirculates less air back to the building and exhausts more air from the building). GTS fan B remains in standby, despite the low differential pressure, because it does not have another start signal (high rad, low flow, high Drywell pressure, or low Reactor water level).

A. Incorrect – GTS fan B remains in standby, despite the low differential pressure, because it does not have another start signal (high rad, low flow, high Drywell pressure, or low Reactor water level). Plausible because GTS fan B would automatically start on low differential pressure if it had previously been shutdown with an auto start signal present.
C. Incorrect – 2GTS*PV5A throttles further closed. Plausible because this would be correct if it was in-line with the filter train, rather than a recirc valve. GTS fan B remains in standby, despite the low differential pressure, because it does not have another start signal (high rad, low flow, high Drywell pressure, or low Reactor water level). Plausible because GTS fan B would automatically start on low differential pressure if it had previously been shutdown with an auto start signal present.
D. Incorrect – 2GTS*PV5A throttles further closed. Plausible because this would be correct if it was in-line with the filter train, rather than a recirc valve.

Technical Reference(s): N2-OP-61B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-5

Question Source: Modified Bank – 2017 NRC #19

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 1
K/A # 261000 A4.03
Importance Rating 3.0

SGTS

Ability to manually operate and/or monitor in the control room: Fan

Proposed Question: #19

The plant is operating at 100% power with the following:

- A steam leak develops in the Reactor Building
- Both trains of Standby Gas Treatment (GTS) start based on Reactor Building exhaust radiation levels.
- An Operator secures GTS fan A in accordance with N2-OP-61B, Standby Gas Treatment System, with the initiation signal still present.

3 minutes later, Reactor Building differential pressure drops to -0.10" WG.

Which one of the following describes the position of the GTS Train A initiation switch based on the Operator's actions, in accordance with N2-OP-61B, and the current status of GTS fan A?

	GTS Train A Initiation Switch Position		GTS Fan A
A.	Pull-to-lock		Secured
B.	Pull-to-lock		Running
C.	Auto After Stop		Secured
D.	Auto After Stop		Running

Proposed Answer: D

Explanation: N2-OP-61B section H.2.0 provides direction for securing a GTS train with an initiation signal still present. The GTS train control switch is placed and left in AUTO AFTER STOP. While this secures the fan initially, the fan automatically re-starts because Reactor Building differential pressure drops less than -0.25" WG.

A. Plausible – The control switch is in Auto After Stop. Plausible because many components would need their control switch in pull-to-lock to be secured with an initiation signal present. GTS fan A automatically restarts. Plausible because Reactor Building D/P is still maintained negative and the fan was stopped with an initiation signal present.
B. Plausible – The control switch is in Auto After Stop. Plausible because many components would need their control switch in pull-to-lock to be secured with an initiation signal present.
C. Plausible – GTS fan A automatically restarts. Plausible because Reactor Building D/P is still maintained negative and the fan was stopped with an initiation signal present.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A1.03
	Importance Rating	3.6

AC ELECTRICAL DISTRIBUTION

Ability to predict and/or monitor changes in parameters associated with operation of the AC ELECTRICAL DISTRIBUTION including: Bus voltage

Proposed Question: #46

The plant is operating at 100% power with the following:

- 2NPS-SWG001 and 2NPS-SWG003 are being supplied by Normal Station Service transformer 2STX-XNS1.
- At 2CEC*PNL852, the NORM STA SVCE XFMR LTC AUTO-MANUAL SELECT switch is in MAN.

Which one of the following describes the effect on 2NPS-SWG001 and 2NPS-SWG003 if the NORM STA SVCE XFMR LOAD TAP CHANGER control switch is taken to RAISE?

Bus…

A. voltage rises

B. voltage lowers

C. frequency rises

D. frequency lowers

Proposed Answer: A

Explanation: The given conditions show that the tap changer for Normal Station Service transformer 2STX-XNS1 is in Manual. Taking the tap changer control switch to RAISE adjusts the transformers taps such that voltage rises.

B. Incorrect – Voltage rises, not lowers. Plausible that the function of the tap changer would be reverse acting to the change in voltage.
C. Incorrect – Voltage changes, not frequency. Plausible that frequency would change, such as with changing a governor setting on an EDG.
D. Incorrect – Voltage changes, not frequency. Plausible that frequency would change, such as with changing a governor setting on an EDG.

Technical Reference(s): N2-OP-71A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Importance Rating	3.5

UNINTERRUPTABLE POWER SUPPLY (AC/DC)

Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks that provide for the following: Transfer of power supplies

Proposed Question: #47

Which one of the following identifies the response of Uninterruptible Power Supply 2VBA*UPS2C if the normal AC feed is lost?

2VBA*UPS2C …

A. shifts to 2BYS*SWG002A

B. shifts to 2NJS-US4 via 2VBB-TRS1

C. static switch automatically transfers to 2LAC*PNL100A

D. transfer switch 2VBA*TRS2A automatically shifts to 2VBA*UPS2A

Proposed Answer: A

Explanation: On loss of normal AC, UPS2C will transfer to DC via BYS*SWG002A.

B. Incorrect – UPS2C shifts to 2BYS*SWG002A. Plausible in that this would be correct if the question asked about UPS1B and not UPS2C.
C. Incorrect – UPS2C shifts to 2BYS*SWG002A. Plausible in that this would be correct if UPS 2C had an internal fault and not just a loss of normal AC input. On a UPS internal fault, the static switch would shift supply to the maintenance supply (2LAC*PNL100A) which would make this distracter correct.
D. Incorrect – UPS2C shifts to 2BYS*SWG002A. Plausible in that the candidate would have to know that TRS2A is a manually operated transfer switch and not an automatic switch like TRS1 which supplies UPS 1A/B/G.

Technical Reference(s): N2-OP-71D

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262002-RBO-8

Question Source: Bank – 2017 Cert #12

Question History: 2012 NRC #14

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 2.2.39
	Importance Rating	3.9

DC ELECTRICAL DISTRIBUTION

Knowledge of less than or equal to 1 hour technical specification action statements

Proposed Question: #48

The plant is operating at 100% power when 125 VDC panel 2CES*IPNL414 de-energizes due to a sustained electrical fault.

Which one of the following identifies a system that must be declared inoperable immediately, in accordance with Technical Specifications?

A. High Pressure Core Spray

B. Reactor Core Isolation Cooling

C. Control Room Envelope Filtration

D. Control Room Envelope Air Conditioning

Proposed Answer: A

Explanation: 125 VDC panel 2CES*IPNL414 supplies DC power for Division III. With this panel de-energized, Division III DC electrical power subsystem is inoperable and Technical Specification 3.8.4 Condition B requires declaring HPCS inoperable immediately.

B. Incorrect – HPCS must be declared inoperable immediately. Plausible because Technical Specification 3.5.1 Condition B requires verifying RCIC operable immediately in this situation.
C. Incorrect – HPCS must be declared inoperable immediately. Plausible because Technical Specification 3.3.8.2 Condition F.3.2 requires declaring a CREF subsystem inoperable immediately in other conditions.
D. Incorrect – HPCS must be declared inoperable immediately. Plausible because Technical Specification 3.3.8.2 Condition F.3.2 requires declaring a CREF subsystem inoperable immediately in other conditions and CRE AC is a related system covered by Technical Specification 3.7.3.

Technical Reference(s): N2-SOP-04, Technical Specification 3.8.4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/11/23 – Changed distractor from GTS to CRE AC, based on discussion with Ops and NRC.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 A4.07
	Importance Rating	3.4

EMERGENCY GENERATORS (DIESEL/JET)

Ability to manually operate and/or monitor the EMERGENCY GENERATORS (DIESEL/JET) in the control room: Transfer of control to local/remote panels

Proposed Question: #49

The Control Room has been evacuated due to a fire with the following:

- N2-SOP-78, Control Room Evacuation, is being executed.
- Offsite power has been lost.
- The Division I EDG has failed to start.
- The Unit Supervisor has directed locally starting the Division I EDG per N2-OP-100A Section H.13, Diesel Generator Local Start and Loading.
- The procedure requires the following actions:
  - **PLACE** CONTROL MODE switch in LOCAL
  - **DEPRESS** ENGINE CONTROL START pushbutton

Which one of the following identifies the location(s) where these actions are performed?

	PLACE CONTROL MODE switch in LOCAL	**DEPRESS** ENGINE CONTROL START pushbutton
A.	2CES*PNL405, Remote Shutdown Panel	2CES*PNL405, Remote Shutdown Panel
B.	2CES*PNL405, Remote Shutdown Panel	2CES*IPNL406, Engine Control Panel
C.	2CES*IPNL406, Engine Control Panel	2CES*PNL405, Remote Shutdown Panel
D.	2CES*IPNL406, Engine Control Panel	2CES*IPNL406, Engine Control Panel

Proposed Answer: D

Explanation: Both of the given manipulations are performed at 2CES*IPNL406.

A. Incorrect – Both of the given manipulations are performed at 2CES*IPNL406. Plausible because many of the local controls operated during a Control Room Evacuation are located at the Remote Shutdown Panel.
B. Incorrect – Both of the given manipulations are performed at 2CES*IPNL406. Plausible because many of the local controls operated during a Control Room Evacuation are located at the Remote Shutdown Panel.
C. Incorrect – Both of the given manipulations are performed at 2CES*IPNL406. Plausible because many of the local controls operated during a Control Room Evacuation are located at the Remote Shutdown Panel.

Technical Reference(s): N2-OP-100A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A1.01
	Importance Rating	3.0

COMPONENT COOLING WATER SYSTEM

Ability to predict and/or monitor changes in parameters associated with operation of the COMPONENT COOLING WATER SYSTEM including: CCW flow rate

Proposed Question: #50

The plant is operating at 100% power when the air supply to 2CCP-TV108, RB CLOSED LOOP COOLING TEMPERATURE CONTROL VALVE, is lost.

Which one of the following describes the effect of this malfunction on RBCLC cooling water temperature?

RBCLC cooling water temperature…

A. lowers because RBCLC flow through RBCLC heat exchangers rises.

B. rises because RBCLC flow through RBCLC heat exchangers lowers.

C. lowers because Service Water flow through RBCLC heat exchangers rises.

D. rises because Service Water flow through RBCLC heat exchangers lowers.

Proposed Answer: A

Explanation: On loss of air, 2CCP-TV108 re-positions to put more RBCLC flow through the heat exchanger. This causes RBCLC cooling water temperature to lower.

B. Incorrect – RBCLC flow through the HXs rises, not lowers. Plausible because this would be the correct answer if the valve failed in the opposite direction on a loss of air.
C. Incorrect – This valve adjusts RBCLC flow, not Service Water flow. Plausible because Service Water does flow through the other side of the heat exchangers and varying Service Water flow is another physically possible method of controlling RBCLC temperature.
D. Incorrect – This valve adjusts RBCLC flow, not Service Water flow. Plausible because Service Water does flow through the other side of the heat exchangers and varying Service Water flow is another physically possible method of controlling RBCLC temperature.

Technical Reference(s): N2-OP-13, PID-13C

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A2.03
	Importance Rating	3.8

COMPONENT COOLING WATER SYSTEM

Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High/low CCW temperature

Proposed Question: #51

The plant is operating at 100% power with the following:

- Annunciator 601246, REACTOR BLDG CLOSED LOOP COOLING SYS TROUBLE, alarms.
- Computer point CCPTC04, RBCLCW HX DISCH TEMP, is in alarm.
- CCP heat exchanger discharge temperature indicates 90ºF and stable.
- Service Water intake temperature is 74ºF and stable.
- Two CCP heat exchangers are currently in service.

Which one of the following indicates the status of CCP heat exchanger discharge temperature and the allowable control of CCP heat exchangers, in accordance with N2-OP-13, Reactor Building Closed Loop Cooling System?

	Temperature	Allowable Control of Heat Exchangers
A.	Low	A heat exchanger may be taken out of service
B.	Low	A heat exchanger CANNOT be taken out of service
C.	High	An additional heat exchanger may be placed in service
D.	High	An additional heat exchanger CANNOT be placed in service

Proposed Answer: C

Explanation: Normal CCP heat exchanger discharge temperature is ~86°F. The given temperature is high. N2-OP-13 allows three heat exchangers to be in service when Service Water intake temperature is above 72°F.

A. Incorrect – The given temperature is high. Plausible because it is only slightly high and is well below the upper end of the corresponding indication (130°F).
B. Incorrect – The given temperature is high. Plausible because it is only slightly high and is well below the upper end of the corresponding indication (130°F).
D. Incorrect – An additional heat exchanger may be placed in service. Plausible because normally only two heat exchangers are in service and the allowance to place three in service is dependent on Service Water temperature.

Technical Reference(s): ARP 601246, N2-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	510000 A2.06
	Importance Rating	3.3

SERVICE WATER SYSTEM

Ability to (a) predict the impacts of the following on the SERVICE WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Discharge strainer failure

Proposed Question: #52

The plant is operating at 100% power with the following:

- Service Water pumps A, B, C, D, and F are running.
- Automatic backwash of Service Water discharge strainer D has failed to operate properly.
- Service Water discharge strainer D D/P is 15 psid and rising.

Which one of the following identifies:

- the impact of this malfunction on Service Water pump flows

  and

- how to place Service Water discharge strainer D in continuous backwash,

in accordance with N2-OP-11, Service Water System?

	Flow will change for…	The strainer is placed in continuous backwash…
A.	Service Water pump D, only	from Control Room panel 601
B.	Service Water pump D, only	at Service Water pump bay panel 505
C.	all running Service Water pumps	from Control Room panel 601
D.	all running Service Water pumps	at Service Water pump bay panel 505

Proposed Answer: D

Explanation: The discharges of Service Water pumps B, D, and F are cross-connected prior to the discharge strainers. Therefore, as discharge strainer D D/P rises, flow is affected for all three of these pumps. Flow will also change some on Service Water pumps A and C in response to the changes on pumps B, D, and F. N2-OP-11 provides direction for place the discharge strainer in continuous backwash in Section H.2.0. This operation is performed at Service Water pump bay panel 505.

A. Incorrect – Flow is affected for the other running Service Water pumps also. Plausible that the discharge strainer would be located directly on the pump discharge before the cross-connect. The strainer is placed in continuous backwash at Service Water pump bay panel 505. Plausible because Service Water pumps and some Service Water valves are controlled from Control Room panel 601.
B. Incorrect – Flow is affected for the other running Service Water pumps also. Plausible that the discharge strainer would be located directly on the pump discharge before the cross-connect.
C. Incorrect – The strainer is placed in continuous backwash at Service Water pump bay panel 505. Plausible because Service Water pumps and some Service Water valves are controlled from Control Room panel 601.

Technical Reference(s): PID 11A, N2-OP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-276000-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 A1.10
	Importance Rating	2.9

RECIRCULATION SYSTEM

Ability to predict and/or monitor changes in parameters associated with operation of the RECIRCULATION SYSTEM including: Recirculation seal purge flows

Proposed Question: #53

Which one of the following identifies a malfunction that adversely impacts Recirculation pump seal purge flow rate?

A. CRD pump trip

B. WCS pump trip

C. Service Water pump trip

D. Condensate Transfer pump trip

Proposed Answer: A

Explanation: Seal purge is supplied to the Recirculation pumps from the CRD pumps. Therefore, trip of a CRD pump results in loss of seal purge flow.

B. Incorrect – Trip of a WCS pump would not adversely affect Recirculation pump seal purge flow. Plausible because CRD also supplies seal water to WCS.
C. Incorrect – Trip of a Service Water pump would not adversely affect Recirculation pump seal purge flow. Plausible because Service Water pumps do provide cooling water to various components.
D. Incorrect – Trip of a Condensate Transfer pump would not adversely affect Recirculation pump seal purge flow. Plausible because Condensate Transfer pumps do provide high quality water to various components.

Technical Reference(s): N2-OP-29, N2-SOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 10/11/23 – Changed distractor D from Circulating Water pump to Condensate Transfer pump, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 A4.10
	Importance Rating	4.0

RECIRCULATION FLOW CONTROL SYSTEM

Ability to manually operate and/or monitor the RECIRCULATION FLOW CONTROL SYSTEM in the control room: Flow control valve

Proposed Question: #54

The plant is operating at 100% power when annunciator 602105, RECIRC FCV A MOTION INHIBIT, alarms.

Which one of the following identifies the resulting ability to raise and lower flow with slow and fast detent with Recirc FCV A?

	Raise with Fast Detent	Raise with Slow Detent	Lower with Fast Detent	Lower with Slow Detent
A.	Blocked	Available	Available	Available
B.	Blocked	Blocked	Available	Available
C.	Blocked	Available	Blocked	Available
D.	Blocked	Blocked	Blocked	Blocked

Proposed Answer: D

Explanation: Recirc FCV Motion Inhibit functions by automatically closing valves to trap fluid in the FCV's hydraulic cylinder. This prevents motion both to raise and lower flow regardless of using the fast or slow detent features of the flow controller.

A. Incorrect – Lowering flow is also blocked. Plausible because this would be an analogous design feature to the control rod withdraw block if this were meant to limit only positive reactivity. Raising with slow detent is also blocked. Plausible because this would be a viable design feature if the purpose were to limit reactivity addition rate under certain circumstances.
B. Incorrect – Lowering flow is also blocked. Plausible because this would be an analogous design feature to the control rod withdraw block if this were meant to limit only positive reactivity.
C. Incorrect – Slow detent is also blocked. Plausible because this would be a viable design feature if the purpose were to limit reactivity change rate under certain circumstances.

Technical Reference(s): ARP 602105, N2-202002-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202002-RBO-5

Question Source: Bank – 2017 NRC #35

Question History: 2017 NRC #35

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	215002 K3.01
	Importance Rating	3.9

ROD BLOCK MONITOR SYSTEM

Knowledge of the effect that a loss or malfunction of the ROD BLOCK MONITOR SYSTEM will have on the following systems or system parameters: Reactor manual control system

Proposed Question: #55

A plant startup is in progress with the following:

- APRMs indicate 35%.
- A non-peripheral control rod is selected.

Then, a hardware problem results in an INOP trip on Rod Block Monitor (RBM) A.

Which one of the following describes the status of RBM A and control rod blocks?

RBM A is…

A. auto-bypassed. NO control rod blocks are enforced.

B. NOT auto-bypassed. NO control rod blocks are enforced.

C. NOT auto-bypassed. A control rod withdrawal block is enforced, only.

D. NOT auto-bypassed. Control rod withdrawal and insert blocks are enforced.

Proposed Answer: C

Explanation: The RBM is NOT auto-bypassed because APRMs indicate above the low power setpoint of 28%. The INOP trip of RBM A initiates a withdraw block, but not an insert block.

A. Incorrect – The RBM is NOT auto-bypassed because APRMs indicate above the low power setpoint of 28%. Plausible because this would be correct if power was lower.
B. Incorrect – A withdrawal block is enforced. Plausible because an insert block is NOT enforced. Also plausible because the INOP trip means the RBM is not capable of operating properly.
D. Incorrect – An insert block is NOT enforced. Plausible because both withdrawal and insert blocks are enforced with a failure of the RWM.

Technical Reference(s): N2-OP-92, ARP 603204

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: Bank – SSES LOC26R (2015) NRC #30

Question History: SSES LOC26R (2015) NRC #30

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 A2.03
	Importance Rating	4.3

PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Safety/relief valve leaking or stuck-open

Proposed Question: #56

The plant is operating at 100% power with the following:

- Annunciator 601537, ADS VALVE/SAFETY VALVES LEAKING, alarms.
- SRV 127 tailpipe temperature is 350°F and rising slowly.
- SRV 127 indicates closed on Panel 601.

Which one of the following describes the main Primary Containment parameter of concern for this issue and the threshold for entering an Emergency Operating Procedure based on that parameter?

	Parameter	Threshold
A.	Drywell pressure	0.75 psig
B.	Drywell pressure	1.68 psig
C.	Suppression Pool temperature	85°F
D.	Suppression Pool temperature	90°F

Proposed Answer: D

Explanation: The given indications show that SRV 127 is leaking, but not open. The leaking SRV is passing energy to the Suppression Pool water, therefore, the main Primary Containment parameter of concern is Suppression Pool temperature. The threshold for entering N2-EOP-PC on high Suppression Pool temperature is 90°F.

A. Incorrect – The main Primary Containment parameter of concern is Suppression Pool temperature. Plausible because the rising Suppression Pool temperature may lead to a slight rise in Drywell pressure eventually. Also plausible because Drywell pressure would be of primary concern if there was an SRV tailpipe crack. 0.75 psig is the high Drywell pressure alarm setpoint.
B. Incorrect – The main Primary Containment parameter of concern is Suppression Pool temperature. Plausible because the rising Suppression Pool temperature may lead to a slight rise in Drywell pressure eventually. Also plausible because Drywell pressure would be of primary concern if there was an SRV tailpipe crack. 1.68 psig is the correct value for EOP entry on high Drywell pressure.
C. Incorrect – The threshold for entering N2-EOP-PC on high Suppression Pool temperature is 90°F. Plausible because 85°F is the setpoint for annunciator 601560, SUPPRESSION POOL WATER TEMP HIGH.

Technical Reference(s): ARP 601537, N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 K6.10
	Importance Rating	2.9

RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE: Component cooling water systems

Proposed Question: #57

Which one of the following describes the effect a loss of Reactor Building Closed Loop Cooling Water (CCP) would have on RHR pumps operating in the Suppression Pool spray mode?

A. Pumps must be shutdown because of loss of cooling to pump seals.

B. Pumps must be shutdown because of loss of cooling to pump motors.

C. Pump operation can continue and is now limited by area room temperatures.

D. Pump operation can continue and cooling can be aligned to an alternate source.

Proposed Answer: D

Explanation: With loss of CCP, RHR pumps are affected by loss of cooling water to their seal coolers. Alternate cooling to the seal coolers can be aligned from Service Water. The pumps are allowed to continue operating during this evolution.

A. Incorrect – The pumps may continue to run while alternate seal cooling is established. Plausible because this does lead to loss of seal cooling and would lead to damage if the pump were run indefinitely without alternate seal cooling aligned.
B. Incorrect – The component of concern is the pump seals, not the motor, and the pumps may continue to run while alternate cooling is aligned. Plausible because operation would lead to damage if the pump were run indefinitely without alternate cooling.
C. Incorrect – Room cooling is provided by Service Water, not CCP. Plausible because on loss of Service Water, RHR pump operation is limited by area temperature (Max continuous ambient temperature = 148^{o}F).

Technical Reference(s): ARP 601255

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-8

Question Source: Bank - 2017 NRC #36

Question History: 2017 NRC #36

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K2.02
	Importance Rating	3.3

MAIN AND REHEAT STEAM SYSTEM

Knowledge of electrical power supplies to the following: Main steam line shutoff valves

Proposed Question: #58

Which one of the following describes the power supplies to the Main Steam Isolation Valve (MSIV) trip solenoids?

	Inboard MSIV Solenoids	Outboard MSIV Solenoids
A.	2VBS* panels	2VBS* panels
B.	2VBS* panels	2BYS* panels
C.	2BYS* panels	2VBS* panels
D.	2BYS* panels	2BYS* panels

Proposed Answer: A

Explanation: The inboard MSIV trip solenoids are powered from 120 VAC RPS buses 2VBS*PNLA(B)106. The outboard MSIV trip solenoids are powered by the 120 VAC RPS buses 2VBS*PNLA(B)105.

B. Incorrect – Both the inboard and outboard MSIV trip solenoids are powered by the 120 VAC RPS buses. Plausible because the power supplies are arranged differently. Also plausible because many valves do receive power or control power from 125 VDC sources such as 2BYS*PNL201A(B).
C. Incorrect – Both the inboard and outboard MSIV trip solenoids are powered by the 120 VAC RPS buses. Plausible because the power supplies are arranged differently. 2BYS*PNL201A(B)
D. Incorrect – Both the inboard and outboard MSIV trip solenoids are powered by the 120 VAC RPS buses. Plausible because the power supplies are arranged differently. 2BYS*PNL201A(B)

Technical Reference(s): N2-OP-1, N2-239001-RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-239001-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 10/13/23 – Revised answer choices from just a voltage to actual bus component IDs, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K1.11
	Importance Rating	4.1

REACTOR/TURBINE PRESSURE REGULATING SYSTEM

Knowledge of the physical connections and/or cause and effect relationships between the REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following systems: RPS

Proposed Question: #59

The plant is operating at 40% power with the following:

- Turbine Stop Valve (TSV) 1D drifts 50% closed over 5 seconds.
- Then, TSV 1C drifts 50% closed over the next 5 seconds.
- Then, TSV 1B drifts 50% closed over the next 5 seconds.

Considering only the Turbine Stop Valve Closure scram signal, which one of the following describes the resulting operation of the Reactor Protection System?

Based on the Turbine Stop Valve Closure scram signal, a full Reactor scram…

A. occurs while TSV 1D is drifting.

B. occurs while TSV 1C is drifting.

C. occurs while TSV 1B is drifting.

D. does NOT occur.

Proposed Answer: C

Explanation: TSVs provide an input to RPS when they are less than 95% open. The logic is arranged such that this signal:

- Does not cause any half scram if only 1 of the 4 TSVs is less than 95% open
- May cause a half scram if only 2 of the 4 TSVs are less than 95% open, depending on the combination, but will not cause a full scram, and
- Causes a full scram when 3 of the 4 TSVs are less than 95% open, regardless of the combination.

Therefore, when the 3rd TSV (1B) drifts closed, a full scram occurs.

A. Incorrect – A full scram does not occur while TSV 1D is drifting. Plausible because TSV 1D drifting does provide inputs to RPS and has the potential to affect Reactor pressure and power.
B. Incorrect – A full scram does not occur while TSV 1C is drifting. Plausible because with TSV 1C and 1D drifted less than 95% open, a half scram would occur.
D. Incorrect – A full scram occurs while TSV 1B is drifting. Plausible because initial Reactor power is low, the TSVs only drift partially closed, and one TSV remains full open.

Technical Reference(s): N2-212000-RBO-3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-248000-RBO-8

Question Source: Modified Bank – 4/23 NRC #34

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 10/13/23 – Added times to drifting TSVs to address concern over plausibility of distractors and added detail to explanation, based on NRC comment.
TRH 10/25/23 – Added “the next” to 2 bullets based on NRC comment.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 1
K/A # 212000 K6.08
Importance Rating 3.3

REACTOR PROTECTION SYSTEM

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR PROTECTION SYSTEM: Main turbine generator and auxiliaries systems

Proposed Question: #34

The plant is operating at 40% power with the following:

- Turbine Stop Valve (TSV) 1D drifts 50% closed.
- Then, TSV 1C drifts 50% closed.

Considering only the Turbine Stop Valve Closure scram signal, which one of the following describes the resulting operation of the Reactor Protection System?

Based on the Turbine Stop Valve Closure scram signal, a full Reactor scram…

A. occurs while TSV 1D is drifting.

B. occurs while TSV 1C is drifting.

C. does NOT occur because only these two TSVs have drifted.

D. does NOT occur because these two TSVs have NOT tripped their scram position switches.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 K5.08
	Importance Rating	3.1

MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Generator cooling

Proposed Question: #60

The plant is operating at 100% power with the following:

- Service Water supply temperature is 60ºF and stable.
- 2CCS-TV104, TB CLOSED LOOP COOLING TEMPERATURE CONTROL VALVE, is stuck and cannot move.
- Subsequently, Service Water supply temperature lowers by 5ºF.

Which one of the following describes the response of Main Turbine Lube Oil temperature and the Main Generator leads temperature?

	Main Turbine Lube Oil Temperature	Main Generator Leads Temperature
A.	Remains approximately the same	Remains approximately the same
B.	Remains approximately the same	Lowers
C.	Lowers	Remains approximately the same
D.	Lowers	Lowers

Proposed Answer: B

Explanation: Service Water supplies the cooling water to the TBCLC heat exchangers and TBCLC supplies the cooling water to the Main Turbine Lube Oil and Main Generator Leads coolers. With Service Water temperature lowering and 2CCS-TV104 remaining in the initial position, TBCLC supply temperature will lower. The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant. The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.

A. Incorrect – The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.
C. Incorrect – The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant. The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.
D. Incorrect – The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant.

Technical Reference(s): P&ID 14B and 14E

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-274000-RBO-8

Question Source: Bank – 2015 NRC #27

Question History: 2015 NRC #27

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 2.1.25
	Importance Rating	3.9

CONDENSATE SYSTEM

Ability to interpret reference materials, such as graphs, curves, and tables (reference potential)

Proposed Question: #61

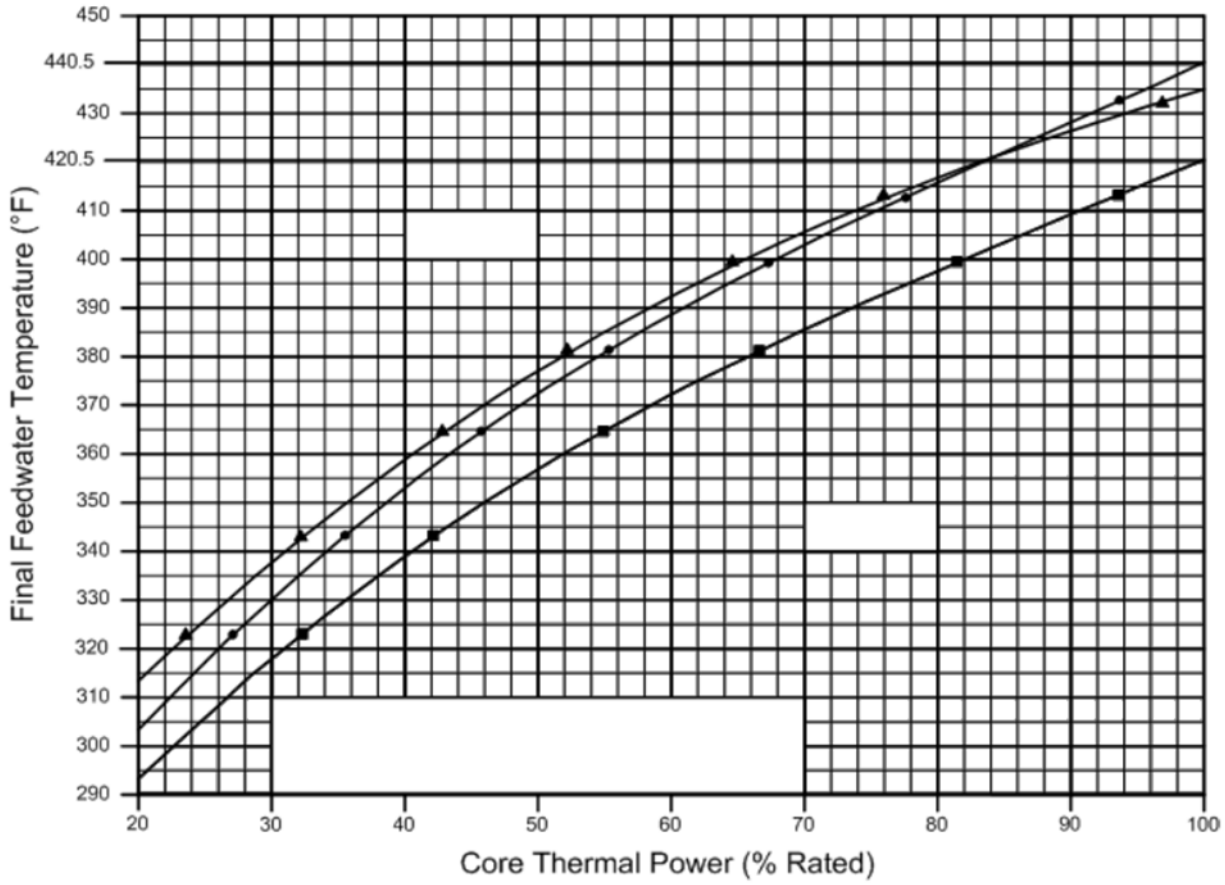
The plant is operating at 75% power with the following:

- 4th point Feedwater Heater A steam is removed per N2-OP-8, Feedwater Heaters and Extraction Steam System, due to a steam leak.
- Reactor power stabilizes at 78%.
- Feedwater temperature stabilizes at 405°F.
- N2-SOP-08, Unplanned Power Changes, has been entered due to the loss of Feedwater Heating.

Note: N2-SOP-08 Figure 1, Feedwater Temperature/Thermal Power Limit, is ***provided on the following page.***

Which one of the following describes the operating point on N2-SOP-08 Figure 1 and the required action for an unacceptable Feedwater temperature, in accordance with N2-SOP-08?

	The operating point on N2-SOP-08 Figure 1 is…	If the operating point is or becomes unacceptable, then it is required to perform a Reactor…
A.	acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
B.	acceptable.	scram per N2-SOP-101C, Reactor Scram.
C.	NOT acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
D.	NOT acceptable.	scram per N2-SOP-101C, Reactor Scram.


Final Feedwater Temperature (°F)
450
440.5
430
420.5
410
400
390
380
370
360
350
340
330
320
310
300
290
20
30
40
50
60
70
80
90
100
Core Thermal Power (% Rated)

Proposed Answer: A

Explanation: The given combination of Reactor power and Feedwater temperature is in the GOOD region of the curve (acceptable). If the operating point becomes unacceptable, then a Reactor power reduction is required per N2-SOP-101D, but NOT a Reactor scram.

B. Incorrect – A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation outside of the analyzed Feedwater temperature for the safety analysis.
C. Incorrect – Feedwater temperature is acceptable. Plausible if the operator mis-plots the operating point or does not understand which area of the curve is acceptable.
D. Incorrect – Feedwater temperature is acceptable. Plausible if the operator mis-plots the operating point or does not understand which area of the curve is acceptable. A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation outside of the analyzed Feedwater temperature for the safety analysis.

Technical Reference(s): N2-SOP-08

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 2101-SOP08C01 EO-2

Question Source: Bank – 2017 NRC #33

Question History: 2017 NRC #33

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/10/23 – Changed “Note” to “K/A Justification” based on NRC comment.
TRH 10/25/23 – Resampled K/A and replaced question based on NRC comment.
TRH 10/26/23 – Lowered temperature from 418 to 405 degrees based on NRC comment.

Proposed Question: #73

The plant is operating at 75% power with the following:

- 4th point Feedwater Heater A steam is removed per N2-OP-8, Feedwater Heaters and Extraction Steam System, due to a steam leak.
- Reactor power stabilizes at 78%.
- Feedwater temperature stabilizes at 390°F.
- N2-SOP-08, Unplanned Power Changes, has been entered due to the loss of Feedwater Heating.

Note: N2-SOP-08 Figure 1, Feedwater Temperature/Thermal Power Limit, is ***provided on the following page.***

Which one of the following describes the operating point on N2-SOP-08 Figure 1 and the required action for an unacceptable Feedwater temperature, in accordance with N2-SOP-08?

	The operating point on N2-SOP-08 Figure 1 is...		If the operating point is or becomes unacceptable, then it is required to perform a Reactor...
A.	acceptable.		power reduction per N2-SOP-101D, Rapid Power Reduction.
B.	acceptable.		scram per N2-SOP-101C, Reactor Scram.
C.	NOT acceptable.		power reduction per N2-SOP-101D, Rapid Power Reduction.
D.	NOT acceptable.		scram per N2-SOP-101C, Reactor Scram.

Proposed Answer: C

Explanation: The given combination of Reactor power and Feedwater temperature is in the BAD region of the curve (unacceptable). If the operating point is unacceptable, then a Reactor power reduction is required per N2-SOP-101D, but NOT a Reactor scram.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 K4.02
	Importance Rating	4.1

RADIATION MONITORING SYSTEM

Knowledge of RADIATION MONITORING SYSTEM design features and/or interlocks that provide for the following: System isolations/initiations

Proposed Question: #62

A plant startup is in progress with the following:

- A Mechanical Vacuum pump is in service.
- Reactor pressure is 150 psig and slowly rising.
- One Turbine Bypass Valve is partially open.

Then, the following alarms are received:

- Annunciator 603133, MN STEAM LINE RADIATION HIGH
- Annunciator 603407, DIV 2 MN STM LINES RADIATION HI-HI/INOP

Annunciator 603107, DIV 1 MN STM LINES RADIATION HI-HI/INOP, is **NOT** yet in alarm.

Main Steam Line radiation monitor indications are as follows:

Monitor	Trip(s) Indicated
A	HIGH
B	HIGH, HI-HI
C	HIGH
D	HIGH

Which one of the following describes the plant response to these indications?

A. NO automatic plant response has occurred yet.

B. The Mechanical Vacuum pump has tripped and isolated. The MSIVs remain open.

C. The Mechanical Vacuum pump remains in service and the MSIVs have closed.

D. The Mechanical Vacuum pump has tripped and isolated and the MSIVs have closed.

Proposed Answer: A

Explanation: Four radiation monitors (A, B, C, D) are provided on the Main Steam Lines. A and C input to Div I. B and D input to Div II. The trip logic requires at least one of the Div I inputs and at least one of the Div II inputs to be above the HI-HI setpoint to cause the associated MSL rad monitor trips. These trips include:

- Trip of the Air Removal System Mechanical Vacuum pump
- Isolation of the Mechanical Vacuum pump suction valve

The given MSL rad monitor indications show 3 monitors (A, C, and D) above the HIGH setpoint but below the HI-HI setpoint, and 1 monitor (B) above the HI-HI setpoint. With only one monitor above the HI-HI setpoint, the trip logic is NOT satisfied, therefore the Mechanical Vacuum pump has NOT tripped. The MSIVs do NOT go closed on high MSL rad level. This automatic isolation is part of original plant design, but has been bypassed at NMP2.

B. Incorrect – The MVP has not yet tripped because only a Div II rad monitor is above the HI-HI setpoint. Plausible that this would be enough to cause isolation based on the relatively low impact of MVP trip/isolation.
C. Incorrect – The MSIVs remain open. Plausible because original plant design includes MSIV isolation on HI-HI MSL radiation, and one rad monitor is above the HI-HI setpoint with the others also in alarm.
D. Incorrect – The MVP has not yet tripped because only a Div II rad monitor is above the HI-HI setpoint. Plausible that this would be enough to cause isolation based on the relatively low impact of MVP trip/isolation. The MSIVs remain open. Plausible because original plant design includes MSIV isolation on HI-HI MSL radiation, and one rad monitor is above the HI-HI setpoint with the others also in alarm.

Technical Reference(s): ARP 603107, ARP 603407, ARP 603133

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-5

Question Source: Bank – 2017 NRC #38

Question History: 2017 NRC #38

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 A4.13
	Importance Rating	3.6

SECONDARY CONTAINMENT

Ability to manually operate and/or monitor the SECONDARY CONTAINMENT in the control room: Secondary containment ventilation systems

Proposed Question: #63

The plant is operating at 100% power when a seismic event results in the following:

- The Reactor scrams.
- Reactor water level reaches a low of 140" during the scram transient.
- Reactor water level is now 190" and stable.
- Drywell pressure is 2 psig and slowly rising.
- One Reactor Building floor drain sump water level is in high-high alarm.
- Reactor Building Ventilation (HVR) exhaust radiation is normal.

Which one of the following describes the current status of HVR?

HVR is currently…

A. in service, with Standby Gas Treatment in standby.

B. in service, with Standby Gas Treatment fans also running.

C. isolated. HVR can be manually restarted without defeating an isolation.

D. isolated. HVR can be manually restarted, but only after defeating an isolation.

Proposed Answer: D

Explanation: HVR automatically isolates on Drywell pressure >1.68 psig. N2-EOP-SC and N2-EOP-6.26 provide guidance for restarting HVR, since the HVR exhaust radiation remains normal. With Drywell pressure >1.68 psig, an isolation must be defeated to restart HVR.

A. Incorrect – HVR is isolated because Drywell pressure is >1.68 psig. Plausible because Reactor water level remained above the isolation setpoint (108.8") and HVR exhaust radiation is normal.
B. Incorrect – HVR is isolated because Drywell pressure is >1.68 psig. Plausible because Reactor water level remained above the isolation setpoint (108.8") and HVR exhaust radiation is normal.
C. Incorrect – An isolation must be defeated to restart HVR. Plausible because this would be correct if HVR isolated on low Reactor water level and then Reactor water level was restored to the normal band, and no high Drywell pressure was present.

Technical Reference(s): N2-OP-52, N2-EOP-SC, N2-EOP-6.26

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-288000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.14
	Importance Rating	3.1

Knowledge of criteria or conditions that require plantwide announcements, such as pump starts, reactor trips, and mode changes

Proposed Question: #64

Given the following events:

(1) Re-energizing switchgear 2NPS-SWG001.
(2) An Injury in the plant requires Fire Brigade response.
(3) The Reactor reaches criticality during a startup.

Which one of the following identifies the events that must be announced to the plant, in accordance with OP-AA-104-101, Communications?

A. (1) and (2) only

B. (2) and (3) only

C. (1) and (3) only

D. (1), (2), and (3)

Proposed Answer: D

Explanation: Event (1) must be announced because it is energization of a major electrical switchgear. Event (2) must be announced because it requires alerting plant personnel to an injury in the plant. Event (3) must be announced for reaching criticality.

A. Incorrect – Event (3) must also be announced. Plausible because Event (3) is an expected condition during a Reactor startup.
B. Incorrect – Event (1) must also be announced. Plausible because event (1) is a normal evolution confined to a single location.
C. Incorrect – Event (2) must also be announced. Plausible if the candidate believes that event (2) only warrants use of radios to dispatch Fire Brigade.

Technical Reference(s): OP-AA-104-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - 2017 NRC #66

Question History: 2017 NRC #66

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.37
	Importance Rating	4.3

Knowledge of procedures, guidelines, or limitations associated with reactivity management

Proposed Question: #65

Which one of the following identifies the limitation on the timing of positive Reactivity additions, in accordance with N2-OP-101D, Power Changes?

Between positive reactivity additions, a minimum wait period of…

A. one (1) minute must be observed. This limitation applies at all Reactor power levels.

B. three (3) minutes must be observed. This limitation applies at all Reactor power levels.

C. one (1) minute must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

D. three (3) minutes must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

Proposed Answer: D

Explanation: N2-OP-101D contains the following requirement:

“When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.”

A. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude and N2-OP-101D discusses use of 1 minute core thermal power indication in a related reactivity management section regarding power maintenance. This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels.
B. Incorrect – This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels.
C. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude and N2-OP-101D discusses use of 1 minute core thermal power indication in a related reactivity management section regarding power maintenance.

Technical Reference(s): N2-OP-101D

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.0

Knowledge of the process for making changes to procedures

Proposed Question: #66

The plant is operating at 100% power with the following:

- It is desired to use a section of a surveillance test to perform post-maintenance testing (PMT) of a valve.
- Another section of the surveillance test CANNOT be performed as written due to an inoperable pump and does NOT specifically give the option to mark it “N/A”.

Which one of the following describes the ability to utilize this surveillance test for the desired PMT, in accordance with HU-AA-104-101, Procedure Use and Adherence?

The surveillance test…

A. CANNOT be used until the inoperable pump is fixed and the entire procedure can be performed as written.

B. can be used by having the performer initial the unnecessary steps as complete.

C. can be used by designating it for partial performance and marking the unnecessary section as N/A.

D. can be used, but only after revising it per AD-AA-101, Processing of Procedures, T&RMs, and Forms.

Proposed Answer: C

Explanation: HU-AA-104-101 section 4.4.3 allows use of N/A in three different circumstances. One of these circumstances is for partial performance. Partial performance is further outlined in section 4.6. The given situation fits the partial performance rules. With Job Supervisor authorization, the surveillance test can be designated for partial use and the steps that are not applicable can be marked N/A.

A. Incorrect – The surveillance test can be used in this way with the partial performance allowance. Plausible because there are significant restrictions on not performing a procedure entirely as written.
B. Incorrect – The unnecessary steps CANNOT be initialed as complete by the performer. Plausible because the partial use process does allow them to be marked N/A after this is authorized by the Job Supervisor.
D. Incorrect – The partial performance allowance provides a way to effectively change the procedure for the desired purpose without processing a formal revision. Plausible because AD-AA-101 could be used to revise the procedure, but it is not necessary.

Technical Reference(s): HU-AA-104-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/10/23 – Revised distractor A to raise plausibility, based on NRC comment.
TRH 10/25/23 – Combined 2nd and 3rd bullets based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms

Proposed Question: #67

The plant is operating at 100% power with the following:

- A nuisance Control Room annunciator has been received multiple times.
- It is desired to disable the faulty annunciator input per S-ARP-001, Disabling Nuisance Annunciators.
- It is determined that the annunciator will be disabled using a knife switch.

Which one of the following describes the proper tag to be used on this annunciator, in accordance with in accordance with S-ARP-001?

A. Caution Tag

B. Information Tag

C. Equipment Status Tag

D. Special Condition Tag

Proposed Answer: C

Explanation: S-ARP-001 provides guidance to eliminate the source of a nuisance alarm. For disabling an annunciator using a knife switch, an Equipment Status Tag is used.

A. Incorrect – An Equipment Status Tag is used for the annunciator. Plausible because Caution Tags are commonly used on the Control Room boards for off-normal controls or indications.
B. Incorrect – An Equipment Status Tag is used for the annunciator. Plausible because Information Tags are commonly used on the Control Room boards for off-normal controls or indications.
D. Incorrect – An Equipment Status Tag is used for the annunciator. Plausible because a Special Condition Tag is another type of tag used at the facility for other processes.

Technical Reference(s): S-ARP-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 17-2 NRC #42

Question History: JAF 17-2 NRC #42

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.12
	Importance Rating	3.2

Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters

Proposed Question: #68

The plant is operating at 15% power with the following:

- Personnel are preparing to enter the Drywell to investigate a problem.
- Drywell oxygen concentration is 20.0%.

Which one of the following describes the ability to access the Drywell under the current conditions, in accordance with RP-AB-462, Access Controls During Drywell Power Entries?

A. Access is allowed to all elevations to the Drywell

B. Access is allowed to the lower Drywell elevations, only

C. Access is allowed to the upper Drywell elevations, only

D. NO access to the Drywell is allowed

Proposed Answer: B

Explanation: Reactor power is below the limit (40%) to allow access to the lower elevations of the Drywell and above the limits to allow access to the upper elevations of the Drywell. Drywell oxygen concentration is within the required range of 19.5-23.5% to allow personnel entry into the Drywell. Therefore, access is allowed to the lower elevations of the Drywell, only.

A. Incorrect – Access is allowed to the lower elevations of the Drywell, only. Plausible because this would be correct if Reactor power was below 5%.
C. Incorrect – Access is allowed to the lower elevations of the Drywell, only. Plausible because this would be correct if the restrictions on the upper and lower Drywell were reversed.
D. Incorrect – Access is allowed to the lower elevations of the Drywell, only. Plausible because this would be correct if Reactor power was above 40% or if oxygen concentration was out of the required band.

Technical Reference(s): RP-AB-462, RP-AB-462-100-F-05

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – NMP1 2020 NRC #71

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 10/11/23 – Revised to have oxygen concentration in band and B correct, based on NRC comment.
TRH 10/25/23 – Revised explanation slightly based on NRC comment.

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.4

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: #71

The plant is operating at power with the following:

- Personnel are preparing to enter the Drywell at power to investigate a problem.
- The personnel are going to need access to all Drywell elevations.
- The Drywell has been de-inerted.

Which one of the following describes the requirements of RP-AB-462, Access Controls During Drywell Power Entries?

The upper limit for Reactor power is ___(1)___ and the lower limit for oxygen concentration is ___(2)___.

	(1)	(2)
A.	5%	10%
B.	5%	19.5%
C.	40%	10%
D.	40%	19.5%

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.6
	Importance Rating	3.7

Knowledge of emergency and abnormal operating procedures major action categories

Proposed Question: #69

The plant has experienced a failure to scram with the following:

- Multiple control rods remain fully withdrawn.
- Reactor power is 8% and stable.
- Reactor water level is 180” and stable with Feedwater injecting.
- Reactor pressure is 920 psig and stable on Turbine Bypass Valves.

Which one of the following describes the need to intentionally lower Reactor water level and to inject boron, in accordance with the Emergency Operating Procedures?

	Intentionally Lowering Reactor Water Level	Injecting Boron
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: A

Explanation: N2-EOP-RPV entry is required because Reactor power is above 4% when a scram is required. With multiple control rods fully withdrawn and the Reactor not shutdown, N2-EOP-C5 must be entered. With Reactor power above 4%, N2-EOP-C5 requires executing Detail J, High Power ATWS Actions. With Reactor water level above 100", Detail J requires intentionally lowering Reactor water level. Detail J also requires injecting boron.

B. Incorrect – N2-EOP-C5 Detail J requires injecting boron. Plausible because the Power leg of N2-EOP-C5 alone would not require injecting boron.
C. Incorrect – N2-EOP-C5 Detail J requires intentionally lowering Reactor water level. Plausible because Reactor water level would not need to be intentionally lowered if Reactor power was <4% or if Reactor water level was <100".
D. Incorrect – N2-EOP-C5 Detail J requires intentionally lowering Reactor water level. Plausible because Reactor water level would not need to be intentionally lowered if Reactor power was <4% or if Reactor water level was <100". N2-EOP-C5 Detail J requires injecting boron. Plausible because the Power leg of N2-EOP-C5 alone would not require injecting boron.

Technical Reference(s): N2-EOP-RPV, N2-EOP-C5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292008 K1.02
	Importance Rating	3.8

Reactor Operational Physics

List reactivity control mechanisms that exist for plant conditions during the approach to criticality

Proposed Question: #70

A plant startup is in progress with the following:

- The Reactor is subcritical.
- Control rods are being withdrawn.
- The current control rod is in RWM Group 2.

Which one of the following identifies:

(1) the milestone SRM count rate that must be monitored for

and

(2) the significance of reaching this count rate,

in accordance with N2-OP-101A, Plant Startup?

	Milestone SRM Count Rate	Significance of Reaching This Count Rate
A.	2x initial count rate	Requires notch control rod withdrawal from 00 to 36
B.	2x initial count rate	Indicates the Reactor is critical
C.	8x initial count rate	Requires notch control rod withdrawal from 00 to 36
D.	8x initial count rate	Indicates the Reactor is critical

Proposed Answer: C

Explanation: N2-OP-101A requires monitoring for 3 SRM count rate doublings. This is equivalent to SRMs indicating 8x initial count rate (1 doubling = 2x initial count rate, 2 doublings = 4x initial count rate, 3 doublings = 8x initial count rate). When this milestone is reached, the procedure requires single notching of control rods from position 00 to 36.

A. Incorrect – The milestone is 8x initial count rate. Plausible because this is equivalent to 1 count rate doubling.
B. Incorrect – The milestone is 8x initial count rate. Plausible because this is equivalent to 1 count rate doubling. The significance is the need to use single notch rod withdrawal. Plausible because the Reactor is subcritical and it does indicate that criticality is being approached.
D. Incorrect – The significance is the need to use single notch rod withdrawal. Plausible because the Reactor is subcritical and it does indicate that criticality is being approached.

Technical Reference(s): N2-OP-101A

Proposed references to be provided to applicants during examination: None

Learning Objective: BR08Ir4_Operational_Physics Objective 2

Question Source: Bank - Exelon GF Bank Reactor Theory #38

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292003 K1.05
	Importance Rating	3.7

REACTOR KINETICS AND NEUTRON SOURCES

Define reactor period

Proposed Question: #71

A plant startup is in progress with the following:

- Control rods are being withdrawn.
- Reactor period is being monitored per N2-OP-101A, Plant Startup.

Which one of the following Reactor periods would indicate the fastest rise in Reactor power?

A. ∞

B. +100

C. +20

D. -100

Proposed Answer: C

Explanation: Reactor period is defined as the length of time (in seconds) required for Reactor power to rise by a factor of e (2.71828…). A positive Reactor period corresponds to rising Reactor power. The smaller the positive number, the faster Reactor power is rising. Therefore, of the given choices, the +20 corresponds to the fastest rise in Reactor power.

A. Incorrect – +20 corresponds to the fastest rise in Reactor power. Plausible because infinity is the largest numerical value given.
B. Incorrect – +20 corresponds to the fastest rise in Reactor power. Plausible because +100 is a larger number than +20.
D. Incorrect – +20 corresponds to the fastest rise in Reactor power. Plausible because this would be correct if asking for the fastest drop in Reactor power.

Technical Reference(s): BR03Ir4_Kinetics, N2-OP-101A Section E.2.23-25

Proposed references to be provided to applicants during examination: None

Learning Objective: BR03Ir4_Kinetics Objective 5

Question Source: Bank – Exelon GF Bank Reactor Theory #9

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Comments:
TRH 10/11/23 – Revised to the bank, parent question, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

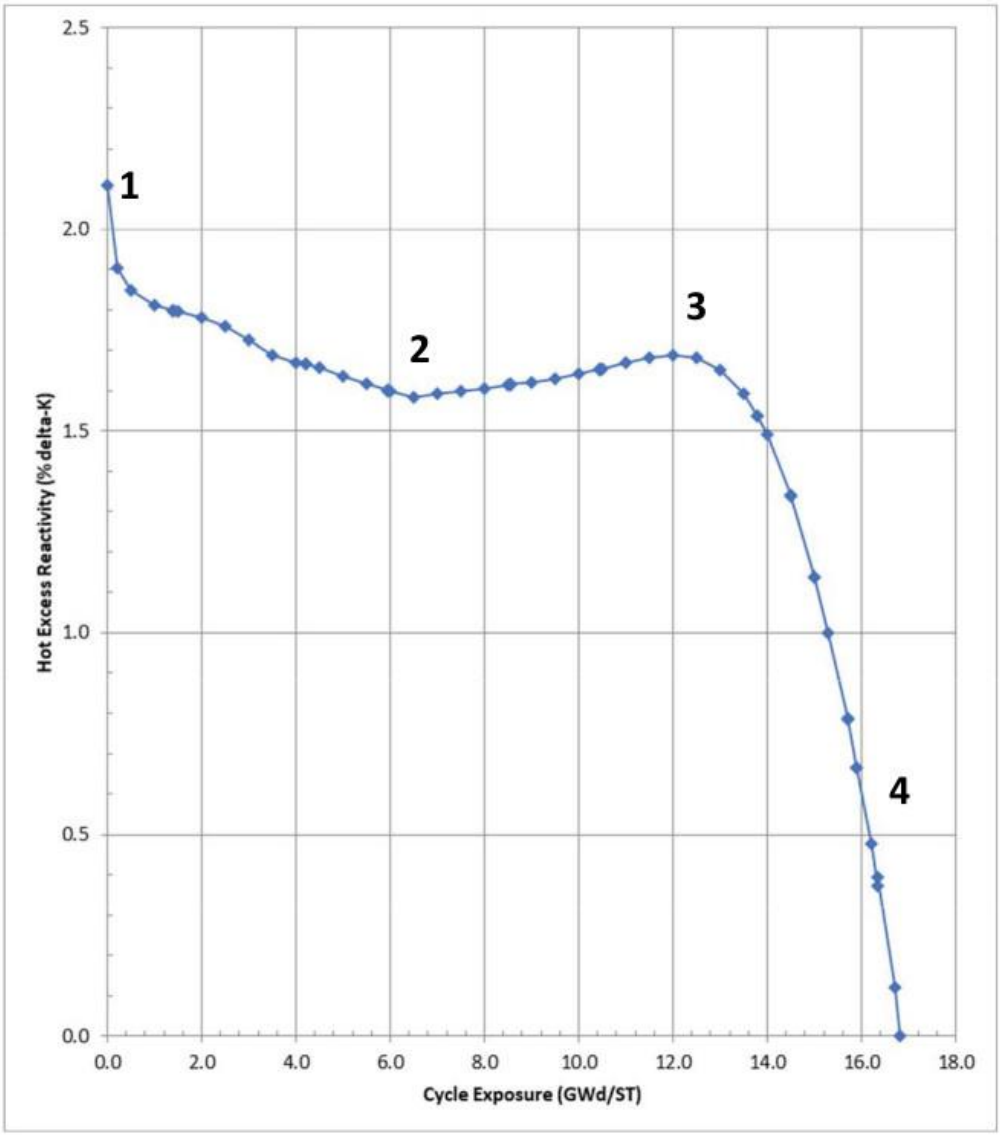
Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292007 K1.03
	Importance Rating	2.7

FUEL DEPLETION AND BURNABLE POISONS

Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points

Proposed Question: #72

Given the following drawing of K_{eff} versus core age from the Nine Mile Point Unit 2 Cycle Management report:



Which one of the following describes the major cause for the change in K_{eff} from point 2 to point 3?

A. Depletion of fuel

B. Depletion of control rods

C. Burnout of burnable poisons

D. Burnout of fission product poisons

Proposed Answer: C

Explanation: From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. The burnable poisons deplete faster than the fuel, raising K_{eff}.

A. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because depletion of fuel is the major cause for the change in K_{eff} from point 3 to 4.
B. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because the neutron absorbers in control rods do deplete over time.
D. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because the buildup of fission product poisons is the major cause for the change in K_{eff} from point 1 to 2. Also plausible because fission product poisons do burnout in some conditions.

Technical Reference(s): BR07Ir4_Fuel_Depletion, Cycle Management Report

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: BR07Ir4_Fuel_Depletion Objective 3

Question Source: Bank – NRC GFE Bank B1364

Question History: NRC GFE Bank B1364

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:
TRH 10/12/23 – Updated graphic based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293005 K1.05
	Importance Rating	2.8

THERMODYNAMIC CYCLES

State the advantages of moisture separators/reheaters and feedwater heaters for a typical steam cycle

Proposed Question: #73

The plant is operating at 100% power.

Which one of the following describes an advantage of having the Moisture Separators/Reheaters in service, in accordance with N2-OP-2, Moisture Separator Reheater System?

Lower the risk of damage to…

A. Feedwater pumps

B. Condensate pumps

C. low pressure Main Turbine blading

D. high pressure Main Turbine blading

Proposed Answer: C

Explanation: One advantage of having the Moisture Separators/Reheaters in service is that the moisture content of the steam entering the low pressure Main Turbine is lower. This lowers the risk of damage to the low pressure Main Turbine blading.

A. Incorrect – The advantage is lower risk of damage to the low pressure Main Turbine blading. Plausible because this is another component in the steam cycle and another thermodynamic concept, condensate depression, lowers the risk of damage to Condensate pumps.
B. Incorrect – The advantage is lower risk of damage to the low pressure Main Turbine blading. Plausible because this is another component in the steam cycle and another thermodynamic concept, condensate depression, lowers the risk of damage to Feedwater pumps.
D. Incorrect – The advantage is lower risk of damage to the low pressure Main Turbine blading. Plausible because this is a similar component. Also plausible because the Steam Separator / Steam Dryer in the Reactor vessel serve this purpose for the high pressure Main Turbine blading.

Technical Reference(s): BT05Ir4_Cycles, N2-OP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: BT05Ir4_Cycles Objective 7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293008 K1.11
	Importance Rating	2.8

THERMAL HYDRAULICS

POOL BOILING CURVE (TEMPERATURE VS. HEAT FLUX)

Describe transition (partial film) boiling

Proposed Question: #74

The plant is operating at 100% power with a MFLCPR of 0.80.

Which one of the following describes:

- the status of the MCPR thermal limit

  and

- what this indicates about the likelihood of transition boiling occurring in the Reactor?

	Status of MCPR	Transition boiling is…
A.	SAT	likely occurring
B.	SAT	NOT likely occurring
C.	UNSAT	likely occurring
D.	UNSAT	NOT likely occurring

Proposed Answer: B

Explanation: MFLCPR is <1, therefore the associated Minimum Critical Power Ratio (MCPR) is SAT.

A. Incorrect – Transition boiling is NOT likely occurring since the MCPR thermal limit is SAT. Plausible if transition boiling was a desired state that assisted in heat transfer from the fuel.
C. Incorrect – MFLCPR is <1, therefore the associated Minimum Critical Power Ratio (MCPR) is SAT. Plausible because MCPR is normally >1, but not its associated ratio (MFLCPR). Transition boiling is NOT likely occurring. Plausible because transition boiling would be a concern if MCPR was UNSAT.
D. Incorrect – MFLCPR is <1, therefore the associated Minimum Critical Power Ratio (MCPR) is SAT. Plausible because MCPR is normally >1, but not its associated ratio (MFLCPR).

Technical Reference(s): BT08Ir4_Thermal_Hydraulics, BT09Ir4_Thermal_Limits, N2-RESP-001, Technical Specification 3.2.2 bases

Proposed references to be provided to applicants during examination: None

Learning Objective: BT08Ir4_Thermal_Hydraulics 10.d

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	4
	Group #	
	K/A #	293010 K1.03
	Importance Rating	2.5

BRITTLE FRACTURE AND VESSEL THERMAL STRESS

Define reference temperature

Proposed Question: #75

A plant cooldown is in progress.

Which one of the following identifies:

- the Reactor coolant temperature at which brittle fracture is more likely

  and

- at what time in the Reactor's life brittle fracture is more likely?

Brittle fracture is more likely if Reactor coolant temperature is ___(1)___ the Reactor's nil-ductility reference temperature.

Brittle fracture is more likely ___(2)___ in the Reactor's life.

	(1)	(2)
A.	below	later
B.	above	later
C.	below	earlier
D.	above	earlier

Proposed Answer: A

Explanation: The nil-ductility reference temperature is the temperature below which a metal is susceptible to brittle fracture. Additionally, exposure to neutron flux embrittles a metal over the Reactor's life, which also makes it more susceptible to brittle fracture. Therefore, the Reactor vessel is most likely to experience brittle fracture when operating below its nil-ductility reference temperature and later in life.

B. Incorrect – Brittle fracture is more likely below the nil-ductility reference temperature. Plausible because higher temperatures can correspond to higher pressures, which place more stress on the Reactor vessel.
C. Incorrect – Brittle fracture is more likely as exposure to neutron flux increases. Plausible that the neutron flux would cause some other effect that lowered the chance of failure.
D. Incorrect – Brittle fracture is more likely below the nil-ductility reference temperature. Plausible because higher temperatures can correspond to higher pressures, which place more stress on the Reactor vessel. Brittle fracture is more likely as exposure to neutron flux increases. Plausible that the neutron flux would cause some other effect that lowered the chance of failure.

Technical Reference(s): BT10Ir4_Brittle_Stress, Technical Specification 3.4.11 bases

Proposed references to be provided to applicants during examination: None

Learning Objective: BT10Ir4_Brittle_Stress Objective 8.b

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.01
	Importance Rating	4.1

PARTIAL OR COMPLETE LOSS OF DC POWER

Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Partial or complete loss of DC power

Proposed Question: #76

The plant is operating at 100% power with the following:

- 125 VDC bus 2CES*IPNL414 de-energizes due to a sustained electrical fault.
- N2-SOP-04, Loss of DC Power, is being executed.
- Electrical Maintenance is determining how long it will be until 2CES*IPNL414 can be restored.

Which one of the following describes the guidance on control of the associated AC electrical bus, in accordance with N2-SOP-04?

A. Immediately commence de-energizing 2ENS*SWG102

B. Immediately commence de-energizing 2ENS*SWG103

C. If 2CES*IPNL414 will be out of service for >12 hours, then consider de-energizing 2ENS*SWG102

D. If 2CES*IPNL414 will be out of service for >12 hours, then consider de-energizing 2ENS*SWG103

Proposed Answer: C

Explanation: Given a loss of 2CES*IPNL414, the subsequent actions of N2-SOP-04 require a determination of how long the DC bus will be out of service. If the bus will be out of service for >12 hours, then the procedure contains guidance to consider de-energizing 2ENS*SWG102 due to operation without bus protection.

A. Incorrect – N2-SOP-04 only directs considering de-energizing the associated AC bus if 2CES*IPNL414 will be out of service for >12 hours. Plausible because the AC bus is currently in service without normal protection. The correct AC bus is 2ENS*SWG102, not 2ENS*SWG103. Plausible because 2ENS*SWG103 would be the associated AC bus for loss of another DC bus, 2BYS*SWG002B.
B. Incorrect – N2-SOP-04 only directs considering de-energizing the associated AC bus if will be out of service for >12 hours. Plausible because the AC bus is currently in service without normal protection.
D. Incorrect – The correct AC bus is 2ENS*SWG102, not 2ENS*SWG103. Plausible because 2ENS*SWG103 would be the associated AC bus for loss of another DC bus, 2BYS*SWG002B.

Technical Reference(s): N2-SOP-04

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP04C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295018 2.1.2
	Importance Rating	4.4

PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)

Knowledge of operator responsibilities during any mode of plant operation

Proposed Question: #77

The plant is operating at 100% power with the following:

- Annunciator 601246, REACTOR BLDG CLOSED LOOP COOLING SYS TROUBLE, alarms.
- CCP Surge Tank level is out of sight low.
- Annunciator 601232, RBCLCW PUMPS 3A/3B/3C AUTO TRIP/FAIL TO START, alarms.
- Annunciator 601234, RBCLCW PUMPS 3A/3B/3C DISCH PRESS LOW, alarms.
- Annunciator 601252, RBCLC PUMP 1A/1B/1C AUTO TRIP/FAIL TO START, alarms.
- All CCP pumps have tripped.
- All CCP booster pumps have tripped.

Which one of the following describes the required control of the Reactor and Reactor Recirc pumps, in accordance with N2-SOP-13, Loss or Degraded CCP System?

A. A power reduction is NOT required. Recirc pumps may remain operating.

B. Perform a rapid power reduction using Recirc. A Reactor scram is NOT required.

C. Trip one Reactor Recirc pump. Further power reduction is NOT required.

D. Perform a Reactor scram and trip both Reactor Recirc pumps.

Proposed Answer: D

Explanation: With all CCP pumps tripped, N2-SOP-13 requires scramming the Reactor per N2-SOP-101C and tripping both Reactor Recirc pumps.

SRO Justification: The question meets SRO level guidelines because it requires assessment of conditions and determining required transitions to other procedures (N2-SOP-101C, N2-SOP-101D, N2-OP-29).

A. Incorrect – With all CCP pumps tripped, N2-SOP-13 requires scramming the Reactor per N2-SOP-101C and tripping both Reactor Recirc pumps. Plausible because this would be correct if cooling capacity was degraded but one or more CCP pump was still running. Also plausible that just lower level actions would be required at this time, such as securing WCS.
B. Incorrect – With all CCP pumps tripped, N2-SOP-13 requires scramming the Reactor per N2-SOP-101C and tripping both Reactor Recirc pumps. Plausible because no high temperature alarms are yet given and this would lower heat load on the system and provide additional time to attempt to recover the CCP system.
C. Incorrect – With all CCP pumps tripped, N2-SOP-13 requires scramming the Reactor per N2-SOP-101C and tripping both Reactor Recirc pumps. Plausible because Recirc pumps are a major system load. This would lower heat load on the system and provide additional time to attempt to recover the CCP system.

Technical Reference(s): N2-SOP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP13C01 EO-2

Question Source: Bank - 2010 NRC #89

Question History: 2010 NRC #89

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 10/10/23 – Added SRO justification based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 2.1.27
	Importance Rating	4.0

PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

Knowledge of system purpose and/or function

Proposed Question: #78

The plant is operating at 100% power with the following:

- A significant Instrument Air leak has occurred.
- N2-SOP-19, Loss of Instrument Air, is being executed.
- 2IAS-PI194 (RB el. 261'), Inst Air Rcvr 2IAS-TK3 Pressure, indicates 76 psig and stable.
- 2IAS-PI101 (P851), Instrument Air Pressure, indicates 76 psig and stable.

Which one of the following describes the need for a Reactor scram and the operability of the outboard MSIVs based on the given air pressures, in accordance with N2-SOP-19?

	Reactor scram	Outboard MSIVs
A.	Required	Operable
B.	Required	Inoperable
C.	NOT required	Operable
D.	NOT required	Inoperable

Proposed Answer: C

Explanation: N2-SOP-19 includes the following steps:

IF	THEN
2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig,	Verify 2IAS-AOV171 auto closes AND dispatch an Operator to monitor 2RDS-PI133 & 2IAS-PI194 (RB el. 261').
2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	**SCRAM** the Reactor per N2-SOP-101C AND declare the OUTBOARD MSIV's inoperable.
2IAS-PI101 (P851), Instrument air pressure, lowers to < 70 psig,	**SCRAM** the Reactor per N2-SOP-101C.

With both of the given air pressures being above 74 psig, N2-SOP-19 does NOT require a Reactor scram and the outboard MSIVs remain operable.

A. Incorrect – A Reactor scram is not required. Plausible because air pressures are significantly degraded and this would be correct if air pressures were lower.
B. Incorrect – A Reactor scram is not required and outboard MSIVs remain operable. Plausible because air pressures are significantly degraded and this would be correct if air pressures were lower.
D. Incorrect – Outboard MSIVs remain operable. Plausible because air pressures are significantly degraded and this would be correct if air pressures were lower.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP19C01 EO-2

Question Source: Modified Bank - 2021 NRC #79

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 10/11/23 – Added parent question based on NRC comment.

Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure

Proposed Question: #79

The plant is operating at 100% power with the following:

- A significant Instrument Air leak has occurred.
- N2-SOP-19, Loss of Instrument Air, is being executed.
- 2IAS-PI194 (RB el. 261'), Inst Air Rcvr 2IAS-TK3 Pressure, indicates 68 psig and stable.
- 2IAS-PI101 (P851), Instrument Air Pressure, indicates 68 psig and stable.

Which one of the following describes the need for a Reactor scram and the operability of the outboard MSIVs based on the given air pressures, in accordance with N2-SOP-19?

	Reactor scram	Outboard MSIVs
A.	Required	Operable
B.	Required	Inoperable
C.	NOT required	Operable
D.	NOT required	Inoperable

Proposed Answer: B

Explanation: N2-SOP-19 includes the following condition steps:

IF	THEN
2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig,	Verify 2IAS-AOV171 auto closes AND dispatch an Operator to monitor 2RDS-PI133 & 2IAS-PI194 (RB el. 261').
2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	**SCRAM** the Reactor per N2-SOP-101C AND declare the OUTBOARD MSIV's inoperable.
2IAS-PI101 (P851), Instrument air pressure, lowers to < 70 psig,	**SCRAM** the Reactor per N2-SOP-101C.

Reactor Scram required ONLY if control rod(s) are withdrawn.

Due to air pressure on 2IAS-PI194 (RB el. 261') being below 74 psig, N2-SOP-19 requires both entering N2-SOP-101C to scram the Reactor and declaring the outboard MSIVs inoperable.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.02
	Importance Rating	3.8

REFUELING ACCIDENTS

Ability to determine or interpret the following as they apply to REFUELING ACCIDENTS: Fuel pool level

Proposed Question: #79

Core shuffle is in progress with the following:

- A leak develops in the Spent Fuel Pool (SFP).
- Annunciator 875111, SPENT FUEL POOL LEVEL HIGH/LOW, alarms.
- Annunciators 873317(875117), DIVISION I(II) SPENT FUEL POOL LEVEL LOW, alarm.
- Available SFP makeup has been maximized.
- All unnecessary personnel have been evacuated from the Refuel Floor and the Drywell.
- Secondary Containment has been established.
- SFP level is 328’ and lowering.

Which one of the following describes:

- whether the top of the fuel racks remains submerged

  and

- the need for SFP spray,

in accordance with N2-EOP-SC, Secondary Containment Control?

	Top of SFP Fuel Racks	SFP Spray
A.	Submerged	NOT required
B.	Submerged	Required
C.	Uncovered	NOT required
D.	Uncovered	Required

Proposed Answer: D

Explanation: With SFP level below 330', the top of the SFP fuel racks is uncovered. N2-EOP-SC requires beginning to line up SFP spray when SFP level cannot be maintained above 340'. When SFP level lowered below 330', SFP spray became required (regardless of RPV injection).

A. Incorrect – With SFP level below 330', the top of the SFP fuel racks is uncovered. Plausible because this would be correct if SFP level was higher. SFP spray is required. Plausible because this would be correct if SFP level was higher.
B. Incorrect – With SFP level below 330', the top of the SFP fuel racks is uncovered. Plausible because this would be correct if SFP level was higher.
C. Incorrect – SFP spray is required. Plausible because this would be correct if SFP level was higher. Also plausible because available SFP makeup sources are maximized and Secondary Containment is established.

Technical Reference(s): N2-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295024 EA2.04
	Importance Rating	4.3

HIGH DRYWELL PRESSURE

Ability to determine or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression chamber pressure

Proposed Question: #80

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is 20" and slowly rising.
- Reactor pressure is 500 psig and slowly lowering.
- Suppression Pool water temperature is 120°F and slowly rising.
- Suppression Pool water level is 202.5' and slowly rising.
- Drywell pressure is 18 psig and slowly lowering.
- Suppression Chamber pressure is 16 psig and slowly lowering.
- Drywell temperature is 220°F and slowly lowering.
- Suppression Chamber and Drywell sprays are in service using RHR pump RHS*P1A.
- RHR pump RHS*P1B is NOT available.

Then, RHR pump RHS*P1A trips.

- Drywell pressure is 19 psig and slowly rising.
- Suppression Chamber pressure is 17 psig and slowly rising.
- Drywell temperature is 225°F and slowly rising.

Note: Portions of select EOP curves are provided on the next page.

Which one of the following describes the required action, in accordance with N2-EOP-PC, Primary Containment Control?

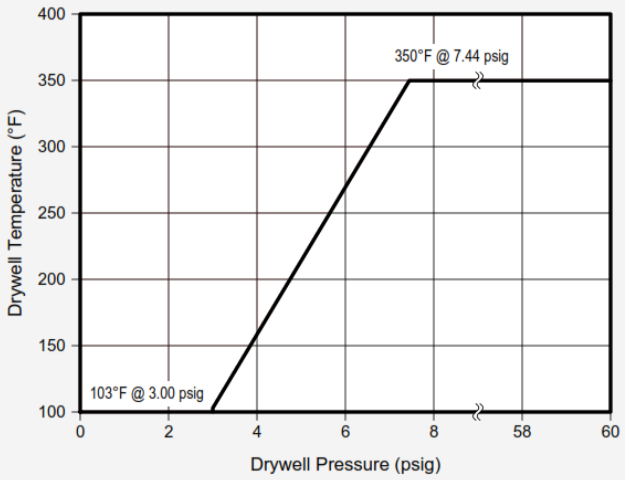
A. Vent the Drywell per N2-EOP-6.21, Containment Venting

B. Perform an RPV Blowdown per N2-EOP-C2, RPV Blowdown

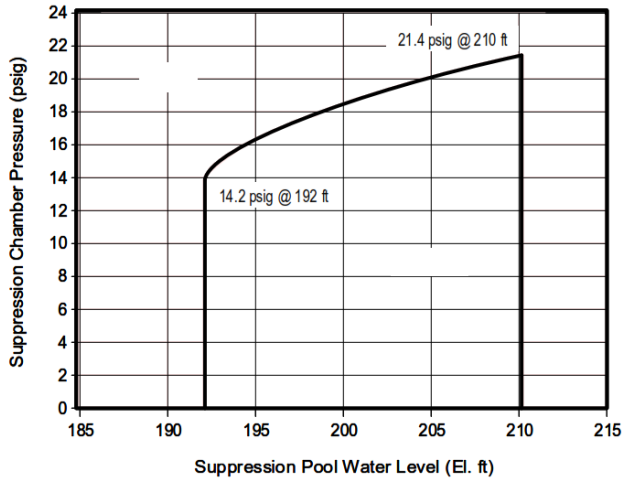
C. Vent the Suppression Chamber per N2-EOP-6.21, Containment Venting

D. Initiate alternate Containment Sprays using Service Water per N2-EOP-6.5

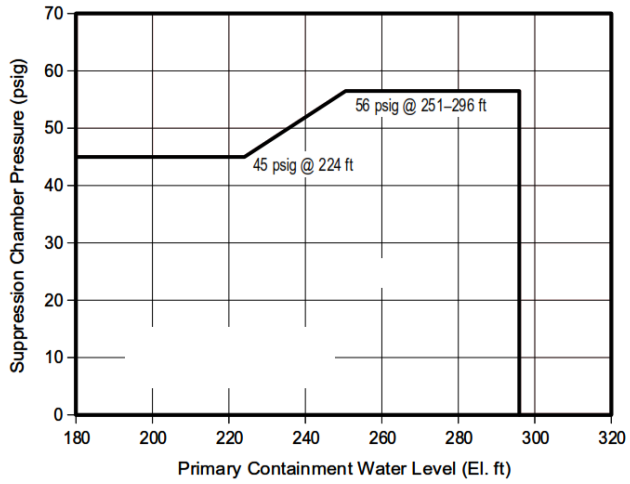
K Drywell Spray Initiation Limit



L Pressure Suppression Pressure



D Primary Containment Pressure Limit



Proposed Answer: D

Explanation: Suppression Chamber pressure above 10 psig and the combination of Drywell temperature and Suppression Chamber within the Drywell Spray Initiation Limit, Containment Sprays are required. With RHR pump RHS*P1A tripped and RHR pump RHS*P1B unavailable, normal Containment Sprays are unavailable. Suppression Chamber pressure is well within the Primary Containment Pressure Limit (PCPL), therefore N2-EOP-PC directs initiating alternate Containment Sprays using Service Water per N2-EOP-6.5. This lineup is still available even with the unavailability of RHR pumps.

A. Incorrect – Venting the Drywell per N2-EOP-6.21 is not required. Plausible because venting per N2-EOP-6.21 would be required if Suppression Chamber pressure was higher (violating PCPL). Also plausible because normal Containment Sprays are not available and Containment pressures are rising.
B. Incorrect – Performing an RPV Blowdown is not required. Plausible because this would be correct if Suppression Chamber pressure was higher (violating PSP). Also plausible because normal Containment Sprays are not available and Containment pressures are rising.
C. Incorrect – Venting the Suppression Chamber per N2-EOP-6.21 is not required. Plausible because venting per N2-EOP-6.21 would be required if Suppression Chamber pressure was higher (violating PCPL). Also plausible because normal Containment Sprays are not available and Containment pressures are rising.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: Embedded graphics

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – SSES LOC26R (2015) NRC #81

Question History: SSES LOC26R (2015) NRC #81

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295028 2.4.45
	Importance Rating	4.3

HIGH DRYWELL TEMPERATURE

Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question: #81

The plant is operating at 100% power with the following:

- Annunciator 601260, Drywell Unit Cooling Group 1 Sys Trouble, alarms.
- Computer points indicate that Drywell Unit Coolers 1A and 3A have tripped.
- Annunciator 601559, Primary Containment Temperature High, is in alarm.
- Drywell average temperature is 155ºF and stable.
- Drywell pressure is 0.2 psig and stable.
- Drywell Unit Coolers 1A and 3A CANNOT be restored.
- Additional Drywell Cooling is NOT available.

Which one of the following describes the required control of the plant, in accordance with plant procedures?

A. Immediately scram the Reactor per N2-SOP-101C, Reactor Scram

B. Perform a rapid power reduction per N2-SOP-101D, Rapid Power Reduction

C. Commence a normal plant shutdown per N2-OP-101C, Plant Shutdown

D. Continued Reactor operation is allowed at this power level indefinitely

Proposed Answer: C

Explanation: Annunciator 601260, along with trip of Drywell Unit Coolers 1A and 3A, requires entry into N2-SOP-60, Loss of Drywell Cooling. With annunciator 601559 in alarm and average Drywell temperature at 155°F, the limit of Technical Specification 3.6.1.5 (150°F) is being exceeded. Given that Drywell Cooling cannot be restored and Drywell temperature is exceeding the Technical Specification limit, N2-SOP-60 contains an override that requires commencing a normal plant shutdown. This is accomplished by using N2-OP-101C, Plant Shutdown.

A. Incorrect – An immediate Reactor scram is not required. Plausible because entry into N2-EOP-PC is required, and it would require a Reactor scram at higher Drywell temperatures.
B. Incorrect – A rapid power reduction is not required. Plausible because a power reduction is required as part of a normal plant shutdown. Also plausible because entry into N2-EOP-PC is required.
D. Incorrect – A normal plant shutdown is required. Plausible because Drywell pressure and temperature are stable, with only Drywell temperature slightly above the Technical Specification setpoint. This would be correct if Drywell average temperature was slightly lower.

Technical Reference(s): N2-SOP-60, Technical Specification 3.6.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP60C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 AA2.10
	Importance Rating	3.5

GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES

Ability to determine or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions

Proposed Question: #82

The plant is operating at 100% power with the following:

- Grid voltage fluctuations are occurring.
- Power Control has confirmed unstable grid conditions.
- Annunciator 851112, Generator Auxiliaries Trouble, alarms.
- Computer point GMCTC02, STTR CLG WTR OUTLET TEMP, is in alarm.
- Computer point GMCBC07, GEN PROT CKT ENERGIZED, is in alarm.
- Stator Water Cooling return temperature is 80°C and slowly rising.

Which one of the following describes the required control of the plant, in accordance with plant procedures?

A. Trip the Main Turbine per N2-SOP-21, Turbine Trip

B. Scram the Reactor per N2-SOP-101C, Reactor Scram

C. Perform a rapid power reduction per N2-SOP-101D, Rapid Power Reduction

D. Maintain Reactor power stable and slowly lower MVARs per N2-SOP-68, Generator Auxiliary Failures

Proposed Answer: C

Explanation: With Stator Water Cooling return temperature >79°C, a Main Generator runback is in progress. Entry into N2-SOP-68 is required. With a Main Generator runback in progress, an override in N2-SOP-68 directs performing a rapid power reduction per N2-SOP-101D.

A. Incorrect – A Main Turbine trip is not required. Plausible because N2-SOP-68 will require this if the runback does not clear during the power reduction.
B. Incorrect – A Reactor scram is not required. Plausible because N2-SOP-68 will require a Main Turbine trip if the runback does not clear during the power reduction, and a Reactor scram would also be required due to the Reactor power level.
D. Incorrect – A Reactor power reduction is required. Plausible because the override of N2-SOP-68 also requires lowering MVARs. Also plausible because Stator Water Cooling return temperature is barely above the runback setpoint, such that lowering MVARs might provide enough improvement to lower temperature.

Technical Reference(s): ARP 851112, N2-SOP-68

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP68C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295007 2.4.14
	Importance Rating	4.5

HIGH REACTOR PRESSURE

Knowledge of general guidelines for emergency and abnormal operating procedures usage

Proposed Question: #83

The plant is operating at 75% power with the following:

- The Reactor scrams due to a spurious Main Turbine trip.
- All control rods insert.
- Reactor pressure reaches a high of 1100 psig during the transient.
- Reactor water level reaches a low of 170" during the transient.

Which one of the following describes how to implement N2-EOP-RPV, RPV Control, and N2-SOP-21, Turbine Trip, in accordance with plant procedures?

A. The Pressure leg of N2-EOP-RPV must be executed before the Level leg. N2-SOP-21 is required to be entered.

B. The Pressure leg of N2-EOP-RPV must be executed before the Level leg. N2-SOP-21 is NOT required to be entered.

C. There is NO pre-determined order for executing the legs of N2-EOP-RPV. N2-SOP-21 is required to be entered.

D. There is NO pre-determined order for executing the legs of N2-EOP-RPV. N2-SOP-21 is NOT required to be entered.

Proposed Answer: C

Explanation: The given conditions show that the high Reactor pressure entry condition (>1052 psig) for N2-EOP-RPV is met and the low Reactor water level entry condition (<159.3”) is NOT met. When any EOP entry condition is met, the EOP is entered and all legs are entered concurrently. There is NO pre-determined order for executing the legs of N2-EOP-RPV. The entry conditions for N2-SOP-21 are met (Main Turbine trip). The procedure is required to be entered, despite also entering the higher-tiered EOPs.

A. Incorrect – There is NO pre-determined order for executing the legs of N2-EOP-RPV. Plausible because only the high Reactor pressure entry condition for N2-EOP-RPV is met.
B. Incorrect – There is NO pre-determined order for executing the legs of N2-EOP-RPV. Plausible because only the high Reactor pressure entry condition for N2-EOP-RPV is met. N2-SOP-21 is required to be entered. Plausible that entry would not be required since an EOP is entered and EOPs are higher-tiered documents than SOPs.
D. Incorrect – N2-SOP-21 is required to be entered. Plausible that entry would not be required since an EOP is entered and EOPs are higher-tiered documents than SOPs.

Technical Reference(s): N2-EOP-RPV, N2-SOP-21, N2-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2, 2101-SOP21C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 10/11/23 – Changed the first half of the answer choices, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295035 EA2.03
	Importance Rating	3.3

SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE

Ability to determine or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Lights and alarms

Proposed Question: #84

The plant is operating at 100% power with the following:

- An un-isolable steam leak develops from the RCIC system into the Reactor Building.
- The Reactor Building has isolated based on high airborne radiation levels.
- GTS Trains A and B are operating.
- Annunciator 870111, DIVISION I RX BLD TO ATM DIFF PRESSURE LOW, is in alarm.
- Annunciator 871111, DIVISION II RX BLD TO ATM DIFF PRESSURE LOW, is in alarm.
- Reactor Building differential pressure is +0.25" H2O and stable.
- NO field surveys have been completed yet for dose assessment (on-site or off-site).
- The Shift Manager has declared a Site Area Emergency.
- The Shift Manager is filling out the "Release of radioactive materials due to the classified event" section on the Part 1 Notification Fact Sheet.

Which one of the following identifies the letter to circle in this section, in accordance with EP-CE-114-100-F-05, Part 1 Notification Fact Sheet?

A. No radiological release in-progress

B. AIRBORNE radiological release in-progress

C. LIQUID radiological release in-progress

D. Radiological release TERMINATED

Proposed Answer: B

Explanation: The Reactor Building functions as the Secondary Containment, however it is NOT designed for zero leakage. Negative pressure is required to prevent an unmonitored release from the Reactor Building. However, the presence of annunciators 870111 and 871111 indicate proper negative pressure is not being maintained. With positive Reactor Building pressure and a RCIC steam leak into the building, air is leaking from the Reactor Building into the outside environment. This is an unmonitored release that requires additional evaluation (i.e. field surveys and dose assessment) in order to quantify. EP-AA-114-F-02 classifies this as "A Release in in Progress" in the Airborne flow chart.

A. Incorrect – An airborne release is in progress. Plausible because the Reactor Building is isolated and both trains of GTS are operating.
C. Incorrect – An airborne release is in progress. Plausible because the release is from a RCIC steam leak, which would result in condensed water (liquid) in the Reactor Building.
D. Incorrect – An airborne release is in progress. Plausible because the Reactor Building is isolated, which does terminate one unfiltered flow path.

Technical Reference(s): EP-CE-114-100-F-05, EP-AA-114-F-02

Proposed references to be provided to applicants during examination: EP-AA-114-F-02

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: Modified Bank – 2017 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 1
Group # 2
K/A # 295035 2.1.19
Importance Rating 3.8

Secondary Containment High Differential Pressure

Ability to use plant computers to evaluate system or component status.

Proposed Question: #85

The plant is operating at 100% power with the following:

- An un-isolable leak develops from the RCIC system into the Reactor Building.
- The Reactor Building has isolated based on high airborne radiation levels
- GTS Trains A and B are operating.
- The following page shows a computer screen image during the event.
- The Shift Manager has declared a Site Area Emergency.
- NO field surveys have been completed yet for dose assessment (on-site or off-site).
- The Shift Manager is filling out the following section of the NMP Notification Fact Sheet – Part 1:

5.	**Release of radioactive Materials due to the classified event:** **A.** No release **B.** Release **below** federal limits (ODCM), □ To atmosphere □ To Water **C.** Release **above** federal limits (ODCM), □ To atmosphere □ To Water **D.** Unmonitored release requiring evaluation. (*Reason documented in Section 7)*

For the above Part 1 Fact Sheet, which one of the following describes which letter to circle on Block 5?

A. A – No release

B. B – Release below federal limits (ODCM)

C. C – Release above federal limits (ODCM)

D. D – Unmonitored release requiring evaluation

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	500000 EA2.02
	Importance Rating	3.8

HIGH CONTAINMENT HYDROGEN CONCENTRATION

Ability to determine or interpret the following as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATION: Oxygen monitoring availability

Proposed Question: #85

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is 120" and lowering slowly.
- Drywell pressure is 7 psig and rising slowly.

Which one of the following describes the current status of the H2/O2 monitors and the required control of these monitors, in accordance with N2-EOP-PC, Primary Containment Control?

	Current Status of H2/O2 Monitors	Required Control
A.	In service	Bypass the isolation
B.	In service	Do NOT bypass the isolation
C.	Isolated	Maintain isolated
D.	Isolated	Bypass the isolation and place in service

Proposed Answer: D

Explanation: The H2/O2 monitors are currently isolated on a high Drywell pressure signal (>1.68 psig). N2-EOP-PC directs use of N2-OP-82 section F.1.0 to monitor hydrogen and oxygen concentrations. With a high Drywell pressure signal present, N2-OP-82 section F.1.0 directs use of N2-OP-82 section H.2.0 to bypass the isolation and then place the monitors in service.

A. Incorrect – The H2/O2 monitors are currently isolated. Plausible because Reactor water level is above the isolation setpoint (108.8"). Also plausible that the isolation would be bypassed to prevent isolation, since N2-EOP-PC directs their use.
B. Incorrect – The H2/O2 monitors are currently isolated. Plausible because Reactor water level is above the isolation setpoint (108.8"). Also plausible that the isolation wouldn't be bypassed because Reactor water level remains well above top of active fuel, such that fuel damage and hydrogen generation would not be likely.
C. Incorrect – The isolation is bypassed and the monitors are placed in service. Plausible because isolations are typically not bypassed and because Reactor water level remains well above top of active fuel, such that fuel damage and hydrogen generation would not be likely.

Technical Reference(s): N2-EOP-PC, N2-OP-82

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	209002 2.4.30
	Importance Rating	4.1

HIGH PRESSURE CORE SPRAY SYSTEM

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator

Proposed Question: #86

The plant is operating at 98% power with the following:

- HPCS has been inoperable for an extended period of time due to a major equipment malfunction.
- The completion time of Technical Specification 3.5.1 Condition B for the HPCS inoperability has expired.
- A plant shutdown is being commenced per Technical Specification 3.5.1 Condition C.

Which one of the following describes the most restrictive time requirement for notifying the NRC, in accordance with 10 CFR 50.72 and/or 10 CFR 50.73?

Notification must be made to the NRC as soon as practical and within a maximum time limit of…

A. 15 minutes

B. 1 hour

C. 4 hours

D. 60 days

Proposed Answer: C

Explanation: 10 CFR 50.72(b)(2)(i) requires notification of the NRC as soon as practical and, in all cases, within 4 hours of the initiation of any nuclear plant shutdown required by the plant's Technical Specifications. The given conditions meet this requirement and do not meet any of the more restrictive notification requirements, such as emergency declaration (1 hour) or deviation from Technical Specifications (1 hour).

A. Incorrect – The most restrictive requirement is to notify the NRC as soon as practical and within a maximum of 4 hours. Plausible because this is the time requirement for notifying the state and county upon entry into the Emergency Plan and for notifying the NRC of certain security events.
B. Incorrect – The most restrictive requirement is to notify the NRC as soon as practical and within a maximum of 4 hours. Plausible because this is the time requirement for notifying the NRC upon entry into the Emergency Plan of deviation from Technical Specifications per 10 CFR 50.54(x).
D. Incorrect – The most restrictive requirement is to notify the NRC as soon as practical and within a maximum of 4 hours. Plausible because this is the time requirement for submitting an LER to the NRC for this event per 10 CFR 50.73

Technical Reference(s): LS-AA-1110, 10 CFR 50.72, 10 CFR 50.73

Proposed references to be provided to applicants during examination: 10 CFR 50.72, 10 CFR 50.73

Learning Objective:

Question Source: Modified Bank – JAF 20-1 NRC #100

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.30
	Importance Rating	4.1

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Proposed Question: #100

The plant is operating at 50% power with the following:

- A Feedwater level control malfunction occurs.
- The Reactor automatically scrams on low Reactor water level.
- Reactor water level reaches a low of 160" and is then recovered to the normal band using manual control of Feedwater.
- NO other complications occur.

Which one of the following describes the most restrictive time requirement for notifying the NRC, in accordance with LS-AA-1110, Safety (SAF)?

Notification must be made to the NRC as soon as practical and within a maximum time limit of…

A. 15 minutes

B. 1 hour

C. 4 hours

D. 60 days

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	223002 A2.03
	Importance Rating	4.0

PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System logic failures

Proposed Question: #87

The plant is operating at 100% power with the following:

- N2-RESP-4, LPRM Calibration, is being performed using Traversing Incore Probe (TIP) Machine #1.

Then, a small steam leak in the Drywell results in the following:

- Drywell pressure is 2 psig and slowly rising.
- One minute later, the following indications are available at the TIP #1 control drawer:
  - Detector position is 0942.
  - CORE TOP light is lit.
  - IN CORE light is lit.
  - IN SHIELD light is extinguished.
  - SQUIB MONITOR light is extinguished.
  - SHEAR VALVE MONITOR light is extinguished.
  - BALL VALVE OPEN light is lit.
  - BALL VALVE CLOSED light is extinguished.

Which one of the following describes the required action(s) to be directed, if any, based on these indications, in accordance with plant procedures?

A. Direct immediately firing the shear valve.

B. The TIP has responded properly for plant conditions, no actions are required to be directed.

C. Direct immediately closing the ball valve. If the ball valve fails to close, then direct firing the shear valve.

D. Direct attempting to withdraw the detector. If the detector withdraws, then direct verifying closed the ball valve. If the detector fails to withdraw, then direct firing the shear valve.

Proposed Answer: D

Explanation: With Drywell pressure >1.68 psig, the TIP should have withdrawn and isolated. The given TIP indications show that the detector is still in the core, the ball valve is still open, and the shear valve has not been fired. N2-SOP-83 Arrow C contains the guidance for responding to a failure of a TIP detector to withdraw and isolate on a high Drywell pressure signal. The procedure directs attempting to withdraw the detector and verifying closed the ball valve as a backup to the failed automatic action. Only if this fails is the shear valve fired.

A. Incorrect – The procedure directs first trying to manually withdraw the detector and verifying closed the ball valve. Plausible because an isolation is required, the isolation has failed, and withdrawing the TIP detector will take additional time.
B. Incorrect – The given indications show that the TIP detector has failed to withdraw and isolate with a high Drywell pressure signal. Plausible that no actions would be required because this would be correct given different indications. Also plausible because there is no evidence of a release path from the TIP system.
C. Incorrect – N2-SOP-83 requires attempting to withdraw the detector first. Plausible because, given different conditions, the TIP could withdraw but fail to isolate.

Technical Reference(s): N2-SOP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP83C01 EO-2

Question Source: Bank - 2014 Cert #88

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 10/10/23 – Edited wording in D slightly (“direct closing” to “direct verifying closed”) based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 2.2.45
	Importance Rating	4.7

UNINTERRUPTABLE POWER SUPPLY (AC/DC)

Ability to determine or interpret technical specifications with action statements of greater than 1 hour

Proposed Question: #88

The plant is operating at 100% power when the normal AC power supply to 2VBA*UPS2D is lost and remains de-energized.

Which one of the following describes the required Technical Specification action, if any?

A. NO Technical Specification action is required.

B. The plant must be placed in Mode 3 within a maximum of 12 hours.

C. The normal AC power supply must be re-energized within a maximum of 24 hours.

D. The normal AC power supply must be re-energized OR 2VBA*UPS2B must be placed in service within a maximum of 24 hours.

Proposed Answer: A

Explanation: Technical Specification 3.8.7 bases do not require the normal AC power supply to the energized for 2VBA*UPS2D to be considered operable. With the DC power supply now powering the UPS, it remains operable and no Technical Specification action is required.

B. Incorrect – 2VBA*UPS2D remains operable and no Technical Specification action is required. Plausible because if more than one emergency UPS inverter were inoperable, this would be the required action per TS 3.8.7 Condition B.
C. Incorrect – 2VBA*UPS2D remains operable and no Technical Specification action is required. Plausible because if the given fault made the UPS inoperable, TS 3.8.7 Condition A would be satisfied by this action.
D. Incorrect – 2VBA*UPS2D remains operable and no Technical Specification action is required. Plausible because if the given fault made the UPS inoperable, TS 3.8.7 Condition A would be satisfied by these actions.

Technical Reference(s): Technical Specification 3.8.7 and bases, N2-OP-71D

Proposed references to be provided to applicants during examination: Technical Specification 3.8.7

Learning Objective: N2-262002-RBO-14

Question Source: Bank – 2017 Cert #90

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	263000 A2.03
	Importance Rating	3.2

DC ELECTRICAL DISTRIBUTION

Ability to (a) predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal battery parameters

Proposed Question: #89

The plant is operating at 100% power with the following:

- A planned battery charger swap is in progress.
- Operators are in the field removing Division 1 battery charger 2BYS*CHGR2A1 and placing in service battery charger 2BYS*CHGR2A2.
- Battery bus 2BYS*SWG002A voltage prior to the battery charger swap reads 135 VDC.
- When 2BYS*CHGR2A1 breaker B302, DC OUTPUT, is placed in OFF, the following occurs:
  - 2BYS*SWG002A terminal voltage drops to 125 VDC.
  - Annunciator 852108, DIV I EMER BUS BYS 002A 125VDC SYSTEM TROUBLE, alarms.

Which one of the following describes the need for Condition entry for this situation, in accordance with Technical Specifications?

Technical Specification Condition entry is…

A. NOT required

B. required in LCO 3.8.4, DC Sources – Operating

C. required in LCO 3.8.6, Battery Cell Parameters

D. required in LCO 3.8.8, Distribution Systems – Operating

Proposed Answer: B

Explanation: Technical Specification Surveillance Requirement 3.8.4.1 requires battery terminal voltage to be ≥130 V on float charge for the DC electrical power subsystem to be considered operable. With the voltage below this value, the subsystem is inoperable. LCO 3.8.4 Condition A must be entered.

A. Incorrect – LCO 3.8.4 Condition A must be entered. Plausible because 125 VDC is the nominal voltage of this system.
C. Incorrect – Condition entry is not required in LCO 3.8.6. Plausible because this is a related Technical Specification and a Battery parameter is given in the stem at a degraded value.
D. Incorrect – Condition entry is not required in LCO 3.8.8. Plausible because this is a related Technical Specification and bus voltage would be lower than normal given these conditions

Technical Reference(s): Technical Specifications 3.8.4, 3.8.6, and 3.8.8 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-14

Question Source: Bank - 2016 NRC #77

Question History: 2016 NRC #77

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	264000 2.2.17
	Importance Rating	3.8

EMERGENCY GENERATORS (DIESEL/JET)

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator

Proposed Question: #90

The plant is operating at 100% power with the following:

- It has been determined that the Division I EDG, EGS*EG1, will NOT automatically start due to a failure in the start circuitry.
- An emergent job is being prepared to perform an inspection on the Division II EDG, EGS*EG3, to rule out a common cause failure.
- This emergent job has the potential to result in unplanned inoperability of the EGS*EG3.
- Line 5, Line 6, and the Division III EDG, EGS*EG2, will remain operable and protected throughout the job.

Which one of the following describes:

(1) the Completion Time for restoring an EDG to operable status if the Division II EDG, EGS*EG3, does become inoperable during this job, in accordance with Technical Specifications,

and

(2) which department is responsible for performing the risk assessment for this job, in accordance with plant procedures?

	(1) Completion Time	(2) Department Responsible for Performing Risk Assessment
A.	2 hours	Operations
B.	2 hours	Work Control
C.	24 hours	Operations
D.	24 hours	Work Control

Proposed Answer: A

Explanation: If the Division II EDG, EGS*EG3, does become inoperable during this job, then Technical Specification 3.8.1 Condition E would be entered. With the Division I and II EDGs inoperable, the Completion Time for restoring operability of an EDG in this Condition is 2 hours. Operations is responsible for performing the risk assessment.

B. Incorrect – Operations is responsible for performing the risk assessment. Plausible because Work Control is responsible for other portions of the planning process.
C. Incorrect – With the Division I and II EDGs inoperable, the Completion Time for restoring operability of an EDG in Technical Specification 3.8.1 Condition E is 2 hours. Plausible because the Completion Time would be 24 hours if one of the two inoperable EDGs was the Division III EDG.
D. Incorrect – With the Division I and II EDGs inoperable, the Completion Time for restoring operability of an EDG in Technical Specification 3.8.1 Condition E is 2 hours. Plausible because the Completion Time would be 24 hours if one of the two inoperable EDGs was the Division III EDG. Operations is responsible for performing the risk assessment. Plausible because Work Control is responsible for other portions of the planning process.

Technical Reference(s): OP-AA-107, Technical Specification 3.8.1

Proposed references to be provided to applicants during examination: Technical Specification 3.8.1

Learning Objective: N2-264001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 10/10/23 – Edited 1st bullet to give reason for EDG inoperability and edited 2nd bullet to specify that job is emergent, based on NRC comments.
TRH 10/25/23 – Edited 3rd bullet to specify that job is emergent, based on NRC comments.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	201002 2.2.36
	Importance Rating	4.2

REACTOR MANUAL CONTROL SYSTEM

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation

Proposed Question: #91

The plant is operating at 100% power with the following:

- Control rod 10-23 is unable to be selected due to a circuit malfunction.
- Corrective maintenance is in progress.
- Completion time for this corrective maintenance is unknown.
- Surveillance Requirement (SR) 3.1.3.3, “Insert each withdrawn control rod at least one notch”, was last satisfied 92 days ago for this control rod.
- The required frequency for SR 3.1.3.3 is 92 days.

Which one of the following describes the requirement for this surveillance, in accordance with Technical Specifications?

The surveillance must be completed within a maximum of _____ after the surveillance frequency expires or then the associated LCO must be declared NOT met.

A. 8 hours

B. 24 hours

C. 7 days

D. 23 days

Proposed Answer: D

Explanation: Surveillance Requirement (SR) 3.0.2 states that the frequency for each SR is met if the surveillance is performed within 1.25 times the specified frequency. This allows 25% grace period for completion of the surveillance. 25% of 92 days is 23 days.

A. Incorrect – SR 3.0.2 allows a grace period of up to 25% of the specified frequency (23 days for a 92 day frequency). Plausible because 8 hours is a time used in multiple Technical Specifications.
B. Incorrect – SR 3.0.2 allows a grace period of up to 25% of the specified frequency (23 days for a 92 day frequency). Plausible because this would provide one full day (day and night shift) to arrange to complete the surveillance. Also plausible because 24 hours is used in SR 3.0.3.
C. Incorrect – SR 3.0.2 allows a grace period of up to 25% of the specified frequency (23 days for a 92 day frequency). Plausible because 7 days is a time used in multiple Technical Specifications and would provide a full week to arrange to complete the surveillance.

Technical Reference(s): Technical Specification 3.1.1 and SR 3.0.2, Surveillance Frequency Control Program

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-201002-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	2
	K/A #	214000 A2.04
	Importance Rating	3.3

ROD POSITION INFORMATION SYSTEM

Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Power supply loss

Proposed Question: #92

The plant is operating at 100% power with the following:

- All power is lost to the Rod Position Information System (RPIS) due to a breaker malfunction on 2VBS-PNLA101.
- Then, a spurious Reactor scram occurs.
- APRMs indicate downscale.
- Reactor water level lowers to 135" and then rises with Feedwater injecting.

Which one of the following describes the required execution of the Emergency Operating Procedures?

Enter N2-EOP-RPV, RPV Control, and…

A. remain in N2-EOP-RPV until all N2-EOP-RPV entry conditions clear.

B. then exit to N2-EOP-C5, RPV Control - ATWS. N2-EOP-C5 may be exited once all IRMs are fully inserted and indicate less than range 7.

C. then exit to N2-EOP-C5, RPV Control - ATWS. N2-EOP-C5 may NOT be exited until control rod positions are known. Control rod positions CANNOT be determined until power is restored to RPIS.

D. then exit to N2-EOP-C5, RPV Control - ATWS. N2-EOP-C5 may NOT be exited until control rod positions are known. Control rod positions can be determined by alternate means before power is restored to RPIS.

Proposed Answer: D

Explanation: N2-EOP-RPV entry is required because Reactor water level lowered below 159.3". The loss of power to RPIS results in loss of all control rod position indication in the Main Control Room. With no control rod position information, you cannot verify the Reactor will remain shutdown without boron, therefore N2-EOP-RPV must be exited and N2-EOP-C5 must be entered. N2-EOP-C5 is only exited once it is determined that the Reactor will stay shutdown without boron. This determination can only be made once control rod positions are known. Therefore N2-EOP-C5 cannot be exited until control rod positions are known. N2-OP-96 contains guidance for determining control rod positions by measuring resistances. This method works without power from 2VBS-PNLA101.

A. Incorrect – With control rod position unknown, N2-EOP-RPV must be exited and N2-EOP-C5 must be entered. Plausible because APRMs being downscale indicate that the scram has been at least partially successful.
B. Incorrect – N2-EOP-C5 cannot be exited until control rod positions are known. Plausible because IRMs less than range 7 allows RPV cooldown while still in N2-EOP-C5.
C. Incorrect – Control rod positions can be determined by alternate means before power is restored to RPIS. Plausible because all installed control rod position indication is lost.

Technical Reference(s): N2-EOP-RPV, N2-EOP-C5, N2-OP-96

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Modified Bank – 2017 Cert #82

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 1
Group # 1
K/A # 295006 AA2.02
Importance Rating 4.4

SCRAM

Ability to determine and/or interpret the following as they apply to SCRAM: Control rod position

Proposed Question: #82

The plant is operating at 100% power with the following:

- All control rod position indication is lost.
- Then, a Main Turbine trip occurs.
- APRMs indicate downscale.
- Reactor water level lowers to 135" and then rises with Feedwater injecting.

Which one of the following describes the proper Emergency Operating Procedure (EOP) execution for this transient?

Enter N2-EOP-RPV, RPV Control, and…

A. remain in N2-EOP-RPV until all N2-EOP-RPV entry conditions clear.

B. then exit to N2-EOP-C5, Failure to Scram. N2-EOP-C5 may NOT be exited until control rod positions are known.

C. then exit to N2-EOP-C5, Failure to Scram. N2-EOP-C5 may be exited once all IRMs are fully inserted and indicate less than range 7.

D. then exit to N2-EOP-C5, Failure to Scram. N2-EOP-C5 may be exited if Standby Liquid Control is used to inject the Cold Shutdown Boron Weight.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	290002 2.1.23
	Importance Rating	4.4

REACTOR VESSEL AND INTERNALS

Ability to perform general or normal operating procedures during any plant condition

Proposed Question: #93

The plant is operating at 3% power during a startup with the following:

- Reactor pressure is 900 psig and stable.
- At time 0400, it is determined that Reactor Vessel Pressure Boundary Leakage exists.
- Attempts to fix the Pressure Boundary Leakage have failed.

Which one of the following is the latest time to place the Reactor Mode Switch in SHUTDOWN, in accordance with Technical Specifications?

A. 0800 today

B. 1600 today

C. 2000 today

D. 2000 tomorrow

Proposed Answer: C

Explanation: At time 0400, Technical Specification 3.4.5 Condition A must be entered due to discovery of Reactor Vessel Pressure Boundary Leakage. This Condition gives 4 hours to eliminate the Pressure Boundary Leakage. If the leakage is not fixed, then Condition D must be entered. This Condition gives 12 hours to be in Mode 3 and 36 hours to be in Mode 4. Mode 3 is entered when the Reactor Mode Switch is placed in SHUTDOWN. Therefore, the latest time to place the Reactor Mode Switch in SHUTDOWN is 2000 today (0400 + 4 hours +12 hours = 2000 on the same day).

A. Incorrect – The latest time to place the Reactor Mode Switch in SHUTDOWN is 2000 today. Plausible because this is the time when the completion time for Condition A expires.
B. Incorrect – The latest time to place the Reactor Mode Switch in SHUTDOWN is 2000 today. Plausible because this is the time of discovery plus just the completion time for Required Action D.1 (Be in Mode 3).
D. Incorrect – The latest time to place the Reactor Mode Switch in SHUTDOWN is 2000 today. Plausible because this is the time for being in Mode 4.

Technical Reference(s): Technical Specification 3.4.5

Proposed references to be provided to applicants during examination: Technical Specification 3.4.5

Learning Objective: N2-101001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 10/13/23 – Replaced question based on NRC comment.
TRH 10/25/23 – Moved “attempts to fix PBL” from question to stem, based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.36
	Importance Rating	4.1

Knowledge of procedures and limitations involved in core alterations

Proposed Question: #94

The plant is in Mode 5 with the following:

- Core shuffle is in progress.
- A four cell section of the core is ***displayed on the following page***.

Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with N2-FHP-13.3, Core Shuffle?

A. The fuel is loaded correctly. Core shuffle may continue with NO additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to core verification.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Refuel Floor SRO.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Completion of a formal evaluation is required before fuel movement may resume.

LYZ167
LYZ148
LYZ101
LYZ123
LYZ483
LYZ297
LYZ491
LYZ318
LYZ023
LYZ132
LYZ154
LYZ127
LYZ199
LYZ426
LYZ327
LYZ018

Proposed Answer: D

Explanation: The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell, the position of the channel fastener, and the orientation of the text on the handle. N2-FHP-13.3 section 3.3.4 defines this as a Fuel Movement Discrepancy (FMD). N2-FHP-13.3 requires fuel movement to be stopped and requires completion of a formal evaluation prior to resuming fuel movements.

A. Incorrect – There is a discrepancy in the fuel loading. Plausible because 15 of the 16 assemblies are loaded correctly and the overall shape is symmetrical.
B. Incorrect – Core shuffle cannot continue with the given error. Plausible because no significant adverse consequence would be expected from this fuel movement discrepancy until Reactor operation recommenced.
C. Incorrect – The discrepancy cannot be immediately fixed without a formal evaluation. Plausible because the issue would be fairly simple to fix.

Technical Reference(s): N2-FHP-13.3

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: S-101002-RBO-9

Question Source: Bank - 2021 NRC #95

Question History: 2021 NRC #95

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.38
	Importance Rating	3.8

Knowledge of the station's requirements for verbal communications when implementing procedures

Proposed Question: #95

The plant is operating at 100% power when a major transient causes a very large number of alarms.

Which one of the following describes the requirements for entering and exiting the transient annunciator response, in accordance with OP-AA-103-102-1001, Strategies for Successful Transient Mitigation?

Transient annunciator response is entered…

A. automatically. Transient annunciator response is exited only when announced by the Unit Supervisor.

B. automatically. Transient annunciator response is exited automatically when all EOPs are exited.

C. only when announced by the Unit Supervisor. Transient annunciator response is exited only when announced by the Unit Supervisor.

D. only when announced by the Unit Supervisor. Transient annunciator response is exited automatically when all EOPs are exited.

Proposed Answer: A

Explanation: OP-AA-103-102-1001 allows automatically entering transient annunciator response at the start of a major transient. The procedure requires the Unit Supervisor to communicate to the crew when exiting transient annunciator response.

B. Incorrect – The procedure requires the Unit Supervisor to communicate to the crew when exiting transient annunciator response. Plausible because this is a strong indication that the transient is over.
C. Incorrect – OP-AA-103-102-1001 allows automatically entering transient annunciator response at the start of a major transient. Plausible because this transition significantly relaxes annunciator response and communication standards.
D. Incorrect – OP-AA-103-102-1001 allows automatically entering transient annunciator response at the start of a major transient. Plausible because this transition significantly relaxes annunciator response and communication standards. The procedure requires the Unit Supervisor to communicate to the crew when exiting transient annunciator response. Plausible because this is a strong indication that the transient is over.

Technical Reference(s): OP-AA-103-102-1001, OP-AA-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – 4/23 NRC #100

Question History: 4/23 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.37
	Importance Rating	4.6

Ability to determine operability or availability of safety-related equipment (SRO Only)

Proposed Question: #96

The plant is operating at 50% power with the following:

- An outboard Primary Containment Isolation Valve (PCIV) failed its stroke time testing.
- To comply with the associated LCO, the inoperable valve has been closed and deactivated.
- Maintenance has been performed to repair the valve.
- Post Maintenance Testing (PMT) requires the valve to be re-opened for stroke time testing.

Which one of the following describes the ability to re-open this valve, in accordance with Technical Specifications

A. Enter LCO 3.0.3, complete the PMT, then exit LCO 3.0.3

B. The inboard PCIV must be closed before re-opening the valve

C. The plant must be placed in a different Mode before re-opening the valve

D. The valve may be re-opened under administrative controls without entering LCO 3.0.3, closing the inboard PCIV, or changing plant mode

Proposed Answer: D

Explanation: Technical Specification 3.0.5 provides an allowance to re-open the PCIV under administrative control to perform testing required to demonstrate its operability. This is an exception to Technical Specification 3.0.2.

A. Incorrect – Technical Specification 3.0.3 entry is not required. Plausible because entry into Technical Specification 3.0.3 is used when an LCO is not met and the associated actions are not met and no other allowance is provided.
B. Incorrect – The inboard PCIV does not have to be closed. Plausible because closing the inboard PCIV would be another acceptable method of complying with Technical Specifications.
C. Incorrect – A mode change is not required. Plausible because the applicability of Technical Specifications changes based on plant mode. Transitioning to Mode 4 (currently in Mode 1) would be another acceptable method of complying with Technical Specifications.

Technical Reference(s): Technical Specifications 3.0.2, 3.0.5, 3.6.1.3

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 22-1 NRC #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:
TRH 10/13/23 – Replaced question due to NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.38
	Importance Rating	4.5

Knowledge of conditions and limitations in the facility license

Proposed Question: #97

The plant is operating at 100% power when the following sequence of events occurs:

Date and Time (hhmm)	Condition
12/1/23 at 0000	• Technical Specification LCO 3.0.3 is entered.
12/1/23 at 0500	• The plant enters Mode 2.

Which one of the following identifies the latest time by which the plant must be placed in Mode 4, in accordance with Technical Specifications?

A. 12/2/23 at 1100

B. 12/2/23 at 1300

C. 12/3/23 at 0700

D. 12/3/23 at 0900

Proposed Answer: B

Explanation: When LCO 3.0.3 is entered on 12/1/23 at 0000, 7 hours are allowed to place the plant in Mode 2 and 37 hours are allowed to place the plant in Mode 4. The plant is placed in Mode 2 earlier than required (by 2 hours). The bases for LCO 3.0.3 state, “If a lower Mode is entered in less time that allowed, however, the total allowable time to enter Mode 4, or other applicable Mode, is not reduced.” Therefore, the full 37 hours from 12/1/23 at 0000 are still allowed for entering Mode 4. This results in the latest time being 12/2/23 at 1300 (12/1/23 at 0000 + 37 hours = 12/1/23 at 0000 + 1 day and 13 hours = 12/2/23 at 1300).

A. Incorrect – The full 37 hours from 12/1/23 at 0000 are still allowed for entering Mode 4. Plausible because this would be correct if bases stated that entering Mode 2 two hours early resulted in the need to enter Mode 4 two hours early.
C. Incorrect – The latest time for entering Mode 4 is 12/2/23 at 1300. Plausible because this would be the answer if the time limits in LCO 3.0.3 were added together (7 hours + 13 hours + 37 hours = 57 hours) and a 2 hour penalty were incurred for entering Mode 2 early.
D. Incorrect – The latest time for entering Mode 4 is 12/2/23 at 1300. Plausible because this would be the answer if the time limits in LCO 3.0.3 were added together (7 hours + 13 hours + 37 hours = 57 hours).

Technical Reference(s): Technical Specification 3.0.3 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – 2019 NRC #96

Question History: 2019 NRC #96

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)

Proposed Question: #98

Which one of the following identifies a chemistry parameter and associated threshold used to define Loss of the Fuel Clad Barrier, in accordance with the Emergency Action Levels?

Dose equivalent Reactor coolant activity of…

A. Iodine-131; 0.2 µCi/gm

B. Iodine-131; 300 µCi/gm

C. Cobalt-60; 0.2 µCi/gm

D. Cobalt-60; 300 µCi/gm

Proposed Answer: B

Explanation: The Emergency Action Levels define Loss of the Fuel Clad barrier as dose equivalent Reactor coolant activity of I-131 above the threshold of 300 µCi/gm.

A. Incorrect – The threshold is 300 µCi/gm. Plausible because 0.2 µCi/gm is used in Technical Specification 3.4.8.
C. Incorrect – The isotope is I-131. Plausible because Co-60 is a significant radioisotope related to radiological safety. The threshold is 300 µCi/gm. Plausible because 0.2 µCi/gm is used in Technical Specification 3.4.8.
D. Incorrect – The isotope is I-131. Plausible because Co-60 is a significant radioisotope related to radiological safety.

Technical Reference(s): Emergency Action Levels

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-204000-RBO-10

Question Source: Bank – 2017 NRC #94

Question History: 2017 NRC #94

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.23
	Importance Rating	4.4

Knowledge of the bases for prioritizing emergency operating procedures implementation

Proposed Question: #99

The plant has experienced an earthquake with the following:

- The Diesel Fire pump is the only available pump.
- Firefighting activities are in progress.
- Reactor water level is 130” and slowly lowering.
- Spent Fuel Pool water level is 3” below the level of the weirs and slowly lowering.

Which one of the following describes the prioritization for use of the Diesel Fire pump, in accordance with N2-EOP-B-HOT, NMP2 Emergency Operating Procedures (EOP) Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. is allowed to set the priority for use of the Diesel Fire pump based on plant conditions.

B. must always make lining up the Diesel Fire pump for Reactor injection the priority.

C. must always make lining up the Diesel Fire pump for Spent Fuel Pool makeup the priority.

D. must always make maintaining the Diesel Fire pump lined up to the Fire Water header the priority.

Proposed Answer: A

Explanation: N2-EOP-B-HOT and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for pump use or between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

B. Incorrect – Reactor injection does NOT always take priority over firefighting activities and Spent Fuel Pool makeup. Plausible because adequate core cooling, if challenged, is given priority over other concerns in the EOPs.
C. Incorrect – Spent Fuel Pool makeup does NOT always take priority over Reactor injection and firefighting activities. Plausible because low Spent Fuel Pool water level can lead to a direct release to the environment, whereas the Reactor is inside a Primary Containment.
D. Incorrect – Firefighting activities do NOT always take priority over Reactor injection and Spent Fuel Pool makeup. Plausible because firefighting is the design purpose of the Diesel Fire pump, whereas other systems can be made available for these other functions.

Technical Reference(s): N2-EOP-B-HOT, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2020 NRC #100

Question History: NMP1 2020 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.29
	Importance Rating	4.4

Knowledge of the emergency plan implementing procedures

Proposed Question: #100

An emergency is in progress with the following:

- An Alert has been declared.
- A condition requires a worker to enter a high radiation area.
- The required dose to perform the job is 17,000 mRem.

Which one of the following describes the ability to authorize this exposure, in accordance with EP-AA-113, Personnel Protective Actions?

This exposure…

A. can be authorized for either protecting valuable plant equipment or life-saving activities.

B. can be authorized for life-saving activities, but NOT for protecting valuable plant equipment.

C. CANNOT be authorized unless the emergency is upgraded to a higher declaration.

D. CANNOT be authorized for either protecting valuable plant equipment or life-saving activities regardless of the emergency declaration level.

Proposed Answer: B

Explanation: EP-AA-113 allows emergency dose authorization of up to 10 Rem for protecting valuable plant equipment. A dose authorization of 25 Rem is allowed for life-saving activities. Therefore, the given job that requires 17 Rem can be authorized for life-saving activities, but not for protecting valuable plant equipment.

A. Incorrect – The given dose is too high for protecting valuable plant equipment. Plausible because an emergency dose up to 10 Rem could be authorized for this purpose.
C. Incorrect – Emergency dose authorization is allowed at the Alert level. Plausible because this is the 2nd lowest declaration level and some E-plan actions do not happen until higher levels (eg. PARs, KI).
D. Incorrect – The given dose can be authorized for life-saving activities. Plausible because this is above the max dose for other activities and is well above the normal federal limit.

Technical Reference(s): EP-AA-113

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - JAF 17-2 NRC #98

Question History: JAF 17-2 NRC #98

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Constellation

Training ID: **Nov 2023 NRC JPM SRO COO1** Revision: **0.0**

Title: **Review Jet Pump Surveillance**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mecki Ghabarou	7/12/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 30 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OSP-LOG-D001, Daily Checks Log
2. N2-OP-29, Reactor Recirculation System
3. NUREG 1123 K/A 2.1.18 (3.8)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the operator's ability to review daily logs.

   b. This JPM is not considered Alternate Path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded Pass/Fail.

2. Task Information:

   a. N2-277001-01004, Perform N2-OSP-LOG-D001, Daily Checks

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO ☐ RO ☐ EO ☐ N/A				
Evaluation Method:	☒ Perform ☐ Simulate				
Evaluation Location:	☐ Plant ☐ Simulator ☒ Classroom				
Time Critical Task:	☐ Yes ☒ No				
Alternate Path:	☐ Yes ☒ No				
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	2.1.18 - Ability to make accurate, clear, and concise logs, records, status boards, and reports				
K/A Importance Rating:		RO	3.6	SRO	3.8

4. Recommended Start Location

   a. NLC Classroom

5. JPM Setup (if required)

   a. Ensure a blank copy of N2-OSP-LOG-D001 is available.

   b. Ensure a pre-filled copy of N2-OSP-LOG-D001, Attachment 10 “Two Loop Jet Pump Operability Verification” is available. Base the copy off of the data in JPM Attachment 1. However, in steps 2.4 and 3.4, mark both loops as “Yes” (despite loop B being actually below the low limit).

   c. Ensure each operator has a calculator.

   d. Ensure Technical Specifications are available.

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • N2-OSP-LOG-D001 Attachment 10 has been completed. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *review the completed N2-OSP-LOG-D001 Attachment 10. Summarize your results and document any recommended actions on the provided "JPM Worksheet".*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	Obtains a copy of the reference procedure and review / utilize the correct section of the procedure.	P	SAT / UNSAT ***STD:*** *Refers to the provided N2-OSP-LOG-D001, attachment 10.*
3.	Reviews section 2.0.	P (Attachment 10 section 2.0)	***PASS / FAIL** ***STD:*** *Reviews the recorded data in Section 2.0 and determines that Loop B Jet Pump flow is below the low limit.* ***Failure*** *= does NOT identify that Loop B Jet Pump flow is below the low limit.*
4.	Reviews section 3.0.	P (Attachment 10 section 3.0)	***PASS / FAIL** ***STD:*** *Reviews the recorded data in Section 3.0 and determines that Loop B Jet Pump flow is below the low limit.* ***Failure*** *= does NOT identify that Loop B Jet Pump flow is below the low limit.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
4.	Reviews section 4.0.	P (Attachment 10 section 4.0)	**SAT / UNSAT** ***STD:*** *Reviews the recorded data in Section 4.0 and determines the data is accurate (this may be judged SAT by lack of a report to the contrary).*
5.	Determines Technical Specification requirement	P	***PASS / FAIL** ***STD:*** *Determines Technical Specification 3.4.3 Condition A must be entered (be in Mode 3 within 12 hours).* ***Failure*** *= does NOT identify that Specification 3.4.3 Condition A must be entered.*

TASK STANDARD	*Attachment 10 of N2-OSP-LOG-D001 reviewed. Loop B Jet pump flow is identified outside of allowable limits. Technical Specification entry identified.*

STOP TIME	

Evaluator's Answer Key

Do not Provide to Operator

Record your results below:	
	Name:
	Summary of Review:
	• **Actual Loop A jet pump flows are within limits of Table 10-1** ***(may be graded SAT based on absence of a report to the contrary)*** • **Actual Loop B jet pump flows are NOT within limits of Table 10-1** • **Actual Loop A jet pump flows are within limits of Table 10-2** ***(may be graded SAT based on absence of a report to the contrary)*** • **Actual Loop B jet pump flows are NOT within limits of Table 10-2** • **Loop A and B Jet Pumps are within limits of Table 10-3** ***(may be graded SAT based on absence of a report to the contrary)*** **(or similar wording)**
	Summary of Recommended Actions:
	Enter Technical Specification 3.4.3 Condition A (be in Mode 3 within 12 hours) **(or similar wording)**

SRO JPM Handout

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • N2-OSP-LOG-D001 Attachment 10 has been completed.

INITIATING CUE	**(Operator Name)**, *review the completed N2-OSP-LOG-D001 Attachment 10. Summarize your results and document any recommended actions on the provided "JPM Worksheet".*

JPM Attachment 1: N2-OSP-LOG-D001 Data Sheet

Provide to Operator

Item #	Description	Value
1	**2RCS-HC1603A**, RECIRC LOOP A FLOW CONTROL	68%
2	**2RCS-HC1603B**, RECIRC LOOP B FLOW CONTROL	76%
3	**B22-R611A**, RECIRC LOOP 1A SUM JET PMP FLO	52 Mlbm/Hr (Flow Oscillations Minimal)
4	**B22-R611B**, RECIRC LOOP 1B SUM JET PMP FLO	44 Mlbm/Hr (Flow Oscillations Minimal)
5	**B35-R614** RECIRC FLOW LOOP A	42.0 Kgpm (Flow Oscillations Minimal)
6	**B35-R614** RECIRC FLOW LOOP B	41.0 Kgpm (Flow Oscillations Minimal)
7	**Computer Point Indications for Jet Pump Flows**	

Loop A		**Loop B**	
Jet pump 1	5.27	Jet pump 11	4.92
Jet pump 2	4.90	Jet pump 12	4.85
Jet pump 3	4.95	Jet pump 13	4.40
Jet pump 4	5.44	Jet pump 14	4.68
Jet pump 5	6.26	Jet pump 15	5.28
Jet pump 6	6.15	Jet pump 16	4.52
Jet pump 7	5.24	Jet pump 17	5.05
Jet pump 8	5.44	Jet pump 18	4.48
Jet pump 9	5.06	Jet pump 19	4.76
Jet pump 10	5.71	Jet pump 20	4.92

JPM Worksheet

Provide to Operator

Record your results below:	
	Name:
	Summary of Review:
	Summary of Recommended Actions:

Administrative JPM Handout Reference Packet

SRO JPM Handout

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • N2-OSP-LOG-D001 Attachment 10 has been completed. **Evaluator:** *Ask applicant if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Applicant Name)**, *review the completed N2-OSP-LOG-D001 Attachment 10. Summarize your results and document any recommended actions on the provided "JPM Worksheet".*

N/A, Plant is in single loop operation ..... (__)

N/A, Plant NOT in Mode 1 OR 2. ..... (__)

Initials

1.0 Record Recirc Pump Speed by checking appropriate choice below:

- 15 Hz (__)
- 60 Hz (✓)

2.0 Comparison of Indicated Jet Pump Loop Flows to Predicted Jet Pump Loop Flows by Recirc Flow Control Valve Positions (SR 3.4.3.1.a)

2.1 Record Recirc Flow Control Valve (FCV) Positions, as follows:

2.1.1 IF Recirc FCV Positions, as read on 2RCS-HC1603A, RECIRC LOOP A FLOW CONTROL, AND 2RCS-HC1603B, RECIRC LOOP B FLOW CONTROL, are less than or equal to 85% for FCV A AND less than or equal to 95% for FCV B, THEN record Recirc FCV Positions in Table 10-1.

N/A, One OR BOTH Recirc FCV Positions, as read on 2RCS-HC1603A OR 2RCS-HC1603B, are greater than 85% for FCV A AND greater than 95% for FCV B..... (__) OPE

2.1.2 IF one OR BOTH Recirc FCV Positions as read on 2RCS-HC1603A AND 2RCS-HC1603B, are greater than 85% for FCV A AND greater than 95% for FCV B,
THEN obtain Recirc FCV Positions from TARS as follows:

N/A, BOTH Recirc FCV Positions, as read on 2RCS-HC1603A OR 2RCS-HC1603B, are less than or equal to 85% for FCV A AND less than or equal to 95% for FCV B..... (✓)

a. Using TARS Point ID 2002, obtain Recirc Loop A FCV Position AND record in Table 10-1. N/A

b. Using TARS Point ID 2003, obtain Recirc Loop B FCV Position AND record in Table 10-1. N/A

Initials

2.2 Record Jet Pump Loop Flows as follows:

2.2.1 IF flow oscillations on indicators do NOT make an accurate reading difficult, THEN record Summed Jet Pump Loop Flows from Indicators B22-R611A, RECIRC LOOP 1A SUM JET PMP FLO, AND B22-R611B, RECIRC LOOP 1B SUM JET PMP FLO, on 2CEC*PNL602, in the following places:

N/A, TARS used due to flow oscillations..... (__)

- Table 10-1 — OPE
- Table 10-2 — OPE

2.2.2 IF flow oscillations on indicators make an accurate reading difficult, THEN perform the following:

N/A, 2CEC*PNL602 meters were used ..... (✓)

a. Obtain a 1 min mean of PID 2042, Loop A Jet Pump Flow using TARS Point ID 2674. — N/A

b. Record the value of Jet Pump Loop A Flow obtained in the Step above in the following places:

- Table 10-1 — N/A
- Table 10-2 — N/A

c. Obtain a 1 min mean of PID 2043, Loop B Jet Pump Flow using TARS Point ID 2673. — N/A

d. Record the value of Jet Pump Loop B Flow obtained in the Step above in the following places:

- Table 10-1 — N/A
- Table 10-2 — N/A

Initials

2.3 Determine the High AND Low Limits for Jet Pump Loop Flows as follows:

2.3.1 IF Recirc Pumps are in slow speed operation,
THEN enter the following in Table 10-1 for the High AND Low Limits:

N/A, Recirc Pumps are in high speed operation ..... (✓)

- High Limit: 21.43 Mlb_m/Hr
- Low Limit: 17.54 Mlb_m/Hr — N/A

2.3.2 IF the Recirc Pumps are in high speed operation,
THEN perform the following:

N/A, Recirc Pumps are in low speed operation ..... (__)

a. Using the Recirc FCV Position for Loop A recorded in Table 10-1, obtain the Jet Pump Loop Flow High AND Low Limits for Loop A from Figure 10-1 AND record them in Table 10-1. — OPE

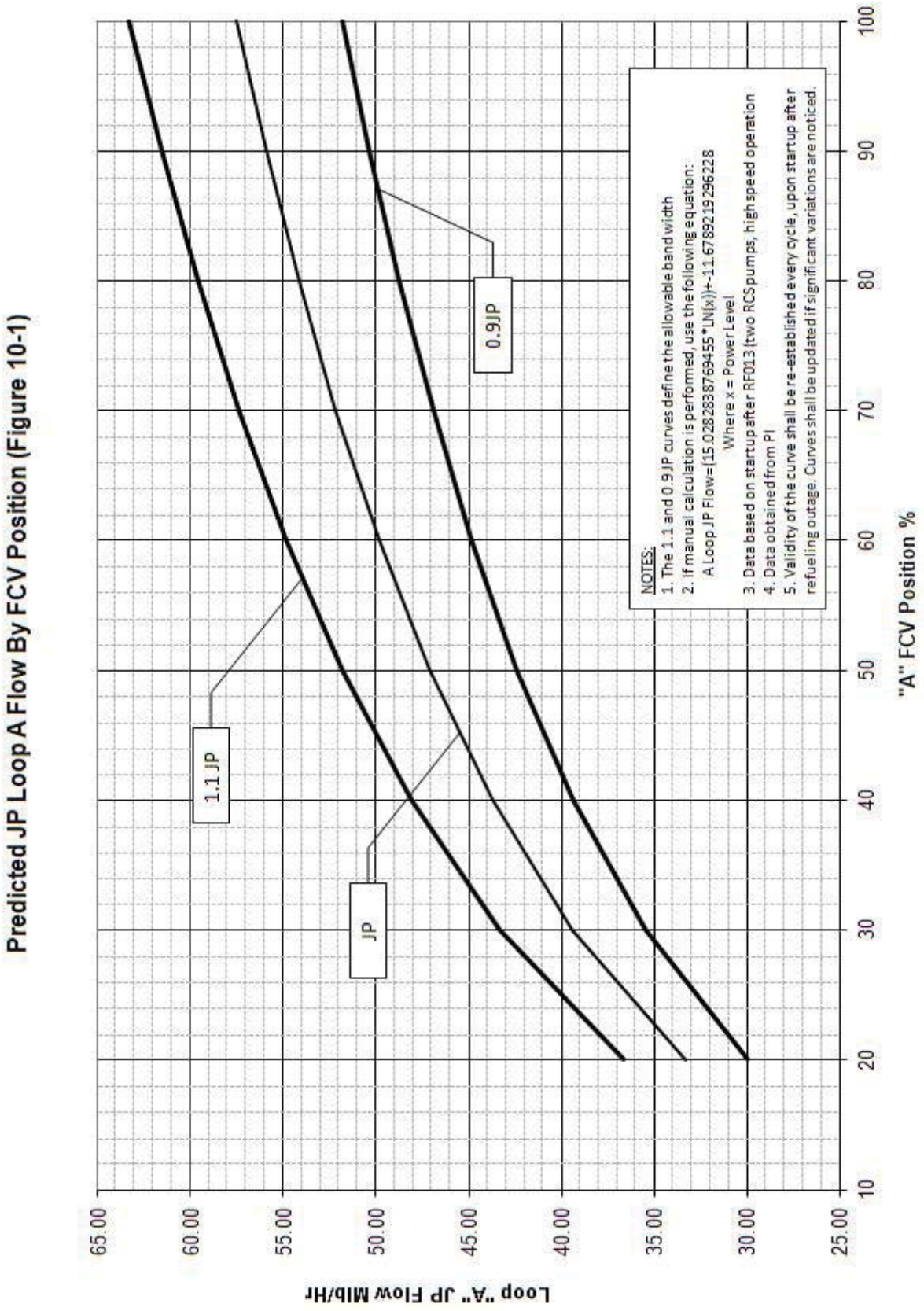
b. Using the Recirc FCV Position for Loop B recorded in Table 10-1, obtain the Jet Pump Loop Flow High AND Low Limits for Loop B from Figure 10-2 AND record them in Table 10-1. — OPE

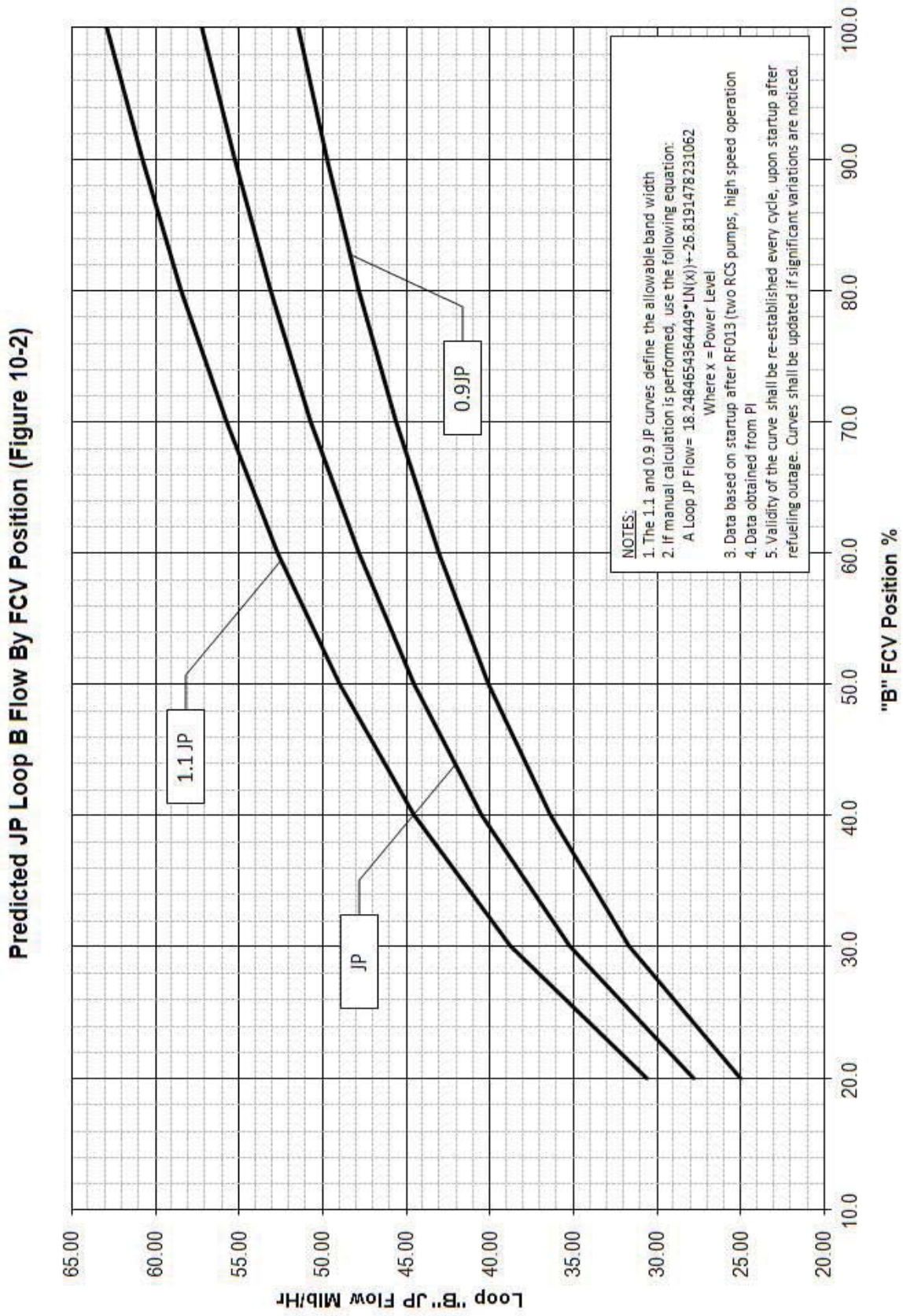
2.4 Compare the actual Loop A AND Loop B Jet Pump Flows to the respective Loop High AND Low Limits, as recorded in Table 10-1, AND indicate below whether the actual values fall within the Limits:

	Yes	No	
Loop A:	(✓)	(__)	
Loop B:	(✓)	(__)	OPE

Table 10-1

Recirc Loop A				Recirc Loop B			
FCV A (%)	Loop A Jet Pump Flow (Mlb_m/Hr)	High Limit (Mlb_m/Hr)	Low Limit (Mlb_m/Hr)	FCV B (%)	Loop B Jet Pump Flow (Mlb_m/Hr)	High Limit (Mlb_m/Hr)	Low Limit (Mlb_m/Hr)
68	52	56.5	46.5	76	44	57.5	47.0


Predicted JP Loop A Flow By FCV Position (Figure 10-1)
Loop "A" JP Flow Mlb/Hr
"A" FCV Position %
65.00
60.00
55.00
50.00
45.00
40.00
35.00
30.00
25.00
10
20
30
40
50
60
70
80
90
100
1.1 JP
JP
0.9JP
NOTES:
1. The 1.1 and 0.9 JP curves define the allowable band width
2. If manual calculation is performed, use the following equation:
A Loop JP Flow=(15.028283876945 5*LN(x))+-11.67892 19296228
Where x = Power Level
3. Data based on startup after RF013 (two RCS pumps, high speed operation
4. Data obtained from PI
5. Validity of the curve shall be re-established every cycle, upon startup after refueling outage. Curves shall be updated if significant variations are noticed.


Predicted JP Loop B Flow By FCV Position (Figure 10-2)
Loop "B" JP Flow Mlb/Hr
"B" FCV Position %
1.1 JP
JP
0.9JP
NOTES:
1. The 1.1 and 0.9 JP curves define the allowable band width
2. If manual calculation is performed, use the following equation:
A Loop JP Flow= 18.248455436444 49*LN(x)+-26.819147823106 2
Where x = Power Level
3. Data based on startup after RF013 (two RCS pumps, high speed operation
4. Data obtained from PI
5. Validity of the curve shall be re-established every cycle, upon startup after refueling outage. Curves shall be updated if significant variations are noticed.
65.00
60.00
55.00
50.00
45.00
40.00
35.00
30.00
25.00
20.00
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0

Initials

3.0 Comparison of Indicated Jet Pump Loop Flows to Predicted Jet Pump Loop Flows by Recirc Loop Drive Flows (TS SR 3.4.3.1.b)

3.1 Verify the Jet Pump Loop Flows have been recorded in Table 10-2. OPE

3.2 Obtain Recirc Loop Drive Flows as follows:

3.2.1 IF flow oscillations on recorder do NOT make an accurate reading difficult, THEN record Recirc Loop Drive Flows from recorder B35-R614, RECIRC FLOW LOOP B/FLOW LOOP A, on 2CEC*PNL602, in Table 10-2.

N/A, TARS used due to flow oscillations..... (__) OPE

3.2.2 IF flow oscillations on recorder makes an accurate reading difficult, THEN perform the following:

N/A, 2CEC*PNL602 recorder was used ..... (✓)

a. Obtain a 1 min mean of PID 2045, RCS Loop A Flow using TARS Point ID 2672. N/A

b. Record the value of Recirc Loop A Drive Flow in Table 10-2. N/A

c. Obtain a 1 min mean of PID 2046, RCS Loop B Flow using TARS Point ID 2671. N/A

d. Record the value of Recirc Loop B Drive Flow in Table 10-2. N/A

e. Attach TARS plot to this procedure. N/A

3.3 Determine the High AND Low Limits for Jet Pump Loop Flow as follows:

- For High Speed Pump Operation, perform 3.3.1 AND 3.3.2.
- For Low Speed Pump Operation, perform 3.3.3 AND 3.3.4.

3.3.1 Using the Recirc Loop A Drive Flow recorded in Table 10-2, obtain the Jet Pump Loop Flow High AND Low Limits for Loop A from Figure 10-3 AND record them in Table 10-2.

N/A, Recirc Pumps in Low Speed Operation ..... (__) OPE

Initials

3.3.2 Using the Recirc Loop B Drive Flow recorded in Table 10-2, obtain the Jet Pump Loop Flow High AND Low Limits for Loop B from Figure 10-4 AND record them in Table 10-2.

N/A, Recirc Pumps in Low Speed Operation ..... (__) OPE

3.3.3 Using the Recirc Loop A Drive Flow recorded in Table 10-2, calculate the Jet Pump Loop Flow High AND Low Limits for Loop A using the following equation AND record them in Table 10-2.

NOTE: "WDA" in the equation is "A" Recirc Loop Drive Flow in gpm, not kgpm, and JP units as calculated are Mlbm/hr.

$JP = 12.197 + 0.0006WDA + 0.000000008WDA^2$
JP high limit = (1.1)JP
JP low limit = (0.9)JP

N/A, Recirc Pumps in High Speed Operation ..... (✓) N/A

N/A
IV

3.3.4 Using the Recirc Loop B Drive Flow recorded in Table 10-2, calculate the Jet Pump Loop Flow High AND Low Limits for Loop B using the following equation AND record them in Table 10-2.

NOTE: "WDB" in the equation is "B" Recirc Loop Drive Flow in gpm, not kgpm, and JP units as calculated are Mlbm/hr.

$JP = 12.197 + 0.0006WDB + 0.000000008WDB^2$
JP high limit = (1.1)JP
JP low limit = (0.9)JP

N/A, Recirc Pumps in High Speed Operation ..... (✓) N/A

N/A
IV

Initials

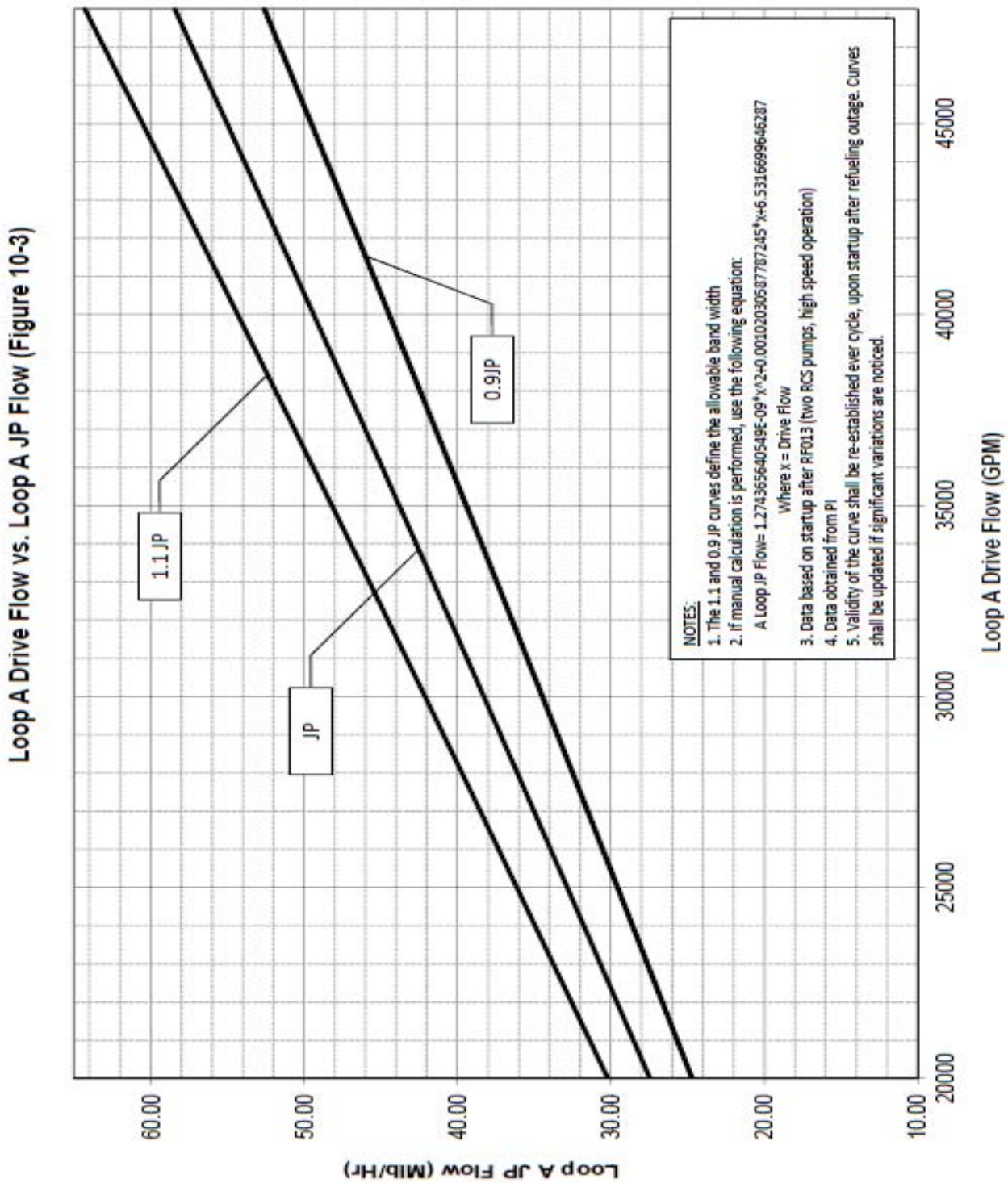
3.4 Compare the actual Loop A AND Loop B Jet Pump Flows to the respective Loop High AND Low Limits, as recorded in Table 10-2, AND indicate below whether the actual values fall within the Limits:

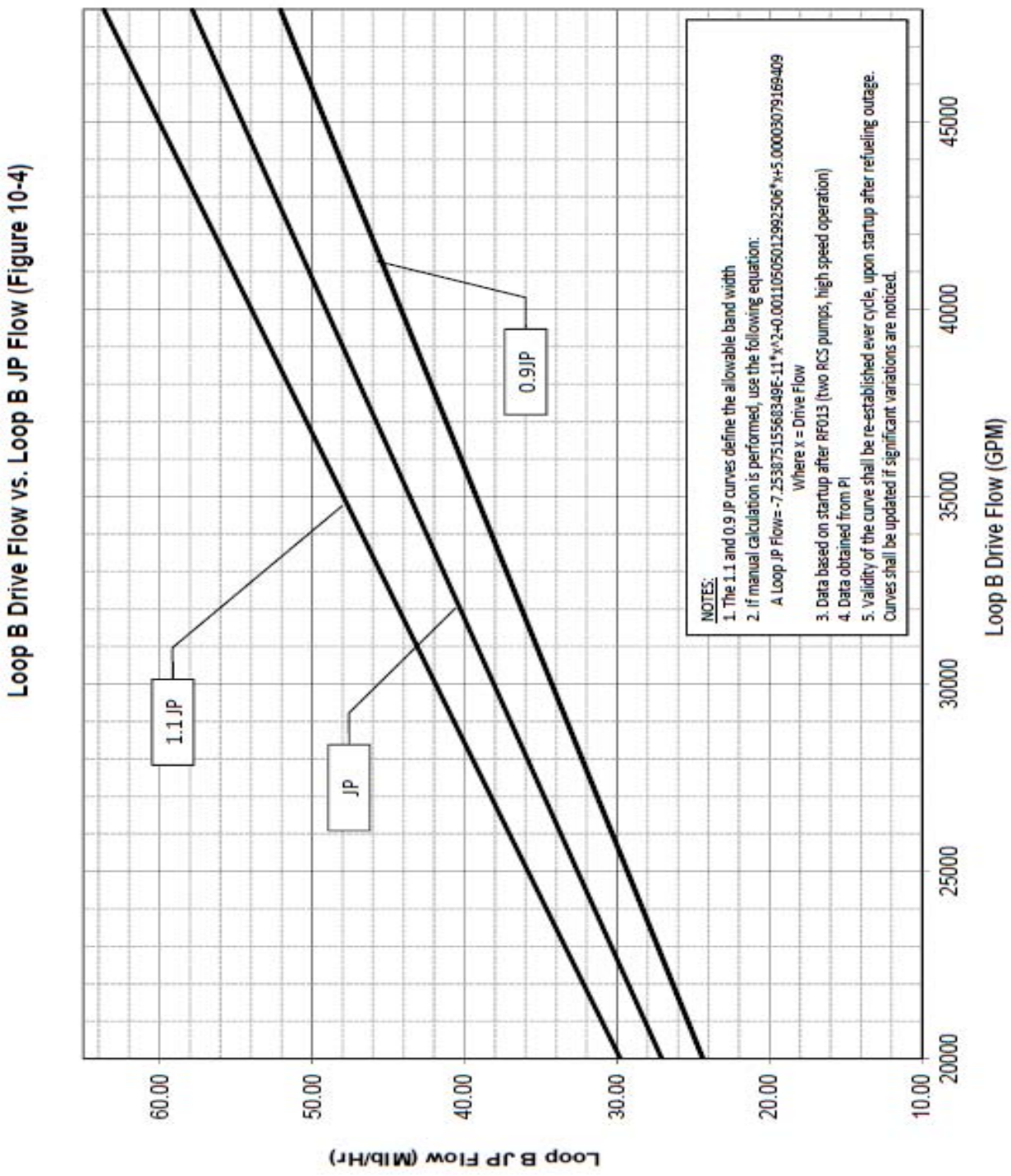
	Yes	No
Loop A:	(✓)	(__)
Loop B:	(✓)	(__)

OPE

Table 10-2

Recirc Loop A				Recirc Loop B			
Recirc Loop A Drive Flow (Kgpm)	Loop A Jet Pump Flow (Mlb_m/Hr)	High Limit (Mlb_m/Hr)	Low Limit (Mlb_m/Hr)	Recirc Loop B Drive Flow (Kgpm)	Loop B Jet Pump Flow (Mlb_m/Hr)	High Limit (Mlb_m/Hr)	Low Limit (Mlb_m/Hr)
42	52	56.5	46.5	41	44	55.0	45.0


Loop A Drive Flow vs. Loop A JP Flow (Figure 10-3)
1.1 JP
JP
0.9JP
NOTES:
1. The 1.1 and 0.9 JP curves define the allowable band width
2. If manual calculation is performed, use the following equation:
A Loop JP Flow= 1.27436566405498E-09*x^2+0.00102030587787245*x+6.53166699646287
Where x = Drive Flow
3. Data based on startup after RF013 (two RCS pumps, high speed operation)
4. Data obtained from PI
5. Validity of the curve shall be re-established ever cycle, upon startup after refueling outage. Curves shall be updated if significant variations are noticed.
Loop A JP Flow (Mlb/Hr)
60.00
50.00
40.00
30.00
20.00
10.00
20000
25000
30000
35000
40000
45000
Loop A Drive Flow (GPM)


Loop B Drive Flow vs. Loop B JP Flow (Figure 10-4)
1.1 JP
JP
0.9JP
NOTES:
1. The 1.1 and 0.9 JP curves define the allowable band width
2. If manual calculation is performed, use the following equation:
A Loop JP Flow= -7.25387515568349E-11*x^2+0.00110505012992506*x+5.00003079169409
Where x = Drive Flow
3. Data based on startup after RF013 (two RCS pumps, high speed operation)
4. Data obtained from PI
5. Validity of the curve shall be re-established ever cycle, upon startup after refueling outage. Curves shall be updated if significant variations are noticed.
Loop B JP Flow (MLb/Hr)
60.00
50.00
40.00
30.00
20.00
10.00
20000
25000
30000
35000
40000
45000
Loop B Drive Flow (GPM)

Initials

4.0 Comparison of Individual Jet Pumps ΔP to Average Jet Pump Loop ΔP (TS SR 3.4.3.1.c)

NOTE: Due to cracking in the Jet Pump #6 sensing line (reference CR-2008-002793), the potential for failure of the sensing line exists. If the Jet Pump #6 sensing line fails such that it reads downcomer pressure, its D/P reading will show a step jump of approximately 8% to 9% above its baseline and the indicated loop flow and core flow will increase. Jet Pump #5 D/P and generator output will not change. Should the step jump in Jet Pump #6 be observed, in addition to following the guidance in N2-OP-29, an IR shall be initiated.

4.1 Record value for each Jet Pump ΔP in Loop A, as read on computer points NSSFA102 to NSSFA111, in Table 10-3. OPE

4.2 Calculate Loop A Average Jet Pump ΔP for AND record in Table 10-3. OPE

4.3 Divide each Loop A Jet Pump ΔP by Loop A Average Jet Pump ΔP AND record the resulting Individual to Average ΔP Ratios in Table 10-3. OPE

4.4 For ALL Jet Pumps in Loop A, compare each Jet Pump's Individual to Average ΔP Ratio to the Limits given in Table 10-3 AND indicate below whether the actual values are within the Limits:

Jet Pump	Yes	No
1	(✓)	(__)
2	(✓)	(__)
3	(✓)	(__)
4	(✓)	(__)
5	(✓)	(__)
6	(✓)	(__)
7	(✓)	(__)
8	(✓)	(__)
9	(✓)	(__)
10	(✓)	(__)

OPE

Initials

4.5 Record value for each Jet Pump ΔP in Loop B, as read on computer points NSSFA112 to NSSFA121, on Table 10-3. OPE

4.6 Calculate Loop B Average Jet Pump ΔP for AND record on Table 10-3. OPE

4.7 Divide each Loop B Jet Pump ΔP by Loop B Average Jet Pump ΔP AND record the resulting Individual to Average ΔP Ratios in Table 10-3. OPE

4.8 For ALL Jet Pumps in Loop B, compare each Jet Pump's Individual to Average ΔP Ratio to the Limits given in Table 10-3 AND indicate below whether the actual values are within the Limits:

Jet Pump	Yes	No
11	(✓)	(__)
12	(✓)	(__)
13	(✓)	(__)
14	(✓)	(__)
15	(✓)	(__)
16	(✓)	(__)
17	(✓)	(__)
18	(✓)	(__)
19	(✓)	(__)
20	(✓)	(__)

OPE

Table 10-3 (Low Speed Operation)

Jet Pump	Computer Points (Mlbm/hr)	Individual to Average ΔP Ratio	Low Limit	High Limit
Loop A				
1	NSSFA102=		0.81	1.22
2	NSSFA103=		0.79	1.19
3	NSSFA104=		0.79	1.19
4	NSSFA105=		0.79	1.18
5	NSSFA106=		0.84	1.25
6	NSSFA107=		0.85	1.27
7	NSSFA108=		0.78	1.17
8	NSSFA109=		0.77	1.16
9	NSSFA110=		0.79	1.19
10	NSSFA111=		0.80	1.19
Total		Total) 10 = _____ % Loop A Average Jet ΔP Pump		
Loop B				
11	NSSFA112=		0.81	1.21
12	NSSFA113=		0.78	1.16
13	NSSFA114=		0.78	1.18
14	NSSFA115=		0.77	1.16
15	NSSFA116=		0.83	1.25
16	NSSFA117=		0.84	1.27
17	NSSFA118=		0.81	1.21
18	NSSFA119=		0.78	1.18
19	NSSFA120=		0.79	1.19
20	NSSFA121=		0.80	1.20
Total		Total) 10 = _____ % Loop B Average Jet ΔP Pump		
Calc Performed By (Initials):		Independently Verified By (Initials):		

Jet Pump	Computer Point (Mlbm/hr)	Individual to Average ΔP Ratio	Low Limit	High Limit
Loop A				
1	NSSFA102=		0.81	1.22
2	NSSFA103=		0.79	1.19
3	NSSFA104=		0.79	1.19
4	NSSFA105=		0.79	1.18
5	NSSFA106=		0.84	1.25
6	NSSFA107=		0.85	1.27
7	NSSFA108=		0.78	1.17
8	NSSFA109=		0.77	1.16
9	NSSFA110=		0.79	1.19
10	NSSFA111=		0.80	1.19
Total		Total) 10 = _____ % Loop A Average Jet ΔP Pump		
Loop B				
11	NSSFA112=		0.81	1.21
12	NSSFA113=		0.78	1.16
13	NSSFA114=		0.78	1.18
14	NSSFA115=		0.77	1.16
15	NSSFA116=		0.83	1.25
16	NSSFA117=		0.84	1.27
17	NSSFA118=		0.81	1.21
18	NSSFA119=		0.78	1.18
19	NSSFA120=		0.79	1.19
20	NSSFA121=		0.80	1.20
Total		Total) 10 = _____ % Loop B Average Jet ΔP Pump		
Calc Performed By (Initials):		Independently Verified By (Initials):		

Table 10-3 (75-100% Power)

Jet Pump	Computer Point (Mlbm/hr)	Individual to Average ΔP Ratio	Low Limit	High Limit
Loop A				
1	NSSFA102= 5.27	0.97	0.81	1.22
2	NSSFA103= 4.90	0.90	0.79	1.19
3	NSSFA104= 4.95	0.91	0.79	1.19
4	NSSFA105= 5.44	1.00	0.79	1.18
5	NSSFA106= 6.26	1.15	0.84	1.25
6	NSSFA107= 6.15	1.13	0.85	1.27
7	NSSFA108= 5.24	0.96	0.78	1.17
8	NSSFA109= 5.44	1.00	0.77	1.16
9	NSSFA110= 5.06	0.93	0.79	1.19
10	NSSFA111= 5.71	1.05	0.80	1.19
Total	54.42	Total) 10 = 5.44 % Loop A Average Jet ΔP Pump		
Loop B				
11	NSSFA112= 4.92	1.03	0.81	1.21
12	NSSFA113= 4.85	1.01	0.78	1.16
13	NSSFA114= 4.40	0.92	0.78	1.18
14	NSSFA115= 4.68	0.98	0.77	1.16
15	NSSFA116= 5.28	1.10	0.83	1.25
16	NSSFA117= 4.52	0.94	0.84	1.27
17	NSSFA118= 5.05	1.06	0.81	1.21
18	NSSFA119= 4.48	0.94	0.78	1.18
19	NSSFA120= 4.76	0.99	0.79	1.19
20	NSSFA121= 4.92	1.03	0.80	1.20
Total	47.86	Total) 10 = 4.79 % Loop B Average Jet ΔP Pump		
Calc Performed By (Initials): OPE		Independently Verified By (Initials): TRH		

JPM Attachment 1: N2-OSP-LOG-D001 Data Sheet

Provide to Applicant

Item #	Description	Value
1	**2RCS-HC1603A**, RECIRC LOOP A FLOW CONTROL	68%
2	**2RCS-HC1603B**, RECIRC LOOP B FLOW CONTROL	76%
3	**B22-R611A**, RECIRC LOOP 1A SUM JET PMP FLO	52 Mlbm/Hr (Flow Oscillations Minimal)
4	**B22-R611B**, RECIRC LOOP 1B SUM JET PMP FLO	44 Mlbm/Hr (Flow Oscillations Minimal)
5	**B35-R614** RECIRC FLOW LOOP A	42.0 Kgpm (Flow Oscillations Minimal)
6	**B35-R614** RECIRC FLOW LOOP B	41.0 Kgpm (Flow Oscillations Minimal)

7	**Computer Point Indications for Jet Pump Flows**			
	Loop A		**Loop B**	
	Jet pump 1	5.27	Jet pump 11	4.92
	Jet pump 2	4.90	Jet pump 12	4.85
	Jet pump 3	4.95	Jet pump 13	4.40
	Jet pump 4	5.44	Jet pump 14	4.68
	Jet pump 5	6.26	Jet pump 15	5.28
	Jet pump 6	6.15	Jet pump 16	4.52
	Jet pump 7	5.24	Jet pump 17	5.05
	Jet pump 8	5.44	Jet pump 18	4.48
	Jet pump 9	5.06	Jet pump 19	4.76
	Jet pump 10	5.71	Jet pump 20	4.92

JPM Worksheet

Provide to Applicant

Record your results below:	
	Name:
	Summary of Review:
	Summary of Recommended Actions:

Training ID: **Nov 2023 NRC JPM SRO COO2** Revision: **0.0**

Title: **Reactivate SRO Licenses**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mecki Ghabarou	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. OP-AA-105-102, NRC Active License Maintenance
2. NUREG 1123, 2.1.4 (3.8)

Instructor Information

A. JPM Information

1. Description
   a. This JPM tests the operator's ability to evaluate license reactivation documentation.
   b. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.
2. Task Information:
   a. S-ODP-TQS0101-00002, Reactivate an inactive license
3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO ☐ RO ☐ EO ☐ N/A			
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☐ Simulator ☒ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	2.1.4 - Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.			
K/A Importance Rating:	RO	3.3	SRO	3.8

4. Recommended Start Location

   a. Training Classroom

5. JPM Setup (if required)

   a. Provide copy of OP-AA-105-102.

   b. If multiple operators are going to perform the JPM at the same time, ensure each operator has a copy of OP-AA-105-102.

   c. Provide marked up copies of OP-AA-105-102, attachment 2.

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • Due to recent SRO attrition, several SRO licenses are being reactivated. • Two inactive license holders are completing the requirements to reactivate their licenses. • OP-AA-105-102 Attachment 2, Reactivation of License Log, is completed up to the Shift Manager review. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *perform the Shift Manager review of OP-AA-105-102 Attachment 2, Reactivation of License Log, and document results on JPM handout.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.	P	SAT / UNSAT ***STD:*** *OP-AA-105-102 obtained*
3.	Reviews shift hours for SRO #1	P	SAT / UNSAT ***STD:*** *Reviews attachment 2*
4.	Determines SRO #1 does not meet the requirements for license reactivation.	P	***PASS / FAIL** ***STD:*** *Recognize that not all required shift hours were completed in the same calendar quarter, as required in step 4.2.1*
5.	Reviews shift hours for SRO #2	P	SAT / UNSAT ***STD:*** *Reviews attachment 2*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
6.	Determines SRO #2 does not meet the requirements for license reactivation.	P	***PASS / FAIL** ***STD:*** *Recognize that not all required shift hours were completed in a shift position required by technical specifications, per Step 4, Main Body note. STA is a required Staff position, per Tech Specs Section 5.2, however, the STA is not required to be a license holder.*

TASK STANDARD	Shift Manager review of Attachment 2, Reactivation of License Log, is complete recognizing neither SRO meets the requirements for license reactivation.

STOP TIME	

SRO JPM Handout

INITIAL CONDITIONS	Given: • Due to recent SRO attrition, several SRO licenses are being reactivated. • Two inactive license holders are completing the requirements to reactivate their licenses. • OP-AA-105-102 Attachment 2, Reactivation of License Log, is completed up to the Shift Manager review.

INITIATING CUE	**(Operator Name)**, *perform the Shift Manager review of OP-AA-105-102 Attachment 2, Reactivation of License Log, and document results on JPM handout.*

Administrative JPM Handout Reference Packet

SRO JPM Handout

INITIAL CONDITIONS	Given: • Due to recent SRO attrition, several SRO licenses are being reactivated. • Two inactive license holders are completing the requirements to reactivate their licenses. • OP-AA-105-102 Attachment 2, Reactivation of License Log, is completed up to the Shift Manager review. **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *perform the Shift Manager review of OP-AA-105-102 Attachment 2, Reactivation of License Log, and document results on JPM handout.*

Constellation

Training ID: **Nov 2023 NRC JPM SRO EC** Revision: **0.0**

Title: **Explain RPS Operation Using Electrical Prints, Determine Technical Specification Impact of Inoperability**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mecki Ghabarou	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 25 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. Electrical print 807E166TY Sheet 1
2. Electrical print 807E166TY Sheet 2
3. Electrical print 807E166TY Sheet 2a
4. Electrical print 807E166TY Sheet 9
5. Electrical print 807E166TY Sheet 10
6. Electrical print 807E166TY Sheet 11
7. Electrical print 807E166TY Sheet 18
8. Technical Specifications
9. NUREG 1123, 2.2.41 (3.5/3.9)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the operator's ability to use station electrical prints to identify components related to the operation of the high Reactor pressure scram. This JPM also tests their ability to determine the Technical Specification impact of a failed Reactor pressure transmitter and RPS relay.

   b. Critical steps are annotated in the Evaluator standard column with a bolded ***Pass/Fail**.

2. Task Information:

   a. NUREG K/A 2.2.41, Ability to obtain and interpret station electrical and mechanical drawings

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☐ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☐ Simulator ☒ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)	3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	2.2.41 - Ability to obtain and interpret station electrical and mechanical drawings			
K/A Importance Rating:	RO	3.5	SRO	3.9

4. Recommended Start Location

   a. Training Classroom

5. JPM Setup (if required)

   a. Ensure adequate copies of the identified prints are available:

      1) 0007225001003-001 (807E166TY Sheet 1)
      2) 0007225001004-002 (807E166TY Sheet 1)
      3) 0007225001037-02A (807E166TY Sheet 1)
      4) 0007225001011-009 (807E166TY Sheet 9)
      5) 0007225001012-010 (807E166TY Sheet 10)
      6) 0007225001013-011 (807E166TY Sheet 11)
      7) 0007225001020-018 (807E166TY Sheet 18)

   b. Ensure adequate copies of Technical Specifications are available.

B. Read Before Every JPM Performance

For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • Reactor pressure transmitter B22-NO78A has failed upscale. • Investigation reveals that RPS relay K14A has failed to de-energize in response to this condition. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *using the provided station electrical prints, complete the attached worksheet to identify the impacted RPS trip logic channel and determine the Technical Specification impact.* *You may mark the provided prints as desired.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	Identifies the impacted RPS trip logic channel **Note:** See the provided key for assistance in grading JPM steps 2-3.	P	***PASS / FAIL** **STD:** *Identifies RPS trip logic channel A1 is impacted.*
3.	Determines and reports the Technical Specification impact of the given conditions	P	***PASS / FAIL** **STD:** *Reports that Technical Specification 3.3.1.1 Condition A must be entered and requires either placing a channel in trip or placing the trip system in trip within 12 hours (or similar).*

TASK STANDARD	RPS trip logic channel A1 identified. Technical Specification 3.3.1.1 impact of inoperable Reactor pressure instrument determined.

STOP TIME	

Key – Do NOT provide to Operator

Which RPS trip logic channel is impacted by the given conditions?
(circle answer)

(A1) A2 B1 B2

What is the Technical Specification impact of the given conditions?

Technical Specification 3.3.1.1 Condition A must be entered and requires either placing a channel in trip or placing the trip system in trip within 12 hours (or similar).

Key – Do NOT provide to Operator

JPM Handout

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • Reactor pressure transmitter B22-NO78A has failed upscale. • Investigation reveals that RPS relay K14A has failed to de-energize in response to this condition.

INITIATING CUE	**(Operator Name)**, *using the provided station electrical prints, complete the attached worksheet to identify the impacted RPS trip logic channel and determine the Technical Specification impact.* *You may mark the provided prints as desired.*

Worksheet

Which RPS trip logic channel is impacted by the given conditions?
(circle answer)

A1 A2 B1 B2

What is the Technical Specification impact of the given conditions?

Administrative JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant is operating at 100% power. • Reactor pressure transmitter B22-NO78A has failed upscale. • Investigation reveals that RPS relay K14A has failed to de-energize in response to this condition.

INITIATING CUE	**(Operator Name)**, *using the provided station electrical prints, complete the attached worksheet to identify the impacted RPS trip logic channel and determine the Technical Specification impact.* *You may mark the provided prints as desired.*

Worksheet

Which RPS trip logic channel is impacted by the given conditions?
(circle answer)

A1 A2 B1 B2

What is the Technical Specification impact of the given conditions?

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING						TITLE ELEMENTARY DIAGRAM CONT. ON SH.# 2	807E166TY SH.NO.1
APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS FRACTIONS	DECIMALS	ANGLES		REACTOR PROTECTION SYSTEM	
	√	± -	± -	± -		FIRST MADE FOR NMP-2 238X173AF	

GENERAL ELECTRIC
MPL NO. (C72-1050)

LEGEND CONT:

DX RPS — DIV. X RPS.

AX RPS — ASSOCIATED WITH AND TREATED AS DIV X RPS

SSX SPECIAL SOLENOID POWER FOR RPS SOLENOID SEE NOTE 27.

LEGEND:

LOCATION USED ONLY WHEN NECESSARY FOR CLARITY

XXXX — PART NUMBER OR TERMINAL NUMBER

SYSTEM NUMBER OR TERMINAL BOARD IDENT

X — MATCH NUMBER

XXX — ZONE

SHEET NUMBER

MATCH CIRCLE

LAST NUMBER USED - 12

INDICATES GROUND BUS

INDICATES SIGNAL COMMON

UPS UNINTERRUPTABLE POWER SOURCE

DX ESF — CLASS 1E, DIV X, NUCLEAR SAFETY RELATED

AX ESF — ASSOCIATED WITH AND TREATED AS CLASS 1E, DIV X

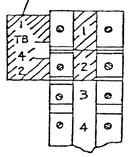
N — NON-DIVISIONAL

CXXXX COMPUTER INPUT

XXX = INDICATES PGCC LINE CODES

V — ENGINEERING TEST AND INFORMATION SYSTEM (ETIS)

COLOR RED - SHS 5,6,7,&8



SHORTING LINK

FIGURE 1

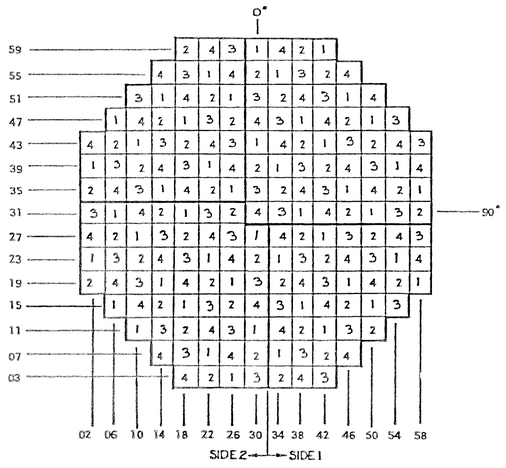


FIG. 9

HYDRAULIC CONTROL UNITS REACTOR SIDE 2			
GROUP 3	GROUP 2	GROUP 1	GROUP 4
02-51	02-19	02-23	02-27
06-23	02-35	02-39	02-43
06-39	06-27	06-15	06-19
10-19	06-43	06-31	06-35
10-35	10-23	06-47	10-15
10-51	10-39	10-11	10-31
14-11	14-15	10-27	10-47
14-27	14-31	10-43	14-07
14-43	14-47	14-19	14-23
18-07	18-11	14-35	14-39
18-23	18-27	14-51	14-55
18-39	18-43	18-15	18-03
18-55	18-59	18-31	18-19
22-15	22-03	18-47	18-35
22-31	22-19	22-07	18-51
22-47	22-35	22-23	22-11
26-11	22-51	22-39	22-27
26-27	26-15	22-55	22-43
26-43	26-31	26-03	22-59
26-59	26-47	26-19	26-07
30-03	30-07	26-35	26-23
30-19	30-23	26-51	26-39
		30-11	26-55
		30-27	30-15

HYDRAULIC CONTROL UNITS REACTOR SIDE 1			
GROUP 3	GROUP 2	GROUP 1	GROUP 4
30-35	30-39	30-43	30-31
30-51	30-55	30-59	30-47
34-15	34-03	34-07	34-11
34-31	34-19	34-23	34-27
34-47	34-35	34-39	34-43
38-07	34-51	34-55	34-59
38-23	38-11	38-15	38-03
38-39	38-27	38-31	38-19
38-55	38-43	38-47	38-35
42-03	38-59	42-11	38-51
42-19	42-07	42-27	42-15
42-35	42-23	42-43	42-31
42-51	42-39	42-59	42-47
46-11	42-55	46-19	46-07
46-27	46-15	46-35	46-23
46-43	46-31	46-51	46-39
50-23	46-47	50-15	46-55
50-39	50-11	50-31	50-19
54-15	50-27	50-47	50-35
54-31	50-43	54-23	50-51
54-47	54-19	54-39	54-27
58-27	54-35	58-19	54-43
58-43	58-31	58-35	58-23
			58-39

TABLE 1

VALVE AND CONTROL TABULATION							
DEVICE		SWITCH	INDICATOR LAMPS			LOCATION	
REF DESIG	FUNCTION		RED	GREEN	OTHER	MTG LOC	SH
------	REACTOR SCRAM TRIP LOGIC A1	---	---	---	X(WHT)	H13-P609	9
------	REACTOR SCRAM TRIP LOGIC B1	---	---	---	X(WHT)	H13-P611	9
C12-F010	SCRAM DISCHARGE VOLUME VENT VALVE	---	X	X	---	H13-P603	11
C12-F011	SCRAM DISCHARGE VOLUME DRAIN VALVE	---	X	X	---	H13-P603	11
------	TRIP ACTUATOR LOGIC "A" GP1	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "A" GP2	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "C" GP3	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "C" GP4	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "B" GP1	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "B" GP2	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "D" GP3	---	---	---	X(WHT)	H13-P603	10
------	TRIP ACTUATOR LOGIC "D" GP4	---	---	---	X(WHT)	H13-P603	10
C12-F009	SCRAM DISCHARGE VENT & DRAIN	52A	---	---	---	H13-P603	5
C12-F182	SCRAM DISCHARGE VENT & DRAIN	52B	---	---	---	H13-P603	6
C12-F110A	BACK-UP SCRAM VALVE	---	---	---	---	-------	11
C12-F110B	BACK-UP SCRAM VALVE	---	---	---	---	-------	11
------	REACTOR SCRAM TRIP LOGIC A2	---	---	---	X(WHT)	H13-P609	9
------	REACTOR SCRAM TRIP LOGIC B2	---	---	---	X(WHT)	H13-P611	9
------	RECIRC PUMP TRIP SYSTEM "A"	---	X	---	X(WHT)	H13-P609	11
		---	---	---	X(WHT)	H13-P611	11
------	RECIRC PUMP TRIP SYSTEM "B"	---	X	---	X(WHT)	H13-P609	11
		---	---	---	X(WHT)	H13-P611	11
------	SYSTEM OUT OF SERVICE	98A	---	---	X(AMB)	H13-P603	14
------	SYSTEM OUT OF SERVICE	98B	---	---	X(AMB)	H13-P603	14
------	TRIP UNIT IN CAL OR GROSS FAILURE	---	---	---	4X (AMB)	H13-P603	14
------	24VDC PWR DIV1	---	---	---	X(WHT)	H13-P609	16
------	24VDC PWR DIV2	---	---	---	X(WHT)	H13-P611	16
------	24VDC PWR DIV3	---	---	---	X(WHT)	H13-P609	16
------	24VDC PWR DIV4	---	---	---	X(WHT)	H13-P611	16
C12-F180	SCRAM DISCHARGE VENT VALVE	---	X	X	---	H13-P603	11
C12-F181	SCRAM DISCHARGE VENT VALVE	---	X	X	---	H13-P603	11
------	RPT SYSTEM OUT OF SERVICE	99A	---	---	X(AMB)	H13-P602	11,14
------	RPT SYSTEM OUT OF SERVICE	99B	---	---	X(AMB)	H13-P602	11,14

REFERENCE DOCUMENTS:

1.	C12-1010	CONTROL ROD DRIVE HYD SYS P&ID
2.	C72-1010	REACTOR PROTECTION SYS 1ED
3.	B21-1010	NUCLEAR BOILER P&ID
4.	D13-1050	PROCESS RADIATION MON SYS ELEM DIAG (D13A)
5.	C51-1080	POWER RANGE NEUT MON SYS ELEM DIAG (C51B)
6.	C51-1070	START-UP RANGE NEUT MON SYS ELEM DIAG (C51A)
7.	B21-1090	NUC STM SUPPLY SHUTOFF SYS ELEM DIAG (B21H)
8.	C12-1050	REACTOR MANUAL CONTROL SYS ELEM DIAG (C12A)
9.	C72-1050	RPS MG SET CONTROL SYSTEM ELEM DIAG (C72B)
10.	C72-1070	RPS SEPARATION SCHEME ELEM DIAG (C72C)
11.	C72-4010	REACTOR PROTECTION SYS - DESIGN SPEC
12.	A62-4050	ELECT. EQUIP SEP SAFE. SYS - DESIGN SPEC
13.	C91-4030	PERFORMANCE MONITORING SYSTEM I/O LIST
14.	B35-1030	REACTOR RECIRCULATION SYS ELEM DIAG (B35A)
15.	E31-1050	LEAK DETECTION SYSTEM ELEM DIAG
16.	G33-1050	REACTOR WTR CLEANUP SYS ELEM DIAG (G33A)
17.	DELETED	
18.	S&W DWG	12177-ESK-7SC1 SERIES OFF-NORMAL STATUS DISPLAY
19.	EPG DWG	2538-E-1101 & 2540-E-1101 MSIV ELEM DIAG
20.	C88-1050	ENGINEERING TEST AND INFORMATION SYSTEM ELEMENTARY DIAG (C88A) (10SX----)
21.	0007510001310	ELEMENTARY DIAGRAM, HYDROGEN WATER CHEMISTRY INJECTION SYSTEM
22.	EC-138	VERIFICATION OF SIZES OF CABLES ASSOCIATED WITH Y-PATTERN MAIN STEAM ISOLATION VALVES

NOTES CONT

24 RELAY ISOLATION DEVICE A METAL BARRIER SHALL BE PROVIDED TO SEPARATE THE RELAY CONTACTS AND THEIR WIRING FROM THE COIL AND ITS WIRING.

27. SEE REF. 12 FOR SPECIAL SEPARATION REQUIREMENTS OF SSX POWER TO SOLENOID.

26 COMPUTER POINTS TO BE RETRANSMITTED TO USS COMPUTER PER REF. 13.

29 THE CABLES SELECTED TO CONNECT THE SCRAM CONTACTORS TO THE HCU(S) SHALL PROVIDE MINIMUM VOLTAGE DROP WITH NOT LESS THAN 105 VAC MEASURED ACROSS THE SCRAM SOLENOID COIL LEADS AND SHALL BE PULL THRU CABLES IN CONDUIT

30 RECOMMEND 6AWG WIRE PULL-THRU TO HCU BY OTHERS TO PREVENT EXCESSIVE VOLTAGE DROP

31 RECOMMEND 4AWG WIREPULL-THRU TO HCU BY OTHERS TO PREVENT EXCESSIVE VOLTAGE DROP

32 RECOMMEND 8AWG WIRE PULL-THRU TO HCU BY OTHERS TO PREVENT EXCESSIVE VOLTAGE DROP

33 RECOMMEND 2AWG WIRE PULL-THRU TO HCU BY OTHERS TO PREVENT EXCESSIVE VOLTAGE DROP

34 SOE RELAY CONTACT TO BE USED AS AN INPUT TO OSCILLOGRAPH 25PI-OSC001. SEE AE DWG 12177-ESK-115PI01.

35 REMOTE SHUTDOWN SYSTEM ESK-11RSS12 & 13 ARE AE RESPONSIBILITY. AE TO PROVIDE PROPER SEPARATION BETWEEN RPS DIVISIONS AND/OR RPS AND OTHER WIRING.

36 IDENTIFY WIRES AT PANEL TERMINAL BOARD INTERFACE END, AND ROUTE PGCC CABLES IN SEPARATELY IDENTIFIED CONDUITS AS FOLLOWS.

A) RPS-B (GREEN/WHITE) USE GREEN TAPE WITH WHITE RPS-B LETTERS. MAY BE ROUTED AND BUNDLED WITH DIVISION 1 RPS.

B) RPS-A (YELLOW/WHITE) USE YELLOW TAPE WITH BLACK RPS-A LETTERS. MAY BE ROUTED AND BUNDLED WITH DIVISION 2 RPS.

C) B-RPS (ORANGE/WHITE) USE ORANGE TAPE WITH BLACK B-RPS LETTERS. MAY BE ROUTED AND BUNDLED WITH DIVISION 3 RPS.

D) A-RPS (BLUE/WHITE) USE BLUE TAPE WITH WHITE A-RPS LETTERS. MAY BE ROUTED AND BUNDLED WITH DIVISION 4 RPS.

37 THESE SIGNALS ARE RETRANSMITTED ONLY ONCE TO THE PROCESS COMPUTER SYSTEM.

38 COMPUTER POINTS C1539 AND C1540 ARE USED TO FORM A COMPOSITE SCRAM SIGNAL TO GETARS. (REF. SWEC ESK-10SXS14)

39 THE OFF-NORMAL STATUS DISPLAY SYSTEM HAS BEEN ABANDONED IN PLACE. THIS CONTACT NO LONGER PROVIDES A FUNCTION.

40 THE OFF-NORMAL STATUS DISPLAY SYSTEM HAS BEEN ABANDONED IN PLACE. THIS RELAY AND IT'S CONTACTS NO LONGER PROVIDE A FUNCTION.

INDEX

SH.	CONTENTS
1.	NOTES, REFERENCES, LEGEND, VALVE TAB. & ROD SCRAM GROUP ASSIGNMENTS
2,2A	RELAY TABULATIONS
3.	SWITCH DEVELOPMENT
4,4A	POWER DISTRIBUTION
5.	CHANNEL "A1" SENSOR RELAYS
6.	CHANNEL "B1" SENSOR RELAYS
7.	CHANNEL "A2" SENSOR RELAYS
8.	CHANNEL "B2" SENSOR RELAYS
9	CHANNEL A1,B1,A2,B2 SCRAM TRIP LOGIC
10.	SCRAM SOLENOIDS
11.	SCRAM DISCHARGE VOLUME ISOLATION VALVE & BACK-UP SCRAM VALVES OFF-GAS & VACUUM PUMP LOGIC
12.	ANNUNCIATOR INPUTS, COMPUTER INPUTS
13.	ANNUNCIATOR INPUTS, COMPUTER INPUTS
14	ANNUNCIATOR INPUTS, STATUS LIGHTS
14A.	ANNUNCIATOR INPUTS, COMPUTER INPUTS
15	COMPUTER INPUTS (VOID)
16	TESTABILITY POWER SUPPLIES
17	TESTABILITY CARD FILE TABULATION
18	TESTABILITY
19	SCRAM DISCHARGE VOLUME

NOTES:

1. ALL CABLE AND PANEL WIRING TO BE SEPARATED PER REF DOC 12.
2. ANNUNCIATORS TRIP ON CIRCUIT CLOSURE.
3. PARTS (DS-2A THRU DS-2H) ARE LONG LIFE 55V INDICATOR ASSEMBLIES
4. WIRING FROM DUPLICATE SENSORS ON A COMMON PROCESS TAP MUST COMPLY WITH REF. DOC 12, PARA 4.3.1.2.
5. FOR SEPARATION REQUIREMENTS WITHIN PENETRATION SEE REF DOC 12, PARA 4.3.1.5
6. ALL IND LIGHTS ARE GE TYPE ET16 AND SUPPLIED WITH PANEL, EXCEPT THE LIGHTS FOR THE RPT LOGIC WHICH ARE ET17 TYPE.
7. (XX-XX) THE CONTROL ROD DESIGNATION SYMBOL IS USED WHERE MORE THAN ONE ROD IS REPRESENTED.
8. PROVIDE PHYSICAL ISOLATION AND SEPARATION OF WIRING AND DEVICES AS APPROPRIATE TO PRECLUDE THE POSSIBILITY OF A SINGLE COMPONENT FAILURE (CLOSURE, OPENING, OR SHORT) FROM PREVENTING AUTOMATIC SCRAM. SEE REF DOC 10 AND 12.
9. ALL RPS WIRING IN H13-P603 SHALL BE RUN TO TB'S X, XX, Y, XY DEPENDING ON THEIR DIVISION. ALL RPS WIRING IN H13-P603 SHALL BE RUN IN GROUNDED CONDUIT. DO NOT RUN DIFFERENT DIVISIONS IN THE SAME CONDUIT.
10. UNLESS OTHERWISE INDICATED, THE FOLLOWING REFERENCE DESIGNATIONS SHOWN ON THIS DIAGRAM ARE PREFIXED WITH C72A-.

REF DESIG	NAME	REF DESIG	NAME
KXX	RELAY	ZXX	CARD FILE
SXX	SWITCH	ZXX-1	CALIBRATION UNIT
FXX	FUSE	ATXX	ISOLATOR
CBXX	CIRCUIT BREAKER	JBXX	JUNCTION BOX
TBXX	TERMINAL	DSXX	LIGHT
CRXX	DIODE	WXXX	CABLE
KXXX	PWR SUP		

11. THE COMPONENTS SHALL BE ENCLOSED IN METALLIC BOXES AND INTERCONNECTING CABLES ROUTED IN METALLIC CONDUIT IF 6 INCH SEPARATION CANNOT BE MAINTAINED.
12. THE LOADS SHOWN ARE ESTIMATED, NOT MEASURED QUANTITIES. AC LOADS ARE BASED ON 60 HZ. (SS = STEADY STATE; T = TRANSIENT)
13. LIGHTS CAPABLE OF 20% OVER-VOLTAGE.
14. INDIVIDUAL GROUND WIRES TO BE RUN FROM TERMINALS 3 AND 4 TO COMMON BUS BAR. TERMINALS 1,2,5,6,7 FOR EACH TERMINAL BOARD ARE PREWIRED.
15. THE SCRAM CONTACTORS SHALL BE ENCLOSED IN METAL BOXES AND INTERCONNECTING CABLES ROUTED IN METALLIC CONDUIT. SCRAM SOLENOID A AND B WIRING FOR ANY GIVEN GROUP MAY BE ROUTED TOGETHER IN SEPARATE CONDUIT WITH NO OTHER POWER WIRING FROM RPS MG SET, OR ANY OTHER POWER SOURCE.
16. DELETED
17. STATUS LIGHTS ARE MASTER SPECIALTY CO. 10H SERIES WITH INTEGRAL SWITCH (120V NEON LAMPS)
18. ESSENTIAL POWER WIRING (125V BUS A OR B) WITHIN THE CONTROL ROOM PANELS, DUCTS OR PLANT WIREWAYS SHALL BE SEPARATED FROM RPS POWER WIRING IN THE SAME DIVISION BY 6 INCHES OR IT SHALL BE ROUTED IN GROUNDED METALLIC CONDUIT.
19. THIS IS NON-ESSENTIAL (RPS MG AND UPS) WIRING IN PGCC DUCTS AND PLANT WIREWAYS CONTAINING ESSENTIAL POWER WIRING OF THE SAME DIVISION. THIS WIRING SHALL BE ROUTED IN GROUNDED METAL CONDUITS.
20. DELETED
21. ANNUNCIATOR ALARMS ON OPEN CIRCUIT.
22. SOE POINTS TO BE RETRANSMITTED TO BOP COMPUTER PER AE DWG # 21HC-6 AND TO NSS COMPUTER PER REF. 13.
23. DELETED.
24. LOSS OF SINGLE POWER SOURCE AT RPS SYSTEM SHALL NOT CAUSE OR PREVENT SCRAM.
25. 125 VDC BUS A WIRING SHALL BE SEPARATED FROM ALL OTHER 125 VDC BUS B WIRING IN THE CONTROL ROOM PANELS BY AT LEAST 6 INCHES OR RUN IN METALLIC CONDUIT.

NOTES:
CONT. (SH.1 ZONE D9)

NOTE: THIS DRAWING WAS CREATED USING COMPUTER AIDED DRAFTING. DO NOT MAKE MANUAL CHANGES

NMP2-VENDOR DWG DATA STAMP	
SUPERSEDES SWEC FILE NO.	0007.225-001-003Y
EQUIP. ID.	SEE CDS LISTING
MFG. NAME	G.E.
MFG. CODE	G08000
SPEC. and/or P.O.	P800A
SWEC JOB NO.	12177

DRAWING NO. 807E166TY, SH. 1

LINE CODES:
THIS SHEET: NONE
LAST LINE CODE USED: 251

ExelonGeneration. UNIT 2
Nine Mile Point Nuclear Station
ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

REV	DATE	DESCRIPTION OF REVISION	PREP	CHKD	ENGR			ELEC	MECH	STR
6	SEE FCMS	INCORP. ECN# ECP-15-000434-CN-072 0007225001003-001-05.00	SEE FCMS	SEE FCMS	SEE FCMS			N/A	N/A	N/A

INDEX	SCALE	DRAWING NO	SHEET
N/A	NONE	0007.225-001-003	N/A

0007.225-001-004 (AS DRAWN)

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING APPLIED PRACTICES: SURFACES ✓; TOLERANCES FRACTIONS ±, DECIMALS ±, ANGLES °

TITLE ELEM. DIAG. REACTOR PROTECTION SYSTEM
FIRST MADE FOR NMP-2

AGASTAT TYPE GP RELAY TABULATION (H13-P609)

B1 B4	T1 M1 R1	T2 M2 R2	T3 M3 R3	T4 M4 R4
K51A SH 5	SEE SH 2A	SEE SH.2A	SEE SH 2A	SEE SH 2A
K51J SH.5	SEE SH 2A	SP	SP	SP
K4A SH. 18	K14A SH. 9	---	SP	SP	---	ANN SH. 12	SEE SH.2	SP
K5A SH. 18	K14A SH. 9	---	SP	SP	---	ANN SH. 12	SP	SP
K6A SH. 18	K14A SH. 9	---	SP	SP	---	ANN SH. 12	SEE SH. 2	SP
K9A SH. 18	K14A SH. 9	---	SP	SP	ANN SH 12	---	SP	SP
K11A SH. 5	K14A SH. 9	---	SEE REF 7 SH.5	---	ANN. SH 13	---	SP	SP
K13A SH. 5	K15A SH. 5	---	SP	SP	SP	SP	SP	SP
K15A SH. 5	K14A SH. 9	---	---	ETIS SH 5	---	ANN SH. 13	SP	SP
K18A SH. 5	K14A SH. 9	---	SP	SP	ANN SH. 13	---	SP	SP
K19A SH. 5	K14A SH. 9	---	SP	SP	SP	SP	SP	SP
K70A SH. 9	---	K19A SH 5	K22A SH 5	---	---	C12-F009 F182 SH 5	SP	SP
K21A SH. 11	C12-F110A SH. 11	---	C12-F110A SH. 11	---	SP	SP	SP	SP
K38 SH. 18	ANN SH 14	---	---	SP	SP	SP	SP	SP
K39A SH.18	ANN SH. 14	---	STATUS LT SH. 14	---	SP	SP	SP	SP
K40A SH.18	---	ANN SH. 14	SP	SP	SP	SP	SP	SP
K45A SH. 18	SEE SH. 2	---	SEE SH 2	---	SP	SP	SP	SP
K36 SH. 18	---	ANN SH 13	SP	SP	SP	SP	SP	SP
K37 SH. 18	---	ANN SH 13	SP	SP	SP	SP	SP	SP
K46A SH. 18	SEE SH. 2	---	SEE SH 2	---	SP	SP	SP	SP
K51F SH.6	SEE SH.2A	---	SEE SH.2A	---	SEE SH.2A	---	SEE SH.2A	---
K51P SH.6	SEE SH 2A	---	SP	SP	SP	SP	SP	SP

GE TYPE CR2820 RELAY TABULATION (TDPU-AC FORM) (H13-P609)

L1 L2	AT1 AT2 TD	BT1 BT2 TD
K22A SH. 5	K19A SH. 5	SP

GE TYPE CR2820 RELAY TABULATION (TDPU-AC FORM) (H13-P611)

L1 L2	AT1 AT2 TD	BT1 BT2 TD
K22B SH. 6	K19B SH. 6	SP

T4 M4 K4A — A B K72C — K L K7A — K L K7C

T4 M4 K6A — A B K73C

M1 T1 K45A — M1 T1 K46A — A B K74C — A B K75C

INTERLOCKS TO NUC STM SUPPLY SHUTOFF SYS ELEM DIAG REF 7

A B K79A — A B K79B

INTERLOCKS TO REACTOR MANUAL CONTROL SYS ELEM DIAG REF 8

AGASTAT TYPE GP RELAY TABULATION (H13-P609)

B1 B4	T1 M1 R1	T2 M2 R2	T3 M3 R3	T4 M4 R4
K4C SH. 18	K14C SH. 9	---	SP	SP	---	ANN SH. 12	SP	SP
K5C SH.18	K14C SH. 9	---	SP	SP	---	ANN SH. 12	SP	SP
K6C SH.18	K14C SH. 9	---	SP	SP	---	ANN SH. 12	SP	SP
K9C SH. 18	K14C SH. 9	---	SP	SP	ANN SH. 12	---	SP	SP
K11C SH. 7	K14C SH. 9	---	SEE REF 7 SH 5	---	ANN. SH 13	---	SP	SP
K13C SH. 7	K15C SH. 7	---	SP	SP	SP	SP	SP	SP
K15C SH. 7	K14C SH. 9	---	---	ETIS SH. 7	---	ANN SH. 13	SP	SP
K18C SH. 7	K14C SH. 9	---	SP	SP	ANN SH. 13	---	SP	SP
K19C SH. 7	K14C SH. 9	---	SP	SP	SP	SP	SP	SP
K70C SH. 9	---	K19C SH 7	K22C SH 7	---	SP	SP	SP	SP
K40C SH. 18	---	ANN SH14	SP	SP	SP	SP	SP	SP
K39C SH. 18	ANN SH. 14	---	STATUS LT SH. 14	---	SP	SP	SP	SP

GE TYPE CR2820 RELAY TABULATION (TDPU-AC FORM) (H13-P609)

L1 L2	AT1 AT2 TD	BT1 BT2 TD
K22C SH. 7	K19C SH. 7	SP

GE TYPE CR2820 RELAY TABULATION (TDPU-AC FORM) (H13-P611)

L1 L2	AT1 AT2 TD	BT1 BT2 TD
K22D SH. 8	K19D SH. 8	SP

GE TYPE CR205 CONTACTOR TABULATION (H13-P609)

1 2	1L 1T	2L 2T	3L 3T	3 4	5 6	7 8	9 10
K14A SH. 9	GROUP 1 SH. 10	K71A SH. 9	K14A SH. 9	ETIS SH. 9 ▲	ANN SH. 13 ▲	DS1A SH. 9	K70A SH. 9 ▲
K14E SH. 9	GROUP 1 SH. 10	SP	K71A SH. 9	SP	SP	SP	K70A SH. 9 ▲
K14J SH. 9	GROUP 1 SH. 10	SP	K71A SH. 9	SP	SP	SP	K70A SH. 9 ▲
K14N SH. 9	GROUP 1 SH. 10	SP	K71A SH. 9	SP	SP	SP	K70A SH. 9 ▲

▲ CONTACTS TO BE NORMALLY CLOSED.

GE TYPE CR205 CONTACTOR TABULATION (H13-P611)

1 2	1L 1T	2L 2T	3L 3T	3 4	5 6	7 8	9 10
K14B SH. 9	GROUP 2 SH. 10	K71B SH. 9	K14B SH. 9	ETIS SH. 9 ▲	ANN. SH. 13 ▲	DS1B SH. 9	K70B SH. 9 ▲
K14F SH. 9	GROUP 2 SH. 10	SP	K71B SH. 9	SP	SP	SP	K70B SH. 9 ▲
K14K SH. 9	GROUP 2 SH. 10	SP	K71B SH. 9	SP	SP	SP	K70B SH. 9 ▲
K14P SH. 9	GROUP 2 SH. 10	SP	K71B SH. 9	SP	SP	SP	K70B SH. 9 ▲

▲ CONTACTS TO BE NORMALLY CLOSED.

GE TYPE CR205 CONTACTOR TABULATION (H13-P609)

1 2	1L 1T	2L 2T	3L 3T	3 4	5 6	7 8	9 10
K14C SH. 9	GROUP 3 SH. 10	SP	K14C SH. 9	ETIS SH. 9 ▲	ANN. SH. 13 ▲	DS1C SH. 9	K70C SH. 9 ▲
K14G SH. 9	GROUP 3 SH. 10	SP	SP	SP	SP	SP	K70C SH. 9 ▲
K14L SH. 9	GROUP 3 SH. 10	SP	SP	SP	SP	SP	K70C SH. 9 ▲
K14R SH. 9	GROUP 3 SH. 10	SP	SP	SP	SP	SP	K70C SH. 9 ▲

▲ CONTACTS TO BE NORMALLY CLOSED.

GE TYPE CR205 CONTACTOR TABULATION (H13-P611)

1 2	1L 1T	2L 2T	3L 3T	3 4	5 6	7 8	9 10
K14D SH. 9	GROUP 4 SH. 10	SP	K14D SH. 9	ETIS SH. 9 ▲	ANN. SH. 13 ▲	DS1D SH. 9	K70D SH. 9 ▲
K14H SH. 9	GROUP 4 SH. 10	SP	SP	SP	SP	SP	K70D SH. 9 ▲
K14M SH. 9	GROUP 4 SH. 10	SP	SP	SP	SP	SP	K70D SH. 9 ▲
K14S SH. 9	GROUP 4 SH. 10	SP	SP	SP	SP	SP	K70D SH. 9 ▲

▲ CONTACTS TO BE NORMALLY CLOSED.

AGASTAT TYPE GP RELAY TABULATION (H13-P611)

B1 B4	T1 M1 R1	T2 M2 R2	T3 M3 R3	T4 M4 R4
K4B SH. 18	K14B SH. 9	---	SP	SP	---	ANN SH. 12	SEE SH. 2	---
K5B SH. 18	K14B SH. 9	---	SP	SP	---	ANN SH. 12	SP	SP
K6B SH. 18	K14B SH. 9	---	SP	SP	---	ANN SH. 12	SEE SH. 2	---
K39B SH. 18	ANN SH. 14	---	STATUS LT SH. 14	---	SP	SP	SP	SP
K9B SH. 18	K14B SH. 9	---	SP	SP	ANN SH.12	---	SP	SP
K11B SH. 6	K14B SH. 9	---	SEE REF 7 SH.6	---	ANN. SH 13	---	SP	SP
K13B SH. 6	K15B SH. 6	---	SP	SP	SP	SP	SP	SP
K15B SH. 6	K14B SH. 9	---	---	ETIS SH 6	---	ANN SH. 13	SP	SP
K18B SH. 6	K14B SH. 9	---	---	SP	ANN SH. 13	---	SP	SP
K19B SH. 6	K14B SH. 9	---	SP	SP	SP	SP	SP	SP
K70B SH. 9	---	K19B SH. 6	K22B SH. 6	---	---	C12-F009 F182 SH. 6	SP	SP
K21B SH. 11	C12-F110B SH. 11	---	C12-F110B SH. 11	---	SP	SP	SP	SP
K40B SH. 18	---	ANN SHH	SP	SP	SP	SP	SP	SP
K45B SH. 18	SEE SH. 2	---	SEE SH 2	---	SP	SP	SP	SP
K46B SH. 18	SEE SH. 2	---	SEE SH 2	---	SP	SP	SP	SP

T4 M4 K4B — A B K72D — K L K7B — K L K7D

T4 M4 K6B — A B K73D

T1 M1 K45B — T1 K1 K46B — A B K74D — A B K75D

INTERLOCKS TO NUC STM SUPPLY SHUTOFF SYS ELEM DIAG REF 7

GE TYPE HMA RELAY TABULATION (H13-P609)

5 6	1 7 3	2 8 4
K47A SH 11	E51A-AT8 SH.11	---	CR 3A,3B SH. 11	---

PANEL H13-P611

K47B SH. 11	B22C-AT8 SH.11	---	CR4A,4B SH. 11	---

GE TYPE CR2820 TIME DELAY PU RELAY TABULATION (H13-P609)

L1 L2	1AT1 1AT2 TD	1BT1 1BT2 TD	2AT1 2AT2 TD	2BT1 2BT2 TD
K16A SH. 5	K15A SH. 5	SP	ANN. SH 13	SP

GE TYPE CR2820 TIME DELAY PU RELAY TABULATION (H13-P611)

L1 L2	1AT1 1AT2 TD	1BT1 1BT2 TD	2AT1 2AT2 TD	2BT1 2BT2 TD
K16B SH. 6	K15B SH. 6	SP	ANN. SH 13	SP

GE TYPE CR2820 TIME DELAY PU RELAY TABULATION (H13-P609)

L1 L2	1AT1 1AT2 TD	1BT1 1BT2 TD	2AT1 2AT2 TD	2BT1 2BT2 TD
K16C SH. 7	K15C SH. 7	SP	ANN. SH 13	SP

GE TYPE CR2820 TIME DELAY RELAY TABULATION (H13-P611)

L1 L2	1AT1 1AT2 TD	1BT1 1BT2 TD	2AT1 2AT2 TD	2BT1 2BT2 TD
K16D SH. 8	K15D SH. 8	SP	ANN. SH 13	SP

T2 M2 K45A — T2 M2 K46A — D E K74C — D E K75C

INTERLOCKS TO REMOTE SHUTDOWN SYS ELEM DIAG BY OTHERS SEE SH.14A FOR INTERCONNECTION DETAILS (35)

T2 M2 K45B — T2 M2 K46B — D E K74D — D E K75D

INTERLOCKS TO REMOTE SHUTDOWN SYS ELEM DIAG BY OTHERS SEE SH. 14A FOR INTERCONNECTION DETAILS (35)

ORIGINAL

SUPERSEDES SWEC FILE NO.	0007.225-001-004L
EQUIP. ID.	SEE CDS LISTING
MFG. NAME	G.E.
MFG. CODE	08000
SPEC. and/or P.O.	P800A
SWEC JOB NO.	12177

DRAWING NO. 807E166TY SH. 2

LINE CODES: THIS SHEET: NONE

NOTE: THIS DRAWING WAS CREATED USING COMPUTER GRAPHICS AND CONTAINS NO MANUAL CHANGES

NUCLEAR SAFETY RELATED

NIAGARA MOHAWK — NINE MILE POINT NUCLEAR STATION - UNIT 2 SCRIBA, N.Y.

ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

REV	DATE	BY	DESCRIPTION	CK	APP
2	5-12-90	FOC	REVISED PER DOC 2E113836		

SCALE NONE — DRAWING NO 0007.225-001-004

AGASTAT TYPE GP RELAY TABULATION (H13-P609)

B1 B4	T1 M1 R1		T2 M2 R2		T3 M3 R3		T4 M4 R4	
K58 SH11	VLV POS IND SH11	SP	ANN SH14A	SP	SP	SP	SP	SP
K52 SH11	VLV POS IND SH11	SP	ANN SH14A	SP	SP	SP	SP	SP
K53 SH11	VLV POS IND SH11	SP	ANN SH14A	SP	SP	SP	SP	SP
K54 SH11	VLV POS IND SH11	SP	ANN SH14A	SP	SP	SP	SP	SP

AGASTAT TYPE GP RELAY TABULATION (H13-P609)

B1 B4	T1 M1 R1		T2 M2 R2		T3 M3 R3		T4 M4 R4	
K55C SH19	K14C SH9	SP	SP	SP	SP	ANN SH12	SP	SP
K56C SH19	ANN SH14A	SP	SP	SP	SP	SP	SP	SP
K57C SH19	SP	ANN SH14A	SP	SP	SP	SP	SP	SP

AGASTAT TYPE GP RELAY TABULATION (H13-P609)

B1 B4	T1 M1 R1		T2 M2 R2		T3 M3 R3		T4 M4 R4	
K55A SH19	K14A SH9	SP	SP	SP	SP	ANN SH12	SP	SP
K56A SH19	ANN SH14A	SP	SP	SP	SP	SP	SP	SP
K57A SH19	SP	ANN SH14A	SP	SP	SP	SP	SP	SP

POTTER & BRUMFIELD MDR TABULATION (H13-P609)

1 2	A B C		D E F		G H J		K L M	
K81 SH11	SEE SH11	----	SEE SH.11	----	SP	SP	SP	SP
K82 SH11	SEE SH11	----	SEE SH 11	----	SP	SP	SP	SP
K83 SH11	SEE SH11	----	SEE SH 11	----	SP	SP	SP	SP
K84 SH11	SEE SH11	----	SEE SH 11	----	SP	SP	SP	SP

AGASTAT TYPE GP RELAY TABULATION (H13-P611)

B1 B4	T1 M1 R1		T2 M2 R2		T3 M3 R3		T4 M4 R4	
K55D SH19	K14D SH9	SP	SP	SP	SP	ANN SH12	SP	SP
K56D SH19	ANN SH14A	SP	SP	SP	SP	SP	SP	SP
K57D SH19	SP	ANN SH14A	SP	SP	SP	SP	SP	SP

AGASTAT TYPE GP RELAY TABULATION (H13-P611)

B1 B4	T1 M1 R1		T2 M2 R2		T3 M3 R3		T4 M4 R4	
K55B SH19	K14B SH9	SP	SP	SP	SP	ANN SH12	SP	SP
K56B SH19	ANN SH14A	SP	SP	SP	SP	SP	SP	SP
K57B SH19	SP	ANN SH14A	SP	SP	SP	SP	SP	SP

POTTER & BRUMFIELD MDR TABULATION (H13-P611)

1 2	A B C		D E F		G H J		K L M	
K1D SH8	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
K3D SH8	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
K3H SH8	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
K7D SH8	K14D SH9	----	SP	SP	----	ANN SH12	SEE SH2	----
K8D SH 8	K14D SH9	----	SP	SP	SP	SP	----	ANN Sh 12
K10D SH 8	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
K10H SH.8	K14D SH9	----	----	SEE, REF 7 SH6	----	ANN SH12	SP	SP
K12D SH 8	K14D SH9	----	SP	SP	----	ANN SH13	K13D SH 8	----
K12H SH 8	K14D SH9	----	SP	SP	----	ANN SH13	K13D SH 8	----

POTTER & BRUMFIELD MDR TABULATION (H13-P609)

1 2	A B C		D E F		G H J		K L M	
K1A SH5	K14A SH9	----	SP	SP	----	ANN SH12	SP	SP
K3A SH5	K14A SH9	----	SP	SP	----	ANN SH12	SP	SP
K3E SH5	K14A SH9	----	SP	SP	----	ANN SH12	SP	SP
K7A SH5	K14A SH9	----	SP	SP	----	ANN SH12	SEE SH 2	----
K8A SH5	K14A SH9	----	SP	SP	SP	SP	----	ANN SH12
K10A SH5	K14A SH9	----	----	SEE REF 7 SH5	----	ANN SH12	SP	SP
K10E SH5	K14A SH9	----	SP	SP	----	ANN SH12	SP	SP
K12A SH5	K14A SH9	----	SP	SP	----	ANN SH13	K13A SH5	----
K12E SH5	K14A SH9	----	SP	SP	----	ANN SH13	K13A SH5	----

POTTER & BRUMFIELD MDR TABULATION (H13-P609)

1 2	A B C		D E F		G H J		K L M	
K1C SH7	K14C SH9	----	SP	SP	----	ANN SH12	SP	SP
K3C SH7	K14C SH9	----	SP	SP	----	ANN SH12	SP	SP
K3G SH7	K14C SH9	----	SP	SP	----	ANN SH12	SP	SP
K7C SH7	K14C SH9	----	SP	SP	----	ANN SH12	SEE SH2	----
K8C SH7	K14C SH9	----	SP	SP	SP	SP	----	ANN SH12
K10C SH7	K14C SH9	----	SP	SP	----	ANN SH12	SP	SP
K10G SH7	K14C SH9	----	----	SEE REF 7 SH 5	----	ANN SH12	SP	SP
K12C SH7	K14C SH9	----	SP	SP	----	ANN SH13	K13C SH7	----
K12G SH7	K14C SH9	----	SP	SP	----	ANN SH13	K13C SH7	----

POTTER & BRUMFIELD MDR TABULATION (H13-P611)

1 2	A B C		D E F		G H J		K L M	
K1B SH6	K14B SH9	----	SP	SP	----	ANN SH12	SP	SP
2C K3B SH6	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
2C K3F SH6	K14D SH9	----	SP	SP	----	ANN SH12	SP	SP
K7B SH6	K14B SH9	----	SP	SP	----	ANN SH12	SEE SH2	----
K8B SH6	K14B SH9	----	SP	SP	SP	SP	----	ANN SH12
K10B SH6	K14B SH9	----	SP	SP	----	ANN SH12	SP	SP
K10F SH6	K14B SH9	----	----	SEE REF 7 SH 6	----	ANN SH12	SP	SP
K12B SH6	K14B SH9	----	SP	SP	----	ANN SH13	K13B SH6	----
K12F SH6	K14B SH9	----	SP	SP	----	ANN SH13	K13B SH6	----

POTTER & BRUMFIELD MDR TABULATION (H13-P611)

1 2	A B C		D E F		G H J		K L M	
2C K3M SH 8	K14C SH 9	----	SP	SP	----	ANN SH 12	SP	SP
K3S SH 8	K14C SH 9	----	SP	SP	----	ANN SH 12	SP	SP
K72D SH.18	SEE SH 2	SP	SP	SP	SP	SP	SP	SP
K73D SH 18	SEE SH 2	----	SP	SP	SP	SP	SP	SP
K76D SH.18	SP	SP	----	K47B SH 11	SP	SP	SP	SP
K74D SH.18	SEE SH 2	----	SEE SH 2	----	SP	SP	SP	SP
K75D SH.18	SEE SH 2	----	SEE SH 2	----	SP	SP	SP	SP
K48D SH 8	----	K47B SH 11	SP	SP	SP	SP	SP	SP
K50H SH 8	----	K47A SH 11	SP	SP	SP	SP	SP	SP
K59J SH 9	K14N SH 9	----	SP	SP	SP	SP	SP	SP
K59K SH 9	K14F SH 9	----	SP	SP	SP	SP	SP	SP
K59L SH.9	K14L SH.9	----	SP	SP	SP	SP	SP	SP
K77D SH.8	K15C SH 7	----	SP	SP	SP	SP	SP	SP
K51D (SH. 8) 40	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----
K51H (SH. 8) 40	SEE SH 2A	----	SEE SH.2A	----	SEE SH 2A	----	SEE SH.2A	----
K51M (SH. 8) 40	SEE SH 2A	----	SEE SH 2A	----	SP	SP	SP	SP
K51S SH 8 40	SP	SP	SP	SP	----	SEE SH. 2A	SP	SP
2C K80 H SH. 8	K10E SH 5	----	SP	SP	SP	SP	SP	SP

POTTER & BRUMFIELD MDR TABULATION (H13-P609)

1 2	A B C		D E F		G H J		K L M	
2C K78A SH 9	SP	SP	----	C12-FOOG, F162 SH 6	SP	SP	SP	SP
K3J SH 5	K14B SH 9	----	SP	SP	----	ANN SH 12	SP	SP
K3N SH.5	K14B SH 9	----	SP	SP	----	ANN SH 12	SP	SP
K76A SH.18	SP	SP	----	K47A SH.11	SP	SP	SP	SP
K48A SH 5	----	K47A SH11	SP	SP	SP	SP	SP	SP
K50A SH 5	----	K47A SH 11	SP	SP	SP	SP	SP	SP
K59B SH 9	K14K SH 9	----	SP	SP	SP	SP	SP	SP
K59C SH 9	K14G SH 9	----	SP	SP	SP	SP	SP	SP
K59D SH 9	K14S SH 9	----	SP	SP	SP	SP	SP	SP
K71A SH 9	SP	K21A SH 11	SP	SP	SP	SP	SP	SP
K77A SH 5	K15B SH 6	----	SP	SP	SP	SP	SP	SP
K79A SH.5	SEE SH 2	----	SP	SP	SP	SP	SP	SP
K80A SH 5	K10B SH 6	----	SP	SP	SP	SP	SP	SP
K51E SH 5 40	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----
K51N SH 5 40	SEE SH 2A	----	SP	SP	SP	SP	SP	SP
2C K71E SH 9	----	K21B SH.11	SP	SP	SP	SP	SP	SP

POTTER & BRUMFIELD MDR TABULATION (H13-P611)

1 2	A B C		D E F		G H J		K L M	
2C K3K SH 6	K14A SH 9	----	SP	SP	----	ANN SH.12	SP	SP
K3P SH 6	K14A SH 9	----	SP	SP	----	ANN SH.12	SP	SP
K76B SH.18	SP	SP	----	K47A SH 11	SP	SP	SP	SP
K48B SH 6	----	K47A SH 11	SP	SP	SP	SP	SP	SP
K50F SH 6	----	K47B SH 11	SP	SP	SP	SP	SP	SP
K59E SH 9	K14J SH 9	----	SP	SP	SP	SP	SP	SP
K59G SH 9	K14R SH 9	----	SP	SP	SP	SP	SP	SP
K59M SH 9	K14H SH 9	----	SP	SP	SP	SP	SP	SP
K71B SH 9	SP	K21B SH 11	SP	SP	SP	SP	SP	SP
K77B SH 6	K15A SH 5	----	SP	SP	SP	SP	SP	SP
K78B SH 9	SP	SP	----	C12-FOOG, F162 SH 3	SP	SP	SP	SP
K79B SH 6	SEE SH 2	----	SP	SP	SP	SP	SP	SP
K51B SH 6 40	SEE SH.2A	----	SEE SH 2A	----	SEE SH 2A	----	SEE SH. 2A	----
K51K SH 6 40	SEE SH 2A	----	SP	SP	SP	SP	SP	SP
K80F SH 6	K10C SH 7	----	SP	SP	SP	SP	SP	SP
2C K71F SH 9	----	K21A SH 11	SP	SP	SP	SP	SP	SP

POTTER & BRUMFIELD MDR TABULATION (H13-P609)

1 2	A B C		D E F		G H J		K L M	
2C K3L SH 7	K14B SH 9	----	SP	SP	----	ANN SH.12	SP	SP
K3R SH 7	K14B SH 9	----	SP	SP	----	ANN SH.12	SP	SP
K72C SH 18	SEE SH 2	SP	SP	SP	SP	SP	SP	SP
K73C SH 18	SEE SH 2	----	SP	SP	SP	SP	SP	SP
K76C SH 18	SP	SP	----	K47B SH 11	SP	SP	SP	SP
K74C SH 18	SEE SH 2	----	SEE SH 2	----	SP	SP	SP	SP
K75C SH 18	SEE SH.2	----	SEE SH 2	----	SP	SP	SP	SP
K48C SH 7	----	K47B SH 11	SP	SP	SP	SP	SP	SP
K50G SH 7	----	K47B SH 11	SP	SP	SP	SP	SP	SP
K59A SH 9	K14E SH 9	----	SP	SP	SP	SP	SP	SP
K59F SH 9	K14P SH 9	----	SP	SP	SP	SP	SP	SP
K59H SH 9	K14M SH 9	----	SP	SP	SP	SP	SP	SP
K77C SH 7	K15D SH 8	----	SP	SP	SP	SP	SP	SP
K51G SH 7 40	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----
K80G SH. 7	K10D SH 8	----	SP	SP	SP	SP	SP	SP
2C K51C SH 7 40	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----	SEE SH 2A	----

K51A D1 RPS: T1 M1 | T2 M2 | T3 M3 | T4 M4
K51J D1 RPS: T1 M1
K51B D1 RPS: A B | D E | G H | K L
K51K D1 RPS: A B
K51E D2 RRS: A B | D E | G H | K L
K51N D2 RPS: A B — NOT USED
K51F D2 RPS: T1 M1 | T2 M2 | T3 M3 | T4 M4
K51P D2 RPS: T1 M1 — NOT USED
K51G D2 RPS: A B | D E | G H | K L
K51C D1 RPS: A B | D E | G H | K L

TO OFF NORMAL STATUS DISPLAY, REF 18 40

K51D (SPARE A B) D1 RPS | (SPARE D E) D1 RPS | (SPARE G H) D1 RPS | (SPARE K L) D1 RPS | (SPARE A B) D2 ESF | (SPARE D E) D2 ESF
K51H (SPARE A B) D2 RPS | (SPARE D E) D2 RPS | (SPARE G H) D2 RPS | (SPARE K L) D2 RPS
K51S H J N — HWC — SHUTDOWN SIGNAL, REF. 21

TO OFF NORMAL STATUS DISPLAY, REF. 18 40

NOTE: THIS DRAWING WAS CREATED USING COMPUTER AIDED DRAFTING DO NOT MAKE MANUAL CHANGES

NMP2-VENDOR DWG DATA STAMP
SUPERSEDES SWEC FILE NO. 0007.225-001-037J
EQUIP. ID. SEE CDS LISTING
MFG. NAME G.E. | MFG. CODE G08000
SPEC. and/or P.O. P800A | SWEC JOB NO. 12177
DRAWING NO. 807E166TY, SH. 2A

Constellation Energy Group
Nine Mile Point Nuclear Station
UNIT 2
ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

REV	DATE	DESCRIPTION OF REVISION	PREP	CHKD	ENGR			ELEC	MECH	STR
4	SEE FCMS	INCORP. ECN# ECP-11-000506-CN-036 0007225001037-02A-03.00	SEE FCMS	SEE FCMS	SEE FCMS			N/A	N/A	N/A

INDEX	SCALE	DRAWING NO	SHEET
N/A	NONE	0007.225-001-037	N/A

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING
APPLIED PRACTICES | SURFACES | TOLERANCES ON MACHINED DIMENSIONS: FRACTIONS | DECIMALS | ANGLES

GENERAL ELECTRIC

TITLE: ELEMENTARY DIAGRAM — REACTOR PROTECTION SYSTEM
807E166TY
CONT. ON SH. 10 | SH. No. 9
FIRST MADE FOR: NMP- 2

REACTOR SCRAM TRIP LOGIC A1

H13-P609 — 120VAC (SEE FIG.2 SH.4)

F14A (REF.) | F70A (REF.) | F71A (REF.) | F14E (REF.) | F27A (REF.)

- TURBINE STOP VALVE CLOSURE — K10A, K10E — T1 M1 K9A — TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
- TURBINE CONTROL VALVE FAST CLOSURE — K8A
- CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL — K1A, K55A — T1 M1 K18 — CRD CRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
- MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS — K11A — W116 (-7 -G, -6 -F) — J411 — 217 — KK32, KK31 — K3E, K3A, K3R, K3K — H13-P611B
- DRYWELL HIGH PRESSURE — K4A
- REACTOR VESSEL HIGH PRESSURE — K5A
- REACTOR VESSEL LOW WATER LEVEL — K6A
- REACTOR SCRAM TRIP LOGIC A1 BYPASS — S10A — K7A
- NEUTRON MONITORING SYSTEM TRIPS — K12A, K12E
- MANUAL SCRAM TEST OR NON-COINCIDENT TRIP — K15A — T1 M1 K19A — 3T 3L K14A

K14A, K14E, K14J, K14N — 2T 2L K14A, 3T 3L K14E, 3T 3L K14J, 3T 3L K14N

K59D, K59C, K59B — 1 2 K14A R14A — K71A, K71E — K70A, K78A — R14E, K14E — K59A

F14A (REF.) — 7 8 K14A — 7 8 S10A — W DS1A — A DS24A — 1 2

H13-P609A — F14J (REF) NN-72 W13B -1 -3 217 A C KK-26 KK-27 — F14N (REF) NN-74 KK-28 KK-29 H13-P611B — B D 217 W13B -2 -4 NN-73 NN-75 — K14J K14N — H13-P609A

120VAC (SEE FIG.22 SH.4) — D1 RPS-B GREEN/WHITE 36A

K14A 3 4 — V 05-01C — SEE REF. 20 — D1 RPS

REACTOR SCRAM TRIP LOGIC A1 — D1 RPS

REACTOR SCRAM TRIP LOGIC A2

H13-P609 — 120VAC (SEE FIG.4 SH.4)

F14C (REF.) | F70C (REF.) | F14G (REF.) | F27C (REF.)

- TURBINE STOP VALVE CLOSURE — K10C, K10G — T1 M1 K9C — TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
- TURBINE CONTROL VALVE FAST CLOSURE — K8C
- CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL — K1C, K55C — T1 M1 K18C — CRD CRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
- MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS — K11C — W118 (G -5, K -8) — J112 — 219 — FF-12, FF-15 — K3C, K3G, K3M, K3S — H13-P611C
- DRYWELL HIGH PRESSURE — K4C
- REACTOR VESSEL HIGH PRESSURE — K5C
- REACTOR VESSEL LOW WATER LEVEL — K6C
- REACTOR SCRAM TRIP LOGIC A2 BYPASS — S10C — K7C
- NEUTRON MONITORING SYSTEM TRIPS — K12C, K12G
- MANUAL SCRAM TEST OR NON-COINCIDENT TRIP — K15C — T1 M1 K19C — 3T 3L K14C

K14R, K14L — 9 10 10 9 K14G, 9 10 10 9 K14C

K59A, K59F, K59H — 1 2 K14C R14C — K70C — K14G R14G — K59C

F14C (REF.) — 7 8 K14C — 7 8 S10C — W DS1C — A DS24C — 1 2

H13-P609D — F14L (REF.) DD-1 W140 -1 -3 219 A C FF-8 FF-9 — F14R (REF.) DD-3 FF-10 FF-11 — B D 219 W140 -2 -4 DD-2 DD-4 — K14L K14R — H13-P609D

120 VAC (SEE FIG.24 SH.4) — D3 B-RPS ORANGE/WH 36C

K14C 3 4 — V 07-01C — SEE REF.20 — D3 RPS

REACTOR SCRAM TRIP LOGIC A2 — D3 RPS

REACTOR SCRAM TRIP LOGIC B1

H13-P611 — 120VAC (SEE FIG.3 SH.4)

F14B (REF.) | F70B (REF.) | F71B (REF.) | F14F (REF.) | F27B (REF.)

- TURBINE STOP VALVE CLOSURE — K10B, K10F — T1 M1 K9B — TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
- TURBINE CONTROL VALVE FAST CLOSURE — K8B
- CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL — K1B, K55B — T1 M1 K18B — CRD CRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
- MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS — K11B — W117 (-7 -G, -6 -F) — J404 — 218 — KK34, KK33 — K3R, K3J, K3L, K3N — H13-P609B
- DRYWELL HIGH PRESSURE — K4B
- REACTOR VESSEL HIGH PRESSURE — K5B
- REACTOR VESSEL LOW WATER LEVEL — K6B
- REACTOR SCRAM TRIP LOGIC B1 BYPASS — S10B — K7B
- NEUTRON MONITORING SYSTEM TRIPS — K12B, K12F
- MANUAL SCRAM TEST OR NON-COINCIDENT TRIP — K15B — T1 M1 K19B — 3T 3L K14B

K14B, K14F, K14K, K14P — 9 10 10 9 K14B K14F, 9 10 10 9 K14K K14P — 2T 2L K14B, 3T 3L K14F, 3T 3L K14K, 3T 3L K14P

K59E, K59G, K59M — 1 2 K14B R14B — K71B, K71F — K70B, K78B — R14F, K14F — K59K

F14B (REF.) — 7 8 K14B — 7 8 S10B — W DS1B — A DS24B — 1 2

H13-P611A — F14K (REF.) NN-72 W139 -1 -3 218 A C KK-28 KK-29 — F14P (REF.) NN-74 KK-30 KK-31 H13-PG09B — B D 218 W139 -2 -4 NN-73 NN-75 — K14K K14P — H13-P611A

120 VAC (SEE FIG. 23 SH. 4) — D2 RPS A YELLOW/WHITE 36B

K14B 3 4 — V 06-01C — SEE REF. 20 — D2 RPS

REACTOR SCRAM TRIP LOGIC B1 — D2 RPS

REACTOR SCRAM TRIP LOGIC B2

H13-P611 — 120VAC (SEE FIG.8 SH.4)

F14D (REF.) | F70D (REF.) | F14H (REF.) | F27D (REF.)

- TURBINE STOP VALVE CLOSURE — K10D, K10H — T1 M1 K9D — TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
- TURBINE CONTROL VALVE FAST CLOSURE — K8D
- CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL — K1D, K55D — T1 M1 K18D — CRD CRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
- MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS — K11D — K3D, K3H, K3B, K3F
- DRYWELL HIGH PRESSURE — K4D
- REACTOR VESSEL HIGH PRESSURE — K5D
- REACTOR VESSEL LOW WATER LEVEL — K6D
- REACTOR SCRAM TRIP LOGIC B2 BYPASS — S10D — K7D
- NEUTRON MONITORING SYSTEM TRIPS — K12D, K12H
- MANUAL SCRAM TEST OR NON-COINCIDENT TRIP — K15D — T1 M1 K19D — 3T 3L K14D

K14S, K14M, K14H, K14D — 9 10 10 9 K14H K14D, 9 10 10 9 K14M K14S

K59J, K59K, K59L — 1 2 K14D R14D — K70D — K14H R14H — K59M

F14D (REF.) — 7 8 K14D — 7 8 S10D — W DS1D — A DS24D — 1 2

H13-P611D — F14M (REF) A D J104 W119 -1 -3 220 A C FF-8 FF-9 — F14S (REF) FF-10 FF-11 — B D 220 W119 -2 -4 C F — K14M K14S — H13-P611D

120VAC (SEE FIG 25 SH4) — H13-P609C — D4 A RPS BLUE/WHITE 36D

K14D 3 4 — V 08-01C — SEE REF 20 — D4 RPS

REACTOR SCRAM TRIP LOGIC B2 — D4 RPS

NOTE: THIS DRAWING WAS CREATED USING COMPUTER AIDED DRAFTING DO NOT MAKE MANUAL CHANGES

SUPERSEDES SWEC FILE NO. 0007.225-001-011H
EQUIP. ID. SEE COS LISTING
MFG. NAME G.E. | MFG. CODE G08000
SPEC. and/or P.O. P800A | SWEC JOB NO. 12177
MFG's/VENDOR DWG DATA STAMP

DRAWING NO. 807E166TY, SH. 9

Constellation Energy Group
Nine Mile Point Nuclear Station
UNIT 2

ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

INDEX	SCALE	DRAWING NO	SHEET
N/A	NONE	0007.225-001-011	N/A

REV	DATE	DESCRIPTION OF REVISION	PREP	CHKD	ENGR	ELEC	MECH	STR
8	SEE FCMS	INCORP. ECN ECP-11-000426-CN-002 00072250010l1-009-07.00(0000)	SEE FCMS	SEE FCMS	SEE FCMS			

..\0007225001011.dgn 4/12/2012 8:49:19 AM

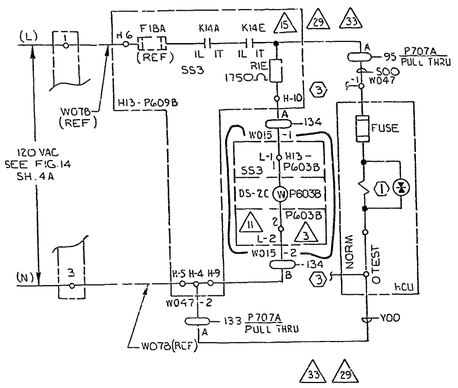
UNLESS OTHERWISE SPECIFIED THE FOLLOWING APPLY	SURFACES	FRACTIONS	DECIMALS	ANGLES
	125	±	±	±

TITLE ELEMENTARY DIAG. CONT. ON SH. 11 SH. NO. 10

REACTOR PROT. PILOT SCR. SOL. SYS.

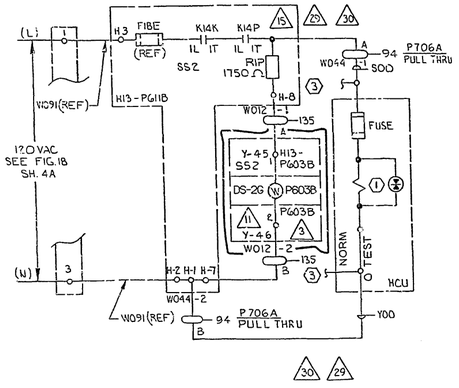
FIRST MADE FOR NMP-2

GENERAL ELECTRIC



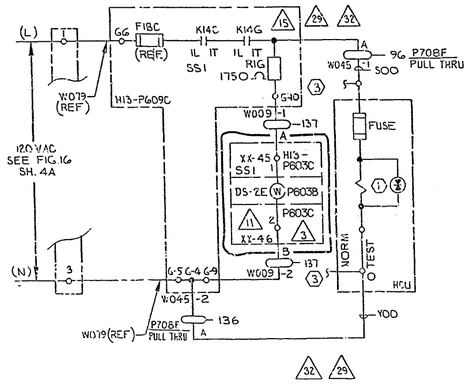
GROUP 1

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSA02



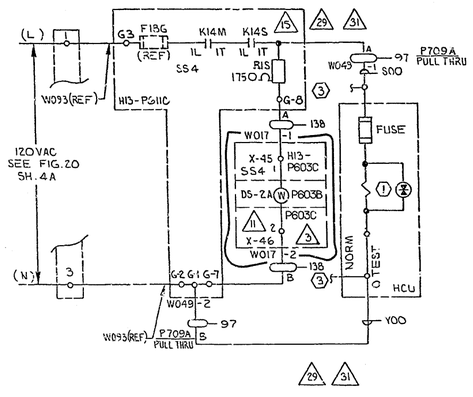
GROUP 2

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSB02



GROUP 3

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSC02



GROUP 4

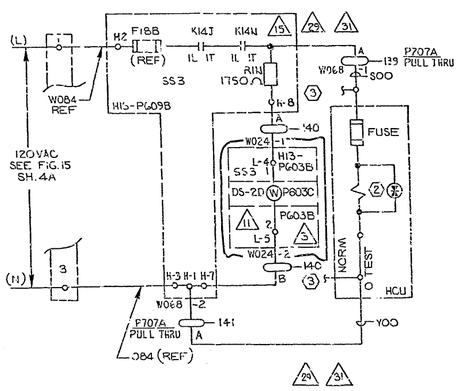
PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSD02

(1) PILOT SOLENOID C12-D001-FXXX (XX-XX) FOR "A" SCRAM VALVE

(2) PILOT SOLENOID C12-D001-FXXX (XX-XX) FOR "B" SCRAM VALVE

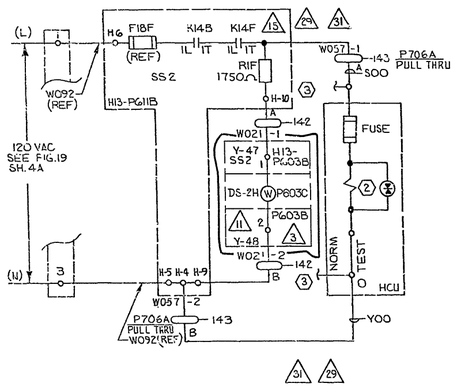
(3) TO OTHER PILOT SCRAM SOLENOID IN GROUP SEE TABLE 1 SH. 1

(4) SEE TABLE 1 AND FIG 9 SH. 1 FOR SOLENOIDS IN THE GROUP.



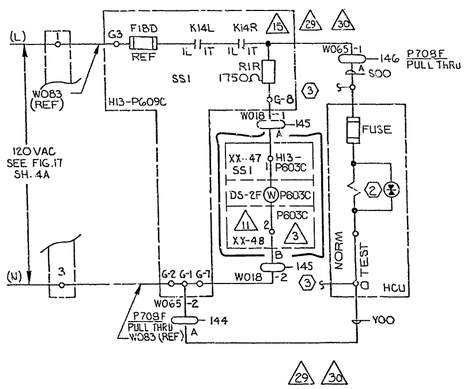
GROUP 1

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSA03



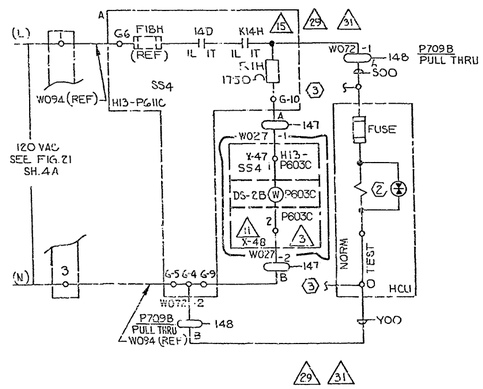
GROUP 2

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSB03



GROUP 3

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSC03



GROUP 4

PILOT SCRAM VALVE SOLENOID
TYPICAL OF (4) SOLENOIDS
CKT 2RPSD03

NOTE:
SEE SHEET 4A FOR CABLE ROUTING.

SUPERSEDES SWEC FILE NO.	0007.225-001-012L
EQUIP. ID.	SEE COS LISTING
MFG. NAME	G. E.
MFG. CODE	G08000
SPEC. and/or P.O.	P800A
SWEC JOB NO.	12177

NMP2-VENDOR DWG DATA STAMP

DRAWING NO. 807E166TY, SH. 10
CADD DESIGN FILE NO. 0007222001012.DGN

LINE CODES: 41,42,51,52,94-97,134,135,137 138,140-143,145-148,173-176

NOTE: THIS DRAWING WAS CREATED USING CADD. DO NOT MAKE MANUAL CHANGES

NUCLEAR SAFETY RELATED

ORIGINAL

NIAGARA MOHAWK	NINE MILE POINT NUCLEAR STATION - UNIT 2 SCRIBA, N.Y.
	ELEMENTARY DIAGRAM
	REACTOR PROTECTION SYSTEM
	2CEC*PNL603
SCALE NONE	DRAWING NO 0007.225-001-012

NO.	DATE	BY	DESCRIPTION	CK	APP
1	12-15-11	SM	ORIGINAL ISSUE- REVISE PER CONFIGURATION CHG NOS 261959 (CORRECTED LAMP LOCATION)	DM	

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING
APPLIED PRACTICES | SURFACES √ | TOLERANCE ON INCHED DIMENSIONS: FRACTIONS ± | DECIMALS ± | ANGLES ±
FIRST MADE FOR

TITLE ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

GENERAL ELECTRIC

CONTACT CLOSED WHEN VALVE IS NOT FULLY CLOSED
CONTACT CLOSED WHEN VALVE IS NOT FULLY OPEN
CONTACT CLOSED WHEN VALVE IS FULLY CLOSED
CONTACT CLOSED WHEN VALVE IS NOT FULLY OPEN
CONTACT CLOSED WHEN VALVE IS NOT FULLY CLOSED

VALVE C12-F180 POS IC P003 10 P003 10 P002 VALVE C12-F180
VALVE C12 F010 POS IC
VALVE C12 F180 POS IG
F010
F180
F011
F181
VALVE C12 F011 POS IC
VALVE C12 F181 POS IC
VALVE C12-F181 POS IO
P003 10 P003 10 R03 VALVE C12 F011 POS IO
LOCAL
LOCAL

P707E TC309J1 A B C D E F G
W111 -1 -2 -3 -4 -25 -5 -6 -7 -201
P707E PULL THRU A B C D E F G
-1 -2 -K04 -K05 W141 -3 -4 -K02 -K03 -201
-K00 -K01

JJ49 JJ50 JJ51 JJ52 JJ60 JJ61 JJ62 JJ63 JJ53 JJ54 JJ55
BI K58 B4 BI K52 B4 I 2 K81 I 2 K82 I 2 K83 I 2 K84 BI K53 B4 BI K54 B4
T1 K58 M1 T1 K52 M1 T1 K53 M1 T1 K54 M1
D1 RPS

JJ64 P308 JJ65 JJ66 JJ67 P308 JJ68 P308 JJ69 H13-P609B
A B C D E F
W110 -1 -2 -3 -4 -5 -6 -200

H13-P603B (L) F20 (REF) YY1 YY2 YY3 TB YY4 TB YY5 YY6
XD0 X01 G00 R00 Y00 R01 G01
18 9
D1 RPS 120 VAC DIV I UPS BUS "A" (SEE FIG 6, SH 4A)
G DS 3 R DS 4 R DS 6 G DS 5
11
Y00
VENT VALVE CLOSED
VENT VALVE OPEN
DRAIN VALVE OPEN
DRAIN VALVE CLOSED
(N) H13-P603B
H13-P603B

SCRAM DISCHARGE VOLUME ISOLATION VALVE POSITION INDICATOR LIGHT (REF 1)
CKT. NO. 2RPS-W01

H13-P609 (+) P00 FZ1A (REF) 18
B C K71A J4130F W036-6 107F J103 0F B C K71F J103 0G H13-P611 P-11
T1 K21A M1 T2 K21A M2 500
P609A P414 A W062 1 102A
P707A TC101J1 A
B1 K21A B4
P707A TC101J1 B 102B
W062 2
P414 P609A B 501
VALVE C12-F110A
D1 ESF 125VDC (SEE FIG 5, SH 4A)
H13-P609 (-) N00 FZ2A (REF)

BACKUP SCRAM VALVE "A" (REF 1)
CKT. NO. 2RPSA04

H13-P611 (+) P00 FZ1B (REF) 18
B C K71B J4110F W038-6 108F J109 0F B C K71E J109 0G H13-FC39
T1 K21B M1 T2 K21B M2 500
P611A P419 A W058 1 104A
P706A TC106J1 A
B1 K21B B4
P706A TC106J1 B 104B
W058 2
P419 P611A B 501
VALVE C12-F110B
D2 ESF 125 VDC (SEE FIG 7, SH 4A)
H13-P611 (-) N00 FZ2B (REF)

BACKUP SCRAM VALVE "B"
CKT. NO. 2RPSB04

K81 K82 K83 K84 A B N
H13-P609C
B A C E D F H G I K J L -255
W142 -2 -1 -3 -5 -4 -6 -8 -7 -9 -11 -10 -12
403-014 403-015 C1971 GND
403-016 403-015 C1972 GND
412-016 412-015 C1973 GND
413-016 413-015 C1974 GND
C91-P620

CLOSES ON CONTROL VALVE FAST CLOSURE
CONTACTS CLOSED WHEN 1ST STAGE TURBINE PRESSURE > SET POINT
CLOSES ON TURBINE STOP VALVE CLOSURE

P00 H13-P609A (+) F53A (REF) P01
DS 7 K48A 26 P609A-P413 B 107B
DS 8 K48B 26 18
DS 9 K50A 26 A 107A
DS 10 K50H 26 15 B C J1030C H13-P611D
W036-3 107C P413 C
P413 D P609A W036-4 107D
J1030D E K76B 18 F J1030E 26 H13-P611D
W036-5 107E J413 E
K76A P609A E F
K47A 2 8
DS 11 DS 12 CR3A CR3B
P609A J419 C W063-3 123A P707A TC101J2 C
52 52 TC1 C85A SWGR (REF 14)
T00 T01
P707A TC101J2 D W063-4 123B
123C 123D
F54A J419 D P609A (-) H13-P609A (REF) N01 N00
RECIRC PUMP TRIP SYSTEM "A"
CKT. NO. 2RPSA04

P415 P609A A 179A
J324 3C 3NC S9A BYPASS 5A H13-P602A
W101-2 179B
P415 P609A B 5 K47A 6
P415 C P609A W101-3 179C
J324 C IC S9A INO A CD D H13-P602A
W101-4 179D
P415 P609A D

P00 H13-P611A (+) F53B (REF) P01
J411A W038-10 108A J1090A
DS 13 K48C 26 18 J109-B0 H13-P609D W038-2 J411 B 108B
DS 15 K48D 26
DS 14 K50F 26 J411 C W038-3 108C J109 C
DS 16 K50G 26 18
J411 D W038-4 108D J109-D
K76C 18 J1090E 26 H13-P609D W038-5 108E J411 E
K76D E F
J418 P611A A 26 W102-1 180A J271 3C 3NC S9B BYPASS H13-P602B
W102-2 180B J418 P611A B 5 K47B 6
J271 C IC S9B INO A D H13-P602B
W102-4 180D J418 P611A D
K47B 2 8
DS 17 DS 18 CR4A CR4B
P611A P420 A W059-1 124A P706A TC106J2 A
52 52 TC1 C84A SWGR (REF 14)
T00 T01
P706A TC106J2 B W059-2 124B
124C 124D
F54B P420 B P611A (-) H13-P611A (REF) N01 N00
RECIRC PUMP TRIP SYSTEM "B"
CKT. NO. 2RPSB04

K47A A1 ESF PP30 PP31 H13-P609A 223 W122
J260 A B E51A-AT8 R 14 15 21Y H13-P629A J151 C B 26 W071 2THAN05K22
2THAN05NO7 H13-P707B 1H202J202 C B K150A B35-P001A (REF. 14, SH. 20)
K47B A2 ESF PP30 PP31 H13-P611A 224 W123
J260 A B B22C-AT5 V 21Y H13-P628A J320 C B 99 W067 2THAN05K42
2THAN05NO8 H13-P706B 1H202J202 C B K150B B35-P001B (REF. 14, SH. 21)

K81 K82 K83 K84 D E N
H13-P609C
B A C E D F H G I K J L -256
W143 -2 -1 -3 -5 -4 -6 -8 -7 -9 -11 -10 -12
403-014 403-013 C1971 GND
403-016 403-015 C1972 GND
412-016 412-015 C1973 GND
413-016 413-015 C1974 GND
C91-P623

LINE CODES: 98-108, 123, 124, 179
180, 200, 201, 223, 224

NOTE: THIS DRAWING WAS CREATED USING COMPUTER AIDED DRAFTING DO NOT MAKE MANUAL CHANGES

ORIGINAL

SUPERSEDES SWEC FILE NO.	0007.225-001-013T
EQUIP. ID.	SEE CDS LISTING
MFG. NAME G.E.	MFG. CODE 08000
SPEC. and/or P.O. P800A	SWEC JOB NO. 12177

MICRO-FICHE/MICRO DATA ITEMS
DRAWING NO. 807E166TY, SH. 11

Constellation Energy Group
Nine Mile Point Nuclear Station
UNIT 2
ELEMENTARY DIAGRAM
REACTOR PROTECTION SYSTEM

REV	DATE	DESCRIPTION OF REVISION	PREP	CHKD	ENGR		ELEC	MECH	STR
3	6/9/10	INCORPORATED CN-004279	LWF	GCB	TOL 6/9/10		N/A	N/A	N/A

INDEX	SCALE	DRAWING NO	SHEET
N/A	NONE	0007.225-001-013	N/A

GENERAL ELECTRIC 807E166TY SH 18

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:

ACTIVITY	SURFACE	TOLERANCES ON UNLISTED DECIMALS: MACHINED	BEND	ANGLES
ACTIVITY PRACTICES	✓	±	±	±

TITLE: ELEM DIAG REACTOR PROTECTION SYSTEM

FIRST MADE FOR

0007225-001-020

P707A TM109J103 W064

PRESSURE TRANSMITTER B22-N078A (+) I01 (-) I00 — LEVEL TRANSMITTER B22-N080A (+) I03 (-) I02 — PRESSURE TRANSMITTER C72-N050A (+) I05 (-) I04 — HZ2-P055 — I06 — I07 — LOCAL

P707A TM109J109 W060 — P609A-J4076J — P609A-J4070V — P609D-J1086V

TRIP UNIT (B22-N678A) — TRIP UNIT (B22-N680A) — TRIP UNIT (C72-N650A) — TRIP UNIT (C72-N651) — TRIP UNIT (C72-N653) — TRIP UNIT (C72-N652A)

Z1A

B22-N679A (REF 7, SH 14)

K5A — K6A — K46A — K4A — K45A — K36 — K37 — K9A — K76A

DE-ENERGIZES WHEN VESSEL PRESSURE ABOVE SET POINT | DE-ENERGIZED WHEN VESSEL LEVEL IS BELOW LEVEL 3 TRIP | DE-ENERGIZE WHEN DRYWELL PRESSURE ABOVE SET POINT | BELOW SET POINT | ABOVE SET POINT | DE-ENERGIZE WHEN 1ST STAGE TURBINE PRESSURE ABOVE SET POINT

CKT. NO. 2RPSA01

Z1A: (B22-N678A) (B22-N680A) (C72-N650A) (C72-N652A) (C72-N651) (C72-N653)

CR1A — B22H-K147A (REF 7, SH 14) — CR2A — K39A — K38 — K40A

TRIP UNIT IN CAL OR GROSS FAILURE (SAFETY) — TRIP UNIT IN CAL OR GROSS FAILURE (NON-SAFETY) — TRIP UNIT OUT OF FILE OR POWER FAILURE

P708F TM420J420 W069

PRESSURE TRANSMITTER B22-N078C (+) I01 (-) I00 — LEVEL TRANSMITTER B22-N080C (+) I03 (-) I02 — PRESSURE TRANSMITTER C72-N050C (+) I05 (-) I04 — HZ2-P054 — I06 — I07 — LOCAL

TRIP UNIT (B22-N678C) — TRIP UNIT (B22-N680C) — TRIP UNIT C72-N650C — TRIP UNIT C72-N652C

Z1C

B22-N679C (REF 7, SH 14)

K5C — K75C — K73C — K72C — K4C — K74C — K9C — K76C

DE-ENERGIZE WHEN VESSEL PRESSURE ABOVE SET POINT | DE-ENERGIZE WHEN VESSEL BELOW LEVEL 3 TRIP | DE-ENERGIZE WHEN DRYWELL PRESSURE IS ABOVE SET POINT | DE-ENERGIZE WHEN 1ST STAGE TURBINE PRESSURE ABOVE SET POINT

CKT. NO. 2RPSC01

Z1C: (B22-N678C) (B22-N680C) (C72-N650C) (C72-N652C)

CR1C — B22H-K147C (REF 7, SH 14) — CR2C — K39C — K40C

TRIP UNIT IN CAL OR GROSS FAILURE — TRIP UNIT OUT OF FILE OR POWER FAILURE

H13-P609

P706A TM109J109 W060

PRESSURE TRANSMITTER B22-N078B (+) I01 (-) I00 — LEVEL TRANSMITTER B22-N080B (+) I03 (-) I02 — PRESSURE TRANSMITTER C72-N050B (+) I05 (-) I04 — PRESSURE TRANSMITTER C72-N052B (+) I07 (-) I06 — HZ2-P027 — LOCAL

P811A-J4150J — P811A-J4140V

TRIP UNIT (B22-N678B) — TRIP UNIT (B22-N680B) — TRIP UNIT (C72-N650B) — TRIP UNIT (C72-N652B)

Z1B

B22-N679B (REF 7, SH 15)

K5B — K6B — K46B — K4B — K45B — K38 — K9B — K76B

DE-ENERGIZES WHEN VESSEL PRESSURE ABOVE SET POINT | DE-ENERGIZED WHEN VESSEL LEVEL IS BELOW LEVEL 3 TRIP | DE-ENERGIZE WHEN DRYWELL PRESSURE IS ABOVE SET POINT | DE-ENERGIZE WHEN 1ST STAGE TURBINE PRESSURE ABOVE SET POINT

CKT. NO 2RPSB01

Z1B: (B22-N678B) (B22-N680B) (C72-N650B) (C72-N652B)

CR1B — B22H-K147B (REF 7, SH 15) — CR2B — K39B — K40B

TRIP UNIT IN CAL OR CROSS FAILURE — TRIP UNIT OUT OF FILE OR POWER FAILURE

P709B TM208J208 W073

PRESSURE TRANSMITTER B22-N078D (+) I01 (-) I00 — LEVEL TRANSMITTER B22-N080D (+) I03 (-) I02 — PRESSURE TRANSMITTER C72-N050D (+) I05 (-) I04 — HZ2-P004 — I06 — I07 — LOCAL

P811D-J1140J — P811D-J1140V

TRIP UNIT B22-N678D — TRIP UNIT B22-N680D — TRIP UNIT C72-N650D — TRIP UNIT C72-N652D

Z1D

B22-N679D (REF 7, SH 15)

K5D — K75D — K73D — K72D — K4D — K74D — K9D — K76D

DE-ENERGIZE WHEN VESSEL PRESSURE ABOVE SET POINT | DE-ENERGIZE WHEN VESSEL BELOW LEVEL 3 TRIP | DE-ENERGIZE WHEN DRYWELL PRESSURE IS ABOVE SET POINT | DE-ENERGIZE WHEN 1ST STAGE TURBINE PRESSURE ABOVE SET POINT

CKT. NO. 2RPSD01

Z1D: (B22-N678D) (B22-N680D) (C72-N650D) (C72-N652D)

CR1D — B22H-K147D (REF 7, SH 15) — CR2D — K39D — K40D

TRIP UNIT IN CAL OR GROSS FAILURE — TRIP UNIT OUT OF FILE OR POWER FAILURE

H13-P611

SUPERSEDES SWEC FILE NO.	0007.225-001-020K
EQUIP. ID.	SEE DCIS LISTING
MFG. NAME G.E.	MFG. CODE G08000
SPEC. and/or P.O. P800A	SWEC JOB NO. 12177

G.E. DWG. NO.(807E166TY, SH.18)

LINE CODES: 125-132

NMP2 807E166TY SH:018 REV:02
NMPC/SW:0007225001020

NIAGARA MOHAWK	STA NINE MILE POINT NUCLEAR STA-UNIT 2
APP	ELEMENTARY DIAGRAM
APP	REACTOR PROTECTION SYSTEM
DES — DR — CK —	DATE 9/1/89
MF	SCALE NONE — DWG NO 0007.225-001-020

MK	DATE	BY	MF	DESCRIPTION OF REVISION AND M.O.	CK	APP
2	9/1/89	GK		REISSUED PER CAR 90.3035		
1	4/19/89	KD		ORIGINAL ISSUE — INCORP FDDR K01-6166, REV. 2 AND FDDR K01-6186, REV. 0		

Constellation

Training ID: **Nov 2023 NRC JPM SRO RC** Revision: **0.0**

Title: **Radiological and Heat Stress Requirements Related to Operator Work In High Radiation Areas - Valve Leak in RWCU Pump Room**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mecki Ghabarou	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 20 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. RP-AA-10, Radiation Protection Process Description
2. RP-AA-11, External Dose Control Program Description
3. RP-AA-12, Internal Dose Control Program Description
4. RP-AA-203, Exposure Control and Authorization
5. RP-AA-403, Administration of the Radiation Work Permit Program
6. RP-AA-460, Controls for High and Locked High Radiation Areas
7. RP-AA-460-001, Controls for Very High Radiation Areas
8. SA-AA-111, Heat Stress Control
9. NISP-RP-013, Radiation Protection Standard Glossary of Terms
10. NUREG 1123 K/A 2.3.12 (3.2/3.7)

Instructor Information

A. JPM Information

1. Description

   a. This JPM is used to test knowledge in calculation of overall dose and control mechanisms to allow the operators to continue or perform work in high dose areas. This JPM tests basic mathematics and understanding of heat stress stay times and remaining dose limitations.

   b. Critical steps are annotated in the Evaluator standard column with a bolded ***Pass/Fail**.

2. Task Information:

   a. GAP-RPP07-00002, Comply with administrative exposure limits

   b. K/A 2.3.12 (3.2/3.7), Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☐ Simulator ☒ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)	3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	2.3.12 - Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters			
K/A Importance Rating:	RO	3.2	SRO	3.7

4. Recommended Start Location

   a. Training Classroom

5. JPM Setup (if required)

a. Ensure sufficient copies of the following procedures are available in the exam area:

- RP-AA-10
- RP-AA-11
- RP-AA-12
- RP-AA-203
- RP-AA-403
- RP-AA-460
- RP-AA-460-001
- SA-AA-111
- NISP-RP-013

b. Ensure calculators are available.

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given:
	• The plant is at 100% power.
	• A valve leak has developed in the WCS-P1B room.
	• Entry into the room is required to assist Maintenance with repairing valve WCS-V29B.
	• An updated survey map is provided.
	• Your current year-to-date exposure is 1800 mRem TEDE.
	• You have not received any dose extension this year.
	• Job conditions are as follows:
	  ○ You will be performing Moderate Work for a total of 45 minutes at valve WCS-V29B.
	  ○ You will be wearing work clothes + plastics.
	  ○ Radiation Protection has determined that a respirator is not required.
	  ○ The wet bulb temperature in the room is 93°F.
	
	Evaluator: *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *address the radiological and heat stress aspects of performing this work, and record your findings on the provided scorecard.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	Obtain a copy of the reference procedures and review / utilize the correct section of the procedures.	P	SAT / UNSAT ***STD:*** *Associated procedures obtained.*
3.	Evaluates Scorecard questions		
3a.	Evaluates Scorecard Question #1: *Classify the room based on radiation level.*	P	***PASS / FAIL** ***STD:*** *Determines area is a Locked High Radiation Area.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3b.	Evaluates Scorecard Question #2: *Determine the highest dose rate in the room and the location.*	P	***PASS / FAIL** ***STD:*** *2,000 mrem/hr by the pump.*
3c.	Evaluates Scorecard Question #3: *Determine the dose rate at the work location.*	P	***PASS / FAIL** ***STD:*** *500 mrem/hr.*
3d.	Evaluates Scorecard Question #4: *Determine the highest contamination level in the area and the location.*	P	***PASS / FAIL** ***STD:*** *20,000 dpm/100cm^2 at location circle 5.*
3e.	Evaluates Scorecard Question #5: *Determine the contamination level at the work location.*	P	***PASS / FAIL** ***STD:*** *6,000 dpm/100cm^2.*
3f.	Evaluates Scorecard Question #6: *Determine the expected dose for the duration of the job.*	P	***PASS / FAIL** ***STD:*** *375 mrem (500 mrem/hr x 0.75 hr).*
3g.	Evaluates Scorecard Question #7: *Determine the heat stress action time limit.*	P	***PASS / FAIL** ***STD:*** *20 minutes per SA-AA-111, Attachment 3.*
Evaluator Note:	• The following JPM step is for SROs only. ROs will complete the JPM through step 3 and SROs will complete all steps of the JPM. • Provide SRO Operators with the SRO Only cue sheet.		
4.	Evaluates SRO Only Scorecard questions		
4a.	Evaluates SRO Only Scorecard Question #1: *Is a dose extension required?*	P	***PASS / FAIL** ***STD:*** *Determines that the answer is "YES" because annual exposure will exceed 2000 mRem dose control level. (1800 mRem + 375 mRem = 2175 mRem).*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
4b.	Evaluates SRO Only Scorecard Question #3: *Identify the form required to be completed for the dose extension.*	P	***PASS / FAIL** ***STD:*** *Determines that RP-AA-203, Attachment 1 is the required form for dose extensions.*
4c.	Evaluates SRO Only Scorecard Question #4: *Check all appropriate boxes below for the approvals required for this dose extension.*	P	***PASS / FAIL** ***STD:*** *Determines the following as the required approvals:* • *Work Group Supervisor* • *Radiation Protection Manager*

TASK STANDARD	Radiological and heat stress requirements related to work in the WCS-P1B room addressed.

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given:
	• The plant is at 100% power.
	• A valve leak has developed in the WCS-P1B room.
	• Entry into the room is required to assist Maintenance with repairing valve WCS-V29B.
	• An updated survey map is provided.
	• Your current year-to-date exposure is 1800 mRem TEDE.
	• You have not received any dose extension this year.
	• Job conditions are as follows:
	  ○ You will be performing Moderate Work for a total of 45 minutes at valve WCS-V29B.
	  ○ You will be wearing work clothes + plastics.
	  ○ Radiation Protection has determined that a respirator is not required.
	  ○ The wet bulb temperature in the room is 93°F.

INITIATING CUE	**(Operator Name)**, *address the radiological and heat stress aspects of performing this work, and record your findings on the provided scorecard.*

Scorecard

Answer the following when performing this task:
1.
Classify the room based on radiation level (check one): ☐ Radiation Area ☐ High Radiation Area ☐ Locked High Radiation Area ☐ Very High Radiation Area
2.
Determine the highest dose rate in the room and the location:
3.
Determine the dose rate at the work location:
4.
Determine the highest contamination level in the area and the location:
5.
Determine the contamination level at the work location:
6.
Determine the expected dose for the duration of the job:
7.
Determine the heat stress action time limit:

Examiner Scorecard. Do Not Provide to Operator.

Answer the following when performing this task:
1.
Classify the room based on radiation level (check one): ☐ Radiation Area ☐ High Radiation Area ☒ **Locked High Radiation Area** ☐ Very High Radiation Area
2.
Determine the highest dose rate in the room and the location: **2000 mrem/hr near the pump**
3.
Determine the dose rate at the work location: **500 mRem/hr**
4.
Determine the highest contamination level in the area and the location: **20,000 dpm/100cm^2 at circle 5**
5.
Determine the contamination level at the work location: **6,000 dpm/100cm^2**
6.
Determine the expected dose for the duration of the job: **375 mRem**
7.
Determine the heat stress stay time: **20 minutes**

SRO Only Handout

INITIATING CUE	**(Operator Name)**, *determine if a dose extension is required to perform this work. If no extension is required, identify the expected margin to the dose limit upon completion of the task. If an extension is required, identify the form required to be completed for the dose extension and the levels of approval needed for the dose extension. Record your findings on the scorecard below."*

SRO Only Scorecard

Answer the following when performing this task:
1.
Is a dose extension required? ☐ No (answer question 2 below only) ☐ Yes (answer questions 3 and 4 below only)
2.
If a dose extension is NOT required, determine expected margin to dose limit upon completion of the job. ___ N/A, a dose extension is required
3.
If a dose extension is required, identify the form required to be completed for the dose extension. ___ N/A, a dose extension is NOT required
4.
If a dose extension is required, check all appropriate boxes below for the approvals required for this dose extension. ___ N/A, a dose extension is NOT required ☐ Work Group Supervisor ☐ Radiation Protection Manager ☐ Station Manager ☐ Site Vice President

Constellation

Examiner Scorecard. Do Not Provide to Operator.

1.
Is a dose extension required? ☐ No (answer question 2 below only) ☒ **Yes (answer questions 3 and 4 below only)**
2.
If a dose extension is NOT required, determine expected margin to dose limit upon completion of the job. _x_ N/A, a dose extension is required
3.
If a dose extension is required, identify the form required to be completed for the dose extension. ___ N/A, a dose extension is NOT required **RP-AA-203 Attachment 1**
4.
If a dose extension is required, check all appropriate boxes below for the approvals required for this dose extension. ___ N/A, a dose extension is NOT required ☒ **Work Group Supervisor** ☒ **Radiation Protection Manager** ☐ Plant General Manager ☐ Site Vice President

Administrative JPM Handout Reference Packet

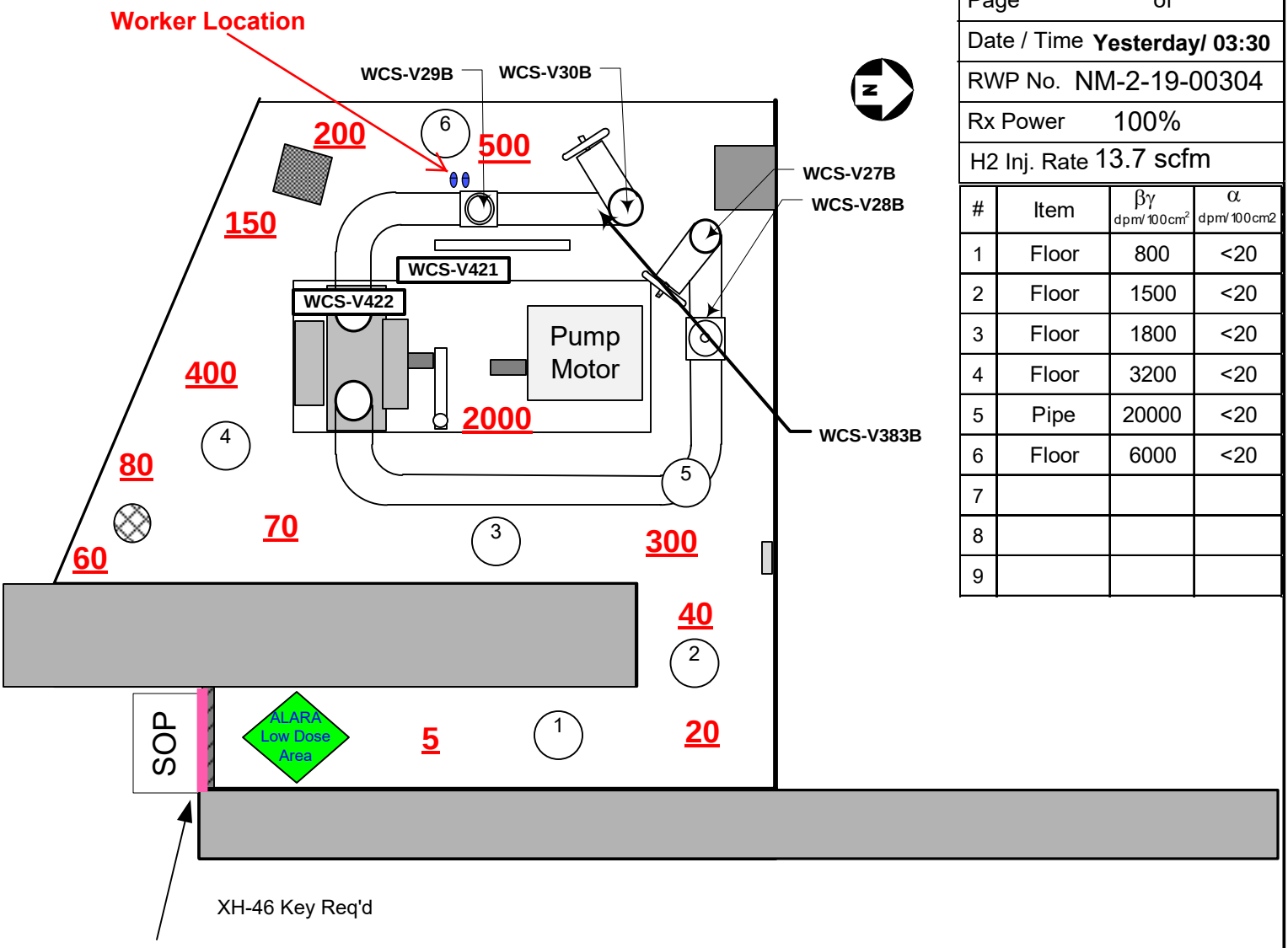
JPM Handout

INITIAL CONDITIONS	Given: • The plant is at 100% power. • A valve leak has developed in the WCS-P1B room. • Entry into the room is required to assist Maintenance with repairing valve WCS-V29B. • An updated survey map is provided. • Your current year-to-date exposure is 1800 mRem TEDE. • You have not received any dose extension this year. • Job conditions are as follows: o You will be performing Moderate Work for a total of 45 minutes at valve WCS-V29B. o You will be wearing work clothes + plastics. o Radiation Protection has determined that a respirator is not required. o The wet bulb temperature in the room is 93°F.

INITIATING CUE	**(Operator Name)**, *address the radiological and heat stress aspects of performing this work, and record your findings on the provided scorecard.*

NMP	Reactor Building 215' WCS * P1B Room	Map No. 13B

Survey No.	2RB-26030
Page	of
Date / Time	Yesterday/ 03:30
RWP No.	NM-2-19-00304
Rx Power	100%
H2 Inj. Rate	13.7 scfm



#	Item	βγ dpm/100cm²	α dpm/100cm2
1	Floor	800	<20
2	Floor	1500	<20
3	Floor	1800	<20
4	Floor	3200	<20
5	Pipe	20000	<20
6	Floor	6000	<20
7			
8			
9			

Survey performed to support Maintenance and Operations in repairs on 2WCS-V29B.

1. ▬ = Rope/Area Boundary
2. (#) = Smear Location
3. # = General Area dose rate in mRem/hr, unless noted
4. $\frac{*\#}{\#}\ \frac{*\#}{\#} = \frac{\gamma}{\gamma}\ \frac{\beta}{\beta}$ Contact / 30 cm
5. △# = Neutron (mRem/hr)
6. [A/S] = Air Sample
7. ◇# = Large Area Wipe
8. [A] = Area Rad Monitor
9. [V #] = VAMP Number

γ = Readings in mRem/hr. β = Readings in mRad/hr.

Instrument	Inst. #	Cal Due date
RO-2	5782	03/06/24
RM14	3948	03/10/24
SAC4	967	01/16/24

Surveyor: D. Falcetti Dose: 2.8 mRem Reviewed By: R. Protector Date: Yesterday

Scorecard

Answer the following when performing this task:
1.
Classify the room based on radiation level (check one): ☐ Radiation Area ☐ High Radiation Area ☐ Locked High Radiation Area ☐ Very High Radiation Area
2.
Determine the highest dose rate in the room and the location:
3.
Determine the dose rate at the work location:
4.
Determine the highest contamination level in the area and the location:
5.
Determine the contamination level at the work location:
6.
Determine the expected dose for the duration of the job:
7.
Determine the heat stress action time limit:

SRO Only Handout

INITIATING CUE	**(Operator Name),** *determine if a dose extension is required to perform this work. If no extension is required, identify the expected margin to the dose limit upon completion of the task. If an extension is required, identify the form required to be completed for the dose extension and the levels of approval needed for the dose extension. Record your findings on the scorecard below."*

SRO Only Scorecard

Answer the following when performing this task:
1.
Is a dose extension required? ☐ No (answer question 2 below only) ☐ Yes (answer questions 3 and 4 below only)
2.
If a dose extension is NOT required, determine expected margin to dose limit upon completion of the job. ___ N/A, a dose extension is required
3.
If a dose extension is required, identify the form required to be completed for the dose extension. ___ N/A, a dose extension is NOT required
4.
If a dose extension is required, check all appropriate boxes below for the approvals required for this dose extension. ___ N/A, a dose extension is NOT required ☐ Work Group Supervisor ☐ Radiation Protection Manager ☐ Station Manager ☐ Site Vice President

DELIVERING THE NUCLEAR PROMISE

nuclear matters:
my work • my plant • my industry

NUCLEAR INDUSTRY STANDARD PROCESS
Radiological Protection

Level 3 – Information Use

Radiation Protection Standard Glossary of Terms

NISP-RP-013

Revision: 1 – Fatal Flaw Review
Industry Approval Date: December 11, 2018

This is an industry document for standardizing radiation protection processes. Standard processes and requirements are established to eliminate site-specific radiation protection procedures. The Institute for Nuclear Power Operations (INPO) maintains current procedures on the INPO website. Approval authority is granted by the industry contingent on a structured review and approval process by representatives of utility radiation protection organizations.

Table of Contents

1.0 Purpose 1
2.0 Scope 1
3.0 Definitions 1
4.0 Responsibilities 14
5.0 General Requirements 14
6.0 Process Instructions 14
7.0 Records/Documentation 14
8.0 References 14
9.0 Attachments 14

1.0 Introduction

1.1. This procedure describes standard definition of radiological terms used in the nuclear power industry and used in various Nuclear Industry Standard Processes (NISP).

2.0 Scope

2.1 The processes described in this procedure are based on common approaches used in the US nuclear power industry to provide job coverage by radiation protection personnel. The radiological concerns addressed by this procedure are based on the common radiological hazards for the PWRs and BWRs operating in the US.

2.2 Member utilities are expected to use this standard to enable supplemental workers to transition between nuclear power plants with minimal site-specific training. Compliance with these instructions is expected without additional site requirements or process deviations being imposed that may require additional training or challenge the performance of supplemental workers.

2.3 This procedure will be used to train and instruct supplemental radiological protection technicians. Member utilities will implement these process requirements in site procedures and update site procedures whenever requirements or process steps in this Nuclear Industry Standard Process (NISP) are revised. Current revisions are maintained on the INPO website.

3.0 Standard Definitions

3.1 **Absorbed Dose** - The energy imparted by ionizing radiation per unit mass of irradiated material. The units of absorbed dose are the Rad and the Gray (Gy).

3.2 **Access Control Guard** - An individual who has responsibility for direct surveillance of an HRA, LHRA and/or VHRA when the locks, barricades, and/or normal physical barriers for such an area are not in place.

3.3 **Accessible Area** - An area that can reasonably be occupied by a major portion of an individual's whole body. An area is not accessible if tools or other exceptional measures are needed to access the area. For example, a tank or vessel that has its cover bolted in place is inaccessible, or an opening in a shield wall that is physically difficult to access without a ladder or mobile platform is considered inaccessible. In addition, a pool of water such as the spent fuel pool is considered inaccessible unless a diver enters the pool.

3.4 **Action Level 1 PCE** - An individual's skin or clothing is contaminated with radioactive material detected ≥ 100 ccpm to 5,000 ccpm as measured at ½" by a standard pancake Geiger-Mueller (GM) detector.

3.5 **Action Level 2 PCE** - An individual's skin or clothing is contaminated with radioactive material > 5,000 ccpm to 50,000 ccpm as measured at ½" by a standard pancake GM detector.

3.6 **Action Level 3 PCE** - An individual's skin or clothing is contaminated with radioactive material > 50,000 ccpm as measured at ½" by a standard pancake GM detector.

3.7 **Beta-Gamma to Alpha ratio** - The beta-gamma activity divided by the alpha activity

3.8 **Administrative Action Level (AAL)** - The level at which the CEDE dose is assigned. If a prospective analysis has been completed, the standard industry AAL is greater than or equal to 10 mrem.

3.9 **Aggressive Work** - Activities that have a high potential to change radiological conditions. Some examples include, but are not limited to, grinding, welding, burning, sandblasting, hydro lasing, use of solvents, and disassembly of components using power or air driven tools.

3.10 **Airborne Radioactivity Area (ARA)** - A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the values of 10CFR20, Appendix B, Table 1, Column 3 (i.e., DAC values) OR to such a degree that an individual without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 12 DAC-hours.

3.11 **Air Purifying Respirator** - A respirator with an air purifying filter, cartridge, or canister that removes specific air contaminants by passing ambient air through the air-purifying element.

3.12 **ALARA** - (Acronym for " as low as reasonably achievable") means making every reasonable effort to maintain exposure to radiation as far below the dose limits as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed material in the public interest.

3.13 **ALARA Plan** - A documented job assessment that considers the radiological conditions expected during each phase of the job and the methods and controls to minimize contamination and collective radiation exposure (person-rem). The term "ALARA Plan" is equivalent to "ALARA Action Review" and "ALARA Review".

3.14 **Alpha Level I: (Minimal)** - Relative abundance of alpha contamination is minimal. Internal exposure from alpha emitters is not likely to exceed 10% of total internal dose. The $\beta\gamma/\alpha$ ratio is greater than 30,000. Areas with low alpha activity levels, such

as less than 20 dpm/100 cm², should be assigned Level 1 Areas. Action levels are recommended to verify low abundance of alpha emitters when high contamination or high airborne radioactivity is present.

3.15 **Alpha Level II (Significant)** - Relative abundance of alpha contamination is significant. The βγ/α ratio is between 30,000 and 300. Alpha inhalation contributes from 10 to 90 percent to the total inhalation hazard. Contamination survey action levels are intended to alert radiation safety personnel of presence of alpha emitters.

3.16 **Alpha Level III (Elevated)** - Relative abundance of alpha contamination is elevated. The βγ/α ratio is less than 300. Alpha inhalation contributes 90 percent or more to the total inhalation hazard. Most smears and all air samples should be counted for alpha contamination.

3.17 **Annual Limit on Intake (ALI)** - means the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue.

3.18 **Assigned Protection Factors (APF)** - The expected workplace level of respiratory protection that would be provided by a properly functioning respirator or a class of respirators to properly fitted and trained users. Operationally, the inhaled concentration can be estimated by dividing the ambient airborne concentration by the APF.

3.19 **Barricade** – A conspicuous obstacle, such as a firmly secured rope or ribbon (by itself or used with physical barriers such as existing walls or hand railings), that completely surrounds an area and obstructs inadvertent entry.

3.20 **Barrier** - A conspicuous obstacle that blocks or is intended to block passage.

3.21 **Bioassay** - Determination of the kind, quantity, concentration and/or location of radioactive material in the human body via excreta removed from the body (i.e., in vitro bioassay) or counting equipment calibrated for external monitoring of internal radioisotope concentrations (i.e., in vivo bioassay)

3.22 **Boundary** - An established line beyond which exposure to radiation and/or radioactive material would occur (e.g., contaminated area rope, radiation tape, stanchion with a posting, or a step off pad).

3.23 **Boundary Guards** - An individual responsible for maintaining direct visual surveillance of a boundary posted for the purposes of radiographic operations in order to prevent any unauthorized entry.

3.24 **Breathing Zone Air Sample (BZAS)** - An air sample where the filter media is within approximately 12 inches of a worker's head (i.e. the nose and mouth).

3.25 **Buffer Zone** -Posting used for the area that surrounds a discrete radioactive particle area to control the migration of discrete radioactive particles from the area.

3.26 **βγ/α Ratio** – The total activity of beta-gamma emitters divided by the total activity of alpha emitters as measured by counting systems. An increasing presence of transuranics is indicated by a decreasing ratio. The term is called an Activity Ratio or the Beta-Gamma to Alpha Ratio.

3.27 **Capture Velocity** - The air velocity at any point in front of a hose or at the hose opening necessary to overcome opposing air currents and to capture the contaminated air at that point by causing it to flow into the hose. Capture velocity varies, depending on the size of the particles and the rate at which they are released into the air.

3.28 **Catch Containment** - Any of several styles of leakage gathering devices (drip funnel, drip pans, catch basin, etc.) designed to catch and funnel liquid or particulates into a suitable collection medium (i.e., floor drain, drain bottle, drum or absorbent material).

3.29 **Category 1 Quantity of Radioactive Material**– A quantity of radioactive material meeting or exceeding the category 1 threshold in Table 1 of Appendix A to 10 CFR 37. This is determined by calculating the ratio of the total activity of each radionuclide to the category 1 threshold for that radionuclide and adding the ratios together. If the sum is equal to or exceeds 1, the quantity would be considered a category 1 quantity. Category 1 quantities of radioactive material do not include the radioactive material contained in any fuel assembly, subassembly, fuel rod, or fuel pellet.

3.30 **Category 2 Quantity of Radioactive Material** - A quantity of radioactive material meeting or exceeding the category 2 threshold but less than the category 1 threshold in Table 1 of Appendix A to 10 CFR 37. This is determined by calculating the ratio of the total activity of each radionuclide to the category 2 threshold for that radionuclide and adding the ratios together. If the sum is equal to or exceeds 1, the quantity would be considered a category 2 quantity. Category 2 quantities of radioactive material do not include the radioactive material contained in any fuel assembly, subassembly, fuel rod, or fuel pellet.

3.31 **Committed Dose Equivalent (CDE) (HT, 50)** - The dose equivalent to organs or tissues of reference (T) that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

3.32 **Committed Effective Dose Equivalent (CEDE) (HE, 50)** – The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed does guarantee to the organs or tissues.

3.33 **Compartment Factors** - Factors used to weight the results from dosimeters placed over various compartments in order to relate the functional risk to the organs underlying the dosimeter to the total risk from uniform irradiation of the whole body.

3.34 **Conditional or Temporary Release** - A release of radioactive material to an individual other than RP with specific radiological restrictions or controls while outside a posted radiologically controlled area.

3.35 **Contact Reading** - Dose rate measurement taken by placing the detector housing on the surface being measured.

3.36 **Contaminated Area** - An area having smearable contamination equal to or greater than 1000 dpm/100 cm² (100 net counts per minute using a pancake frisker probe) beta- gamma or 20 dpm/100 cm² alpha.

3.37 **Continuous Air Monitor (CAM)** - A device for real-time monitoring of airborne radioactivity concentrations designed to alert workers of significant increases in airborne radioactivity. A CAM has an automatic alarm that sounds at a predetermined radioactivity level or rate of collection of radioactivity on the collection medium.

3.38 **Continuous Coverage** - Radiation Protection Technician who performs constant coverage of work activities and radiological conditions and who is available to direct or stop work activities as radiological conditions warrant. There are three types of continuous coverage:

1. **Remote Monitoring:** Used in a work area where dose rates vary due to worker position or changing dose rates. The Radiation Protection Technician has the ability to monitor the worker(s) visually or with a camera, teledosimetry, and voice communication

2. **Direct Monitoring**: Used in a work area where dose rates vary due to worker position or changing dose rates. The Radiation Protection Technician is in the line of sight with worker(s) or is able to restrict worker(s) movement by use of physical barriers. RP tech periodically checks accumulated dose for the worker(s).

3. **Indirect Monitoring** – Used in a work area where dose rates are constant. The Radiation Protection Technician in a low dose area monitoring worker (s) dose by means of maximum stay time or teledosimetry.

3.39 **Correction Factor** - The factor by which the reading of an instrument is multiplied to obtain the actual value of the quantity of the value that is being measured.

3.40 **DAC-Hour** - The product of the concentration of radioactive material in air (expressed as a fraction or multiple of the derived air concentration for each radionuclide) and the time of exposure to that radionuclide, in hours. 2000 DAC-hours equals one ALI.

3.41 **DAC Fraction (fDAC)** - The sum of each radionuclides (beta-gamma or alpha) emitting activity divided by its corresponding DAC value

3.42 **DAC fraction ratio** - The ratio of $fDAC_{\alpha}/ fDAC_{\beta\gamma}$; the term is called the DAC Fraction Ratio. This ratio shows the relative significance of transuranics in contributing to potential internal dose to workers. This value can be determined using conservative assumptions for the nuclides present or use nuclide abundances from site characterization

3.43 **DOP/PAO Test** - the process in which the integrity of a HEPA (High Efficiency Particulate Air) or ULPA (Ultra Low Penetration Air) filter is tested through the introduction of particulates. Emery 3004 Poly-Alpha-Olefin (PAO), a non-hazardous material, is often used as the industry standard for filter testing.

3.44 **Declared Pregnant Worker (DPW)** - A worker who has voluntarily informed their supervisor and Radiation Protection of pregnancy in writing at the station(s) where they access a Radiologically Controlled Area (RCA

3.45 **Deep Dose Equivalent (DDE)** – Whole body dose equivalent at a tissue depth of 1 cm (1,000 mg/cm²)

3.46 **Derived Air Concentration (DAC)** - The concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under

conditions of light work, results in an intake of 1 ALI. DACs are listed in 10 CFR 20, Appendix B, Table 2.

3.47 **Detectable Radioactivity** – Radioactivity is considered detectable whenever instrumentation gives a reproducible positive indication of its presence, i.e., a reproducible signal distinguishable from background. Detectability is different for different instrumentation and survey techniques.

3.48 **Discrete Radioactive Particles (DRP)** - Small, loose, highly radioactive particles with > 500,000 dpm or 50,000 ccpm.

3.49 **Dosimeter of Legal Record (DLR)** - Generic name used to describe primary dosimeters (e.g. TLD, OSL, etc.).

3.50 **Effective Dose Equivalent (EDE)** - The sum of the products of the tissue or organ weighting factors from 10CFR20, and the dose to the corresponding body tissues and organs resulting from the exposure to radiation sources external to the body.

3.51 **Effective Alpha DAC** - Concentration of gross alpha radioactivity in air (from transuranic radionuclides), that if breathed for 2,000 hours, results in an intake of one effective alpha ALI. The effective Alpha DAC is based on Am-241 or site-specific value.

3.52 **Effective Beta DAC** – Concentration of gross beta radioactivity in air that is breathed for 2000 hours results in an intake of one effective ALI. The Effective Beta DAC Value is based on Co-60 ($1X10^{-8}$ µCi/cm^3) or site specific determined value.

3.53 **Emergency Action Level (EAL)** - A pre-determined, site-specific, observable threshold for a plant condition that places the plant in an **emergency** classification.

3.54 **Expected/Anticipated Dose Rate Alarm** – A planned and briefed intermittent or momentary dose rate alarm resulting from anticipated dose rates encountered in the travel path to the work area or anticipated temporary work area dose rate changes, such as leaning into a pipe.

3.55 **External Dose** - That portion of dose equivalent received from sources of exposure outside the body.

3.56 **Extremity** - Hand, elbow, arm below the elbow, foot, knee, or leg below the knee.

3.57 **Eye Dose Equivalent** – Applies to the external exposure of the lens of the eye, is the dose equivalent at a tissue depth of 300 mg/cm^2 (0.3cm).

3.58 **Facial Contamination** - Contamination on the face within what is generally accepted as the sealing surface of a full-face respirator.

3.59 ***$fDAC_{\alpha}$*** – The sum of each alpha emitting nuclide's activity divided by its corresponding DAC value; the term is called the Alpha DAC Fraction. In some cases, the total alpha activity may be divided by the most restrictive nuclide DAC value (e.g. Am-241) for an approximation of the Alpha DAC Fraction.

3.60 ***$fDAC_{\beta\gamma}$*** – The sum of each beta-gamma emitting nuclide's activity divided by its corresponding DAC value; the term is called the Beta-Gamma DAC Fraction. In some cases, the total beta-gamma activity may be divided by the most restrictive nuclide (e.g. Co-60 or Cs-137) for an approximation of the Beta-Gamma DAC Fraction.

3.61 ***$fDAC_{Total}$*** – The sum of $fDAC_{\beta\gamma}$ and $fDAC_{\alpha}$; the term is called the Total DAC Fraction.

3.62 **Field Count** - A quantitative analysis of an air sample or smear to assess contamination using a field survey instrument. Analysis results are used for prioritizing filter analysis

3.63 **Functional Check** - A check (often qualitative) to determine that an instrument is operational and capable of performing its intended function. Such checks may include, for example, battery check, zero setting, or source response check.

3.64 **General Air Sampling** - Air sampling performed to monitor or trend the radioactive airborne particulate and/or iodine concentration (particulate, iodine, noble gas or tritium) in a given area, over an extended period of time. Sample duration (other than CAMs) may vary based on events rates of change and filter-loading rates.

3.65 **General Area Survey** – A dose rate survey performed in the general area at least 30 cm from the radiation source or from any surface that radiation penetrates.

3.66 **General Radiation Work Permit (RWP)** - A permit that controls work tasks that have minimal radiological risk, significant radiation exposure or the potential to spread contamination. A General RWP is customarily used in areas with little or no loose surface contamination or airborne activity, and low direct gamma radiation. General RWPs are for routine work (inspections, rounds), do not involve work with complex radiological conditions, and radiological conditions are static. Additionally, general RWPS are normally not used for conditions >100 mrem/hr at 30 cm.

3.67 **Grab Sampling** - A general term used to denote air samples taken over a short time span (typically 10 to 15 minutes) to determine peak air concentrations. The technique is used to determine posting of area.

3.68 **Hard to Detect Radionuclides** - A term used to describe radionuclides, which due to their type, energy, or abundance of emission, are not readily detected by a specified instrument. For example, Co-58, Cr-51, Mn-54 and TRU isotopes are commonly identified as HTD radionuclides, but they are only hard to detect with instruments such as GM friskers, which are primarily beta-sensitive. Of these three

radionuclides, only Co-58 emits beta particles, but then in only approximately 15% of all decays. With a gamma-sensitive article monitor, only Cr-51 would be classified as HTD since it emits a gamma ray in only about 15% of all decays; Mn-54 and Co-58 would not be HTD since they emit gamma rays with sufficient energies and abundances. C-14, however, would be classified as HTD for both of the above instruments, as it emits only low energy beta particles. Also referred to as "difficult-to-detect" (DTD) radionuclides.

3.69 **HEPA Exhaust Protective Guard:** A structure that can be secured to the HEPA exhaust to prevent workers from inadvertently placing their hand or fingers in the line of fire with internal rotating parts (impeller, fan blades, etc.). The guard must not have openings greater than one-half inch in width. The use of concentric rings with spacing between them not exceeding a one-half inch are acceptable, provided that sufficient radial spokes and firm mountings are used to make the guard rigid enough to prevent it from being pushed into the fan blade during normal use.

3.70 **High Contamination Area (HCA)** - An area where the majority of the area has removable surface contamination equal to or greater than 100,000 dpm/100cm² beta-gamma.

3.71 **High Efficiency Particulate Air (HEPA) Filter** - A filter that provides highly efficient filtration of airborne particulate matter with a minimum of 99.97% efficiency for removing particles of greater than 0.3 microns in diameter.

3.72 **High Radiation Area (HRA)** - Any area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

3.73 **Hot Spot** - Accessible hot spots when components have contact readings of more than 100 mrem/hour and more than five times the general area dose rates (typically 30 cm from the source).

3.74 **Individual Monitoring** – The assessment of dose equivalent using devices designed to be worn by an individual.

3.75 **Intake** - Radioactivity that enters the body through the respiratory tract, the gastrointestinal tract, or the skin. Intake may be acute, meaning a single intake occurring over a very short time, usually taken to be instantaneous, or chronic, occurring over a specified time. Common units used in this guide for intake are microcuries (μCi) and kilo-becquerel (kBq).

3.76 **Intermittent Coverage** - Cognizance of activities within the job and radiological surveillance consistent with the radiological hazards (e.g., touring the job site and having full knowledge of the work and radiological hazards associated with the job and individuals involved).

3.77 **Internal Dose** - That portion of the dose equivalent received for radioactive material taken in the body.

3.78 **Irradiated In-Core Components (ICC)** - Reactor vessel components that are irradiated, such as spent fuel, nuclear instrumentation (NI) control rod blades (CRB) or drives (CRDs), dry tubes, fuel support pieces, jet pump, core barrels, baffle plates and components.

3.79 **Large Area Smear (LAS) Survey** - A qualitative contamination survey performed by wiping a large area and monitoring the cloth with a frisker survey instrument.

3.80 **Lens Dose Equivalent** - The external exposure dose equivalent to the lens of the eye at a tissue depth of 0.3 centimeters (300 mg/cm2).

3.81 **Locked High Radiation Area (LHRA)** - Any area accessible to individuals in which deep dose equivalent rates are greater than or equal to 1 rem per hour (but less than 500 rads in one hour at 1 meter) 30 centimeters from the source of radiation or from any surface that the radiation penetrates.

3.82 **LHRA Physical Barrier** - Physical barriers (such as chain link fencing or fabricated walls) used to prevent unauthorized personnel access to locked high radiation area (LHRA). Barriers used to control access to a locked high radiation areas should provide reasonable assurance that they secure the area against unauthorized access and cannot be easily circumvented. (That is, an individual who incorrectly assumes, for whatever reason, that he or she is authorized to enter the area, would be unlikely to disregard and/or circumvent the barrier.) A fence that is a minimum of 2 meters (approximately 6 ft.) high would normally be adequate to control access to a locked high radiation area at a nuclear power plant.

3.83 **Low Dose Waiting Area (LDWA)** - An area designated to minimize exposure to personnel where significant exposure savings may be realized.

3.84 **Lower Limit of Detection (LLD)** - The smallest concentration of radioactive material in a sample that will yield a net count above system background.

3.85 **Material Release Plan** - A written plan describing the survey requirements for removing material from the RCA when that material does not fall within the bounds of the procedure for release of material.

3.86 **Member of the Public** - Any individual except when that individual is receiving occupational dose.

3.87 **Micro ALARA Plan (MAP) -** A documented review and planning of work below the threshold for initiation of a formal ALARA Plan.

3.88 **Minimum Detectable Activity (MDA)** - The smallest quantity of radioactivity that could be distinguished from the blank under specified conditions. The MDA depends

on the lower limit of detection and on the counting efficiency of the counting system. (Cember, Introduction to Health Physics)

3.89 **Nasal Contamination** - Personnel contamination in or around the nasal tract identified by direct frisk or nasal blow.

3.90 **Naturally Occurring Radioactive Material (NORM)**- Radioactive material that consists of radionuclides found in the environment that are not a result of licensee activity. NORM can include uranium, radium, thorium and their various decay products.

3.91 **Neutron Area** – An area where the neutron dose rate is > 4 mrem/hour General Area at 30 cm from a source.

3.92 **Non-Uniform Field**- Work area dose rate gradients make it likely that total dose to a portion of the whole body will exceed the chest dose by more than 50 percent.

3.93 **Occupational Dose** - The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released under 10CFR35.75, from voluntary participation in medical research programs, or as a member of the public.

3.94 **Personal Air Sample** - Sample collected by a small, lightweight air sampler worn by the worker that draws an air sample from the breathing zone (also known as a Lapel Air Sample).

3.95 **Personal Clothing** - Articles of clothing other than anti-contamination protective clothing (excluding hard hats and safety glasses).

3.96 **Personal Items** – Items normally carried by personnel while in the RCA. This does not include plant tools or equipment. Radiation Protection will develop and post a list of personal items at each RCA egress area. Personal items may be monitored by the individual possessing these items.

3.97 **Positive Control** - Control required by an individual assigned to prevent inadvertent entry into a LHRA or VHRA by unauthorized personnel. This control is provided by an individual positioned at a point sufficient to prevent inadvertent entry into the area by unauthorized personnel.

3.98 **Protection Factor** – A measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne contaminant outside the

respiratory protection equipment to that inside the equipment (usually inside the facepiece under conditions of use)

3.99 **Radiation Area**- An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

3.100 **Radiation Work Permit (RWP)** - A document that provides a method for documenting and controlling work with potential or actual radiological hazards in a Radiologically Controlled Area (RCA)

3.101 **Radioactive Material Area** - An area in which licensed radioactive material in an amount exceeding 10 times the quantity specified in Appendix C, 10CFR20, is used or stored. This does not apply to radioactive materials contained within process equipment or materials in transport and packaged and labeled in accordance with appropriate regulations.

3.102 **Radioactive Material Label** - A label or tag which contains the standard radiation symbol and the words, “Caution Radioactive Material” or “Danger Radioactive Material” and has sufficient information, (dose rates, contamination levels or other information deemed necessary), to permit workers handling the material to keep their exposures ALARA.

3.103 **Radiography** - An examination of the structure of materials by nondestructive methods, utilizing ionizing radiation to make radiographic images.

3.104 **Radiography Area** - A radiographic work area posted and controlled in accordance with the licensee (radiography group) Operating and Emergency Procedures.

3.105 **Radiography Restricted Area** - An area with established boundaries defined and posted by Radiation Protection for assuring that unauthorized personnel are prevented from entering the radiography area

3.106 **Radiologically Controlled Area (RCA)** - An area within the restricted area posted in accordance with procedures for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

3.107 **Radiological Medium Risk Activity** - Radiological work where planned barriers are desirable to prevent inadequately controlled radiation levels, unplanned/unmonitored internal and external dose, minimize potential for EPRI level 2 or 3 personnel contamination events, or potential contamination of non-radiological facilities or the environment within the protected area.

3.108 **Radiological High-Risk Activity** - Radiological work where detailed planning and multiple, diverse barriers are essential to prevent radiological events involving significant radiation levels, threats to individual regulatory radiation exposure limits,

or may result in unanalyzed effluent release pathways to the environment or exposure to members of the public.

3.109 **Discrete Radioactive Particle (DRP) Area** - Posting used for an area that is controlled due to the presence (or concern) of discrete radioactive particles greater than 500,000 dpm (50,000 ccpm).

3.110 **RP Hold Point** - Any point specifically designated by Radiation Protection in a Radiation Work Permit, Work Order, Procedure or RP briefing, where an action by RP personnel is conducted prior to execution of the next work step or activity.

3.111 **RP Self Briefing** - The process where Radworkers can brief themselves to the radiological conditions without having to interface directly with radiation protection

3.112 **Self Reading Dosimeter (SRD)** - A pocket ionization chamber (PIC) or electronic dosimeter (ED) worn by the worker and used to provide an immediate indication of the estimated dose received by the worker.

3.113 **Shallow Dose Equivalent (SDE)** - The external exposure of the skin of the whole body or an extremity taken at a tissue depth of 0.007 cm (7 mg/cm^2)

3.114 **Special Nuclear Material (SNM)** - Plutonium, Uranium-233, Uranium enriched in the isotope 233 or the isotope 235, and any other material that the NRC, pursuant to the provisions of Action 51 of the Act, determines to be special nuclear material, but does not include source material. Any material artificially enriched by any of the foregoing, but does not include source material.

3.115 **Stay Time** - A calculated time an individual may stay in a work area of a given dose rate without exceeding a predetermined dose.

3.116 **Stop Work Authority** - Authority granted to RP personnel to cease any work when worker radiological safety is jeopardized

3.117 **Survey** – An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes the physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present.

3.118 **TEDE ALARA Evaluation** - An ALARA evaluation performed to determine whether or not respirators should be used. The TEDE ALARA evaluation estimates the benefits of internal dose avoided compared with any additional external dose accumulated as a result of wearing a respirator. It should also consider industrial and safety risks associated with respirator use.

3.119 **Tool Equipment Monitor (TEM)**- Generic name given for the various automated devices to count tools and equipment.

3.120 **Total DAC** - The summation of individual DAC values for particulate (beta, alpha), iodine, noble gas, and tritium. Each may not pertain to an air sample, but the total DAC is the summation of each DAC value

3.121 **Total Effective Dose Equivalent (TEDE)** - The sum of the effective dose equivalent (for external exposure) and the committed effective dose equivalent (for internal exposure).

3.122 **Transport Container** - A box, drum, enclosed trailer, sealand, rail car or other approved RAM container that is of robust construction and exhibit essential design features of packages intended for highway transport such that there is low potential for packaging failure during the period of storage.

3.123 **Transuranic Nuclides** - The transuranium elements (also known as **transuranic** elements) are the chemical elements with atomic numbers greater than 92 (the atomic number of uranium). All of these elements are unstable and decay radioactively into other elements.

3.124 **TRU multiplier** - A value equal to 1 + $DAC_{FractionRatio}$ that can be multiplied by the $fDAC_{\beta\gamma}$ to estimate the $fDAC_{Total}$.

3.125 **Unconditional Release** - Material or equipment that has **no** detectable licensee generated material above background and therefore may be released from the site for unrestricted use.

3.126 **Unexpected/Unanticipated Dose Rate Alarm** - Any valid dose rate alarm that is due to: a change in radiological conditions, an inadequate radiological survey, or poor radiation worker practices such as entering an area where radiological conditions were not communicated.

3.127 **Unrestricted Area** – An area to which access is **neither** limited **nor** controlled by the licensee.

3.128 **Uptake -** The quantity of material that enters the body fluids from the respiratory tract, the gastrointestinal tract, or through the skin. The term also is sometimes used to indicate material taken into a tissue or organ from circulation. Common units used in this guide for uptake are µCi and kBq.

3.129 **Very High Radiation Area (VHRA)** - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates.

3.130 **Weighting Factor** - A factor, W_T, representing the proportion of the stochastic risk resulting from a particular organ or tissue to the total risk to the individual when the whole body is irradiated uniformly.

3.131 **Whole Body** - That part of the body including: the head, chest, back, gonads, arms above the elbows, and legs above the knees.

4.0 Responsibilities

4.1 Radiation Protection is responsible for the implementation of the requirements of this procedure per Efficiency Bulletin 17-01 and the Nuclear Industry Standard Process Initiative.

5.0 General Requirements

NONE

6.0 Process Instructions

NONE

7.0 Records/Documentation

NONE

8.0 References

NONE

9.0 Attachments

NONE

Training ID: **Nov 2023 NRC JPM SRO EP** Revision: **0.0**

Title: **Emergency Plan Classification (Alert, EAL CA5)**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mecki Ghabarou	7/12/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 20 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. EP-AA-1013 Addendum 4 Appendix 1, NMP Unit 2 EAL Wallboard
2. EP-AA-1013 Addendum 4, Nine Mile Point Nuclear Station Unit 2 Emergency Classification Technical Bases
3. EP-AA-112-100-F-01, Shift Emergency Director Checklist
4. EPA-28, NMP EP AID
5. NUREG 1123 K/A 2.4.41 (4.6)

Instructor Information

A. JPM Information

1. Description

   a. This JPM is to evaluate the SRO Operator's ability to classify an event and perform the Station specific Initial Notification Form in accordance with Emergency Preparedness Procedures.

   b. This JPM is not considered Alternate Path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded Pass/Fail.

2. Task Information:

   a. NS-EP101-03004, Analyze Indications to Determine That an Emergency Plan Event is in Progress (PRA)

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO ☐ RO ☐ EO ☐ N/A			
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☐ Simulator ☒ Classroom			
Time Critical Task:	☒ Yes ☐ No			
Alternate Path:	☐ Yes ☒ No			
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	2.4.41 - Knowledge of the emergency action level thresholds and classifications			
K/A Importance Rating:	RO	-	SRO	4.6

4. Recommended Start Location

   a. NLC Classroom

5. JPM Setup (if required)

   a. Ensure adequate copies of EP-AA-1013 Addendum 4 are available.

   b. Ensure adequate copies of NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1 are available.

   c. Ensure adequate copies of EP-AA-112-100-F-01 are available.

   d. Ensure adequate copies of NMP EP AID EPA-28 are available.

   e. Ensure electronic Part 1 Notification Fact Sheet is available for use.

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is in mode 5 and was Shutdown 3 days ago at midnight • RPV head de-tensioning is in progress • The equipment hatch was removed 3 hours ago • RHR A tripped approximately three hours ago; RHR B was placed in SDC • RHR B has since tripped. RCS Temperature has risen to 207°F, and exceeded 200°F 20 minutes ago • RPV pressure is 0 psig • Unit 1 is operating at 100% power • No release is in progress • Containment is established **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *classify this event and complete the station specific initial notification form in accordance with EP-AA-112-100-F-01 SHIFT EMERGENCY DIRECTOR CHECKLIST. This JPM is Time Critical.*

Last 15 Minute Emergency Meteorology Report Data

Data from Nine Mile Point Met System

Date: XX/XX/XXXX Time (Local): XX:XX:XX

Elevated				Ground			
200'	Wind Speed (Main)	37	(mph)	30'	Wind Speed (Main)	32	(mph)
200'	Wind Dir From (Main)	352	(deg)	30'	Wind Dir From (Main)	350	(deg)
200'	Delta Temperature	0.05	(deg F)	100'	Delta Temperature	0.7	(deg F)
	Stability Class	E			Stability Class	C	
30'	Air Temperature	50	(deg F)		Precipitation (15 min)	1.02	(in)

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
2.	Obtains a copy of the reference procedure and review / utilize the correct section of the procedure.	P	SAT / UNSAT ***STD:*** *Obtains and references EP-AA-112-100-F-01.*
Evaluator Note:	• Steps may not be performed in the order exactly as listed in the JPM. • The Operator may choose to utilize the Simplified Classification Sequence guide NMP EP AID EPA-28.		
3.	Starts EP-AA-112-100-F-01, Shift Emergency Director Checklist	P (EP-AA-112-100-F-01)	SAT / UNSAT ***STD:*** *Obtains EP-AA-112-100-F-01 packet and reviews for required actions. Determines that section 1.2 is applicable.*
4.	If the event is classified as an Alert, then PERFORMS the following:	P (1.2)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
5.	**RECORDS** the EAL and declaration time EAL/Threshold _____ Declaration Time _____	P (1.2.A)	***PASS / FAIL** ***STD:*** *Determines the appropriate classification is an Alert, EAL CA5 and records the EAL/Threshold code and Declaration Time on EP-AA-112-100-F-01 in the space provided. Emergency classification and declaration shall be completed as soon as possible but no later than 15 minutes from the time indications of an EAL threshold being met or exceeded are available in the Control Room. The time written on the Shift Emergency Director Checklist or SED job aid will count as declaration time if no announcement is made.* ***The time difference below must be 15 minutes or less:*** *JPM Start Time: _____* *Declaration Time: _____* ***Failure =*** *"CA5" recorded in the EAL/Threshold block and Declaration Time determined within 15 minutes of JPM start time and recorded in the Declaration Time block not achieved.*
6.	**ANNOUNCES** the event classification, possible escalation paths, and declaration time to the Control Room staff. ***Cue****: If asked for a peer check, report that the STA/IA is not available for peer check* **DECLARE** the event by announcing the following (from EPA-28): I am declaring a(n) (EAL Classification) at (time) due to (brief reason). Escalation criteria is(are) (Criteria) and I am assuming the role as Shift ED."	P (1.2.B)	SAT / UNSAT ***STD:*** *Announces to Control Room personnel the EAL classification, possible escalation paths, and declaration times.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
7.	**CALLS** the Shift Communicator to the Control Room.	P (1.2.C)	SAT / UNSAT ***STD:*** *Utilizes plant communication methods to call the Shift Communicator to come to the control room. Proper communications used.*
Evaluator Note:	• For the next JPM step, it is not expected that a radiological assessment would need to be performed with the given conditions, therefore the Operator may or may not choose to call the Shift Dose Assessor to the control room.		
8.	**If <u>not</u>** previously performed and if radiological assessment is required as a result of the emergency condition, **then CALLS** the Shift Dose Assessor and the Shift RP Tech to the Control Room (or a pre-designated location).	P (1.2.D)	SAT / UNSAT ***STD:*** *Reads/reviews and may call the Shift Dose Assessor and the Shift RP Tech to the control room and placekeeps step accordingly.*
Evaluator Note:	For the next step, there is no security event in progress. The Operator should placekeep the step, but not reference section 4.1.		
9.	Security Related Events, **GOES TO** Step 4.1 Security Related Events.	P (1.2.F)	SAT / UNSAT ***STD:*** *Reads/reviews and determines step to be Not Applicable and placekeeps step.*
10.	If the ERO has **<u>not</u>** been activated, **then DIRECTS** notification of the ERO using 01, "Actual Event Respond to Facility," or 03, "Actual Event Alternative Facility Response," as appropriate, per EP-AA-112-100-F-06. (Tab 2) ***Cue****: As Shift Communicator, acknowledge direction to notify ERO* **Note:** Record time SED directs the Shift Communicator to activate the ERO as recorded in the SED Checklist	P (1.2.G)	SAT / UNSAT ***STD:*** *Directs Shift Communicator to perform notification of the ERO using EP-AA-112-100-F06. Proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
11.	**SELECTS** the Emergency Public Address Announcements from the form **and DIRECTS** performance of the public address announcement within 15 minutes of event classification. (Tab 1) ***Cue**: As Shift Communicator, acknowledge direction. Wait 1 minute and then report that Station announcement has been made.*	P (1.2.H)	SAT / UNSAT ***STD**: Completes the PA announcement form and directs Shift Communicator to make Emergency PA Announcement. Proper communications used.*
Evaluator Note:	For the next JPM step, step 2 on EP-CE-114-100-F-05 Part 1 Notification Fact Sheet is to be treated a SAT/UNSAT step and therefore will not be graded as PASS/FAIL criteria.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
12.	**DIRECTS** performance of State/Local notifications within 15 minutes of the event classification as required per the Notifications procedure. Notification Procedure: (Tab 3) Notification form: (Tab 4) Release in Progress Determination Guidance: (Tab 21) Release In Progress Determination Guidance Technical Basis (Tab 26) ***Cue**: If operators asks for a peer check, report that the STA/IA is not available for peer check* Provides the completed form to Shift Communicator and directs them to notify State and Local. ***Cue:** As Shift Communicator, acknowledge direction and record time Part 1 Notification received.*	P	***PASS / FAIL** ***STD:** Accurately fills in and completes all information inside BLACK boxed in areas of Part 1 Notifications.* *Step 3: B* *Step 4: Date /Time* *Step 5: A* *Step 6: A* *Step 7: "NMP U2" Box Checked* *Step 7: CA5* *Step 9 -Wind speed filled in correctly* *Step 10 - Wind direction filled in correctly* *This notification is required to be completed within 15 minutes of the Emergency Declaration. This means that the state and local are required to be notified within 15 minutes.* ***The time difference below must be 15 minutes or less:*** *Declaration Time: _____* *Notification Time: _____* ***Failure =** The time difference between Declaration Time and Notification Time being less than 15 minutes not achieved.*
Evaluator Note:	Once the notification form has been completed. Provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*Event has been classified and the Notification Form has been completed.*

STOP TIME	

Part 1 Notification Form Evaluator Answer Key

Do not Provide to Operator

Exelon Generation

EP-CE-114-100-F-05
Revision I
Page 1 of 1

PART 1 NOTIFICATION FACT SHEET

Step Change	Step	
		"This is to report an incident at _____ standby for roll call." ☐ Nine Mile Point Unit 1 ☐ Nine Mile Point Unit 2 ☐ James A. Fitzpatrick
		☐ NY State ☐ Oswego County ☐ James A. Fitzpatrick ☐ NMP Unit 1 ☐ NMP Unit 2 **Notification #**
	1.	**This message is being transmitted on**:_____ (Date) at _____ (Time) via: **A.** RECS **B.** Other
	2.	**This is:** **A.** An actual emergency **B.** An Exercise
	3.	**The Emergency Classification / Condition is:** **A.** Unusual Event **B.** Alert **C.** Site Area Emergency **D.** General Emergency **E.** Recovery **F.** Termination
	4.	**This Emergency Classification / Condition was declared on**: (date)_____Date_____ at (time)_____Time_____
	5.	**Release of radioactive materials due to the classified event:** **A.** No radiological release in-progress **B.** AIRBORNE radiological release in-progress **C.** LIQUID radiological release in-progress **D.** Radiological release TERMINATED
	6.	**The following Protective Actions are recommended to be implemented as soon as practical:** **A.** No need for protective actions outside the site boundary **B. EVACUATE and IMPLEMENT** the KI Plan for the following ERPAs **and** all remaining ERPAs monitor the Emergency Alert System. 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 **C. SHELTER and IMPLEMENT** the KI Plan for the following ERPAs **and** all remaining ERPAs monitor the Emergency Alert System. 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29
	7.	☐ NMP U1 ☒ NMP U2 ☐ JAF **EAL #** CA5 **Brief event description and other significant information:** (Include any Security Related EAL's if applicable)
	8.	**Reactor Status:** NMP U1: **A.** Operating **B.** Shutdown (date):_____ at (time):_____ NMP U2: **A.** Operating **B.** Shutdown (date):_____ at (time):_____ JAF: **A.** Operating **B.** Shutdown (date):_____ at (time):_____
	9.	**A. Elevated wind speed** 37 miles/hr **B. Ground wind speed** 32 miles/hr
	10.	**A. Elevated wind direction** (from) 352 deg **B. Ground wind direction** (from) 350 deg
	11.	**A. Elevated Stability Class:** A B C D E F G **B. Ground Stability Class:** A B C D E F G
	12.	**Reported by:** (Name)_____ at Tel. No. (___)_____ at time _____
		ASK: "**Does Oswego County or New York State need further clarification on any information?**" ***THEN***: "**This is the end of the message, standby for verification roll call.**"
		☐ New York State ☐ Oswego County ☐ James A. Fitzpatrick ☐ NMP Unit 1 ☐ NMP Unit 2
		Then State: "This concludes the notification message".
		Approved by: (Shift ED or Corporate ED): Print Name: _____ Signature: _____

Document Retention SRRS.ID.5B.100

SRO JPM Handout

INITIAL CONDITIONS	Given: • The plant is in mode 5 and was Shutdown 3 days ago at midnight • RPV head de-tensioning is in progress • The equipment hatch was removed 3 hours ago • RHR A tripped approximately three hours ago; RHR B was placed in SDC • RHR B has since tripped. RCS Temperature has risen to 207°F, and exceeded 200°F 20 minutes ago • RPV pressure is 0 psig • Unit 1 is operating at 100% power • No release is in progress • Containment is established

INITIATING CUE	**(Operator Name)**, *classify this event and complete the station specific initial notification form in accordance with EP-AA-112-100-F-01 SHIFT EMERGENCY DIRECTOR CHECKLIST. This JPM is Time Critical.*

Last 15 Minute Emergency Meteorology Report Data

Data from Nine Mile Point Met System

Date: XX/XX/XXXX Time (Local): XX:XX:XX

Elevated				Ground			
200'	Wind Speed (Main)	37	(mph)	30'	Wind Speed (Main)	32	(mph)
200'	Wind Dir From (Main)	352	(deg)	30'	Wind Dir From (Main)	350	(deg)
200'	Delta Temperature	0.05	(deg F)	100'	Delta Temperature	0.7	(deg F)
	Stability Class	E			Stability Class	C	
30'	Air Temperature	50	(deg F)		Precipitation (15 min)	1.02	(in)

Administrative JPM Handout Reference Packet

SRO JPM Handout

INITIAL CONDITIONS	Given: • The plant is in mode 5 and was Shutdown 3 days ago at midnight • RPV head de-tensioning is in progress • The equipment hatch was removed 3 hours ago • RHR A tripped approximately three hours ago; RHR B was placed in SDC • RHR B has since tripped. RCS Temperature has risen to 207°F, and exceeded 200°F 20 minutes ago • RPV pressure is 0 psig • Unit 1 is operating at 100% power • No release is in progress • Containment is established

INITIATING CUE	**(Operator Name)**, *classify this event and complete the station specific initial notification form in accordance with EP-AA-112-100-F-01 SHIFT EMERGENCY DIRECTOR CHECKLIST. This JPM is Time Critical.*

Last 15 Minute Emergency Meteorology Report Data

Data from Nine Mile Point Met System

Date: XX/XX/XXXX Time (Local): XX:XX:XX

Elevated				Ground			
200'	Wind Speed (Main)	37	(mph)	30'	Wind Speed (Main)	32	(mph)
200'	Wind Dir From (Main)	352	(deg)	30'	Wind Dir From (Main)	350	(deg)
200'	Delta Temperature	0.05	(deg F)	100'	Delta Temperature	0.7	(deg F)
	Stability Class	E			Stability Class	C	
30'	Air Temperature	50	(deg F)		Precipitation (15 min)	1.02	(in)

Training ID: **Nov 2023 NRC JPM A** Revision: **0.0**

Title: **HPCS Pump Run Following Maintenance (Alternate Path)**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Dan Delaney	7/12/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-33, Rev. 01800, High Pressure Core Spray System
2. N2-ARP-601700, 2CEC*PNL601 Series 700 Alarm Response Procedures
3. NUREG 1123 K/A 209002 A4.01 (4.3)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to manipulate controls associated with the High Pressure Core Spray System. The operator will perform a HPCS pump run following maintenance in accordance with N2-OP-33.

   b. This JPM is considered alternate path because when the test return to condensate storage tank valves are opened per procedure and the HPCS min flow valve closes, 2CSH*P1 (HPCS Pump) will experience a pump motor overcurrent condition, but fail to trip as designed. The Operator will evaluate panel indications and reference the applicable annunciator response procedure and place 2CSH*P1 control switch in P-T-L.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-209002-01003, Shutdown the High Pressure Core Spray System to Inoperable

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A	
Evaluation Method:	☒ Perform ☐ Simulate				
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom				
Time Critical Task:	☐ Yes ☒ No				
Alternate Path:	☒ Yes ☐ No				
Safety Function:	2		Reactor Water Inventory Control		
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	209002 High-Pressure Core Spray System A4.01 - Ability to manually operate and/or monitor in the control room: HPCS pump				
K/A Importance Rating:	RO	4.3	SRO	-	

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) None

2) Remotes

a) **CS02**, 2CSH*MOV110 Appendix R Ckt Breaker, Final = Close — **Inserted**

3) Overrides

a) **AO-154**, 0-600 AMP Ammeter, Final = 600 — **TRG1**

4) Annunciators

a) **an601730**, hpcs pump 1 motor overcurrent, Final = Crywolf — **TRG1**

5) Event Triggers

Event #	Event Action	Command
1	hzacshr604>0.35 .AND. hzacshr606>0.35	Left Blank
2	zdcshps2(5)==1	dor ao-154
3	zdcshps2(5)==1	dmf an601730

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-OP-33 with steps H.17.1, H.17.2.a and H.17.2.b placekept as complete. Next step to be performed is H.17.3.

8) Miscellaneous

a) IC-204 (For ILT 22-2, paired with JPM G)

-OR-

b) IC setup:

(1) None

6. Strategy Code

a. None

7. Tools and Equipment

a. None

8. Commitments

   a. None

9. Prerequisites

   a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant operating at rated power. • The plant process computer is out of service for corrective maintenance. • Minor maintenance was conducted on the HPCS system. • HPCS has been declared inoperable. • The maintenance activity has been completed. • N2-OP-33, section E.1 Fill and Vent has been completed satisfactorily. • Breaker 2EHS*MCC201-6B, 2CSH*MOV110-COND STG TK TEST BP VALVE is ON and the ALARM CIRCUIT control switch is in ENABLE. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, p*erform a HPCS Pump Run Following Maintenance in accordance with N2-OP-33, section H.17.0. The procedure is in progress up to step H.17.3.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	Verifies 2CSH*MOV110, TEST RETURN TO CONDENSATE TK THROTTLE valve indicates 0%, CLOSED	P (H.17.3)	SAT / UNSAT ***STD:*** *Observes 2CSH*MOV110, TEST RETURN TO CONDENSATE TK THROTTLE percent valve position meter reading 0% on 2CEC*PNL601.*
Procedure Note:	Starting 2CSH*P1 with the suction from the CST can result in Gross Failure alarms on trip units E22-N652, HPCS PMP SUCT LO, and E22-N653, HPCS PMP SUCT HI.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.	STARTS 2CSH*P1, HPCS PUMP 1.	P (H.17.4)	***PASS / FAIL** ***STD:*** *Rotates the control switch for 2CSH*P1, HPCS PUMP 1 on 2CEC*PNL601 clockwise to the Normal-After-Start position and observes the RED light lit and GREEN light not lit.* ***Failure =*** *2CSH*P1 RED light ON and GREEN light OFF not achieved.*
4.	VERIFIES 2CSH*MOV107, PMP 1 INJECTION VLV, is closed.	P (H.17.5)	SAT / UNSAT ***STD:*** *Observes 2CSH*MOV107, PMP 1 INJECTION VLV GREEN light lit and RED light not lit on 2CEC*PNL601.*
5.	OBSERVES 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, opens.	P (H.17.6)	SAT / UNSAT ***STD:*** *Observes 2CSH*MOV105, MINIMUM FLOW BYPASS VLV RED light lit and GREEN light not lit on 2CEC*PNL601.*

Procedure Caution:	• Throttling HPCS return flow to CST with only one throttle valve can cause extensive valve damage. • With HPCS operating in full flow test with suction and return to the CSTs, "A" CST will fill up and overflow if level is not monitored.
Evaluator Note:	If the Operator requests 'A' CST level, provide the following cue: **Cue:** *'A' CST level currently reads 41 feet*
Alternate Path:	During performance of the following step, 2CSH*P1 will experience a motor overcurrent condition with a failure of 2CSH*P1 to trip. The Operator will recognize that annunciator 601730, "HPCS Pump 1 Motor Overcurrent" is in alarm and that that 2CSH*P1 current ammeter indicates upscale with 2CSH*P1 still running. The Operator will reference the ARP and place the pump control switch in P-T-L.

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Evaluator Note:	The PASS/FAIL criteria for the next step does not require the Operator to achieve the procedural directed flow rate of 6350 gpm to 6450 gpm because the alternate path event trigger will initiate prior to achieving this flow value.		
6.	THROTTLES OPEN 2CSH*MOV110 AND 2CSH*MOV112, TEST RETURN TO CONDENSATE TK, in equal increments to establish 6350 gpm (6350 gpm to 6450 gpm) on E22-R603 (P601).	P (H.17.7)	***PASS / FAIL** ***STD:*** *Alternately rotates the control switch for 2CSH*MOV110 AND 2CSH*MOV112, TEST RETURN TO CONDENSATE TK, in equal increments on 2CEC*PNL601 in the clockwise direction to the OPEN position and then releases.* ***Failure =*** *HPCS pump flow rising as read on HPCS System Flow meter E22-R603 not achieved.*
Evaluator Note:	Operator may choose to close 2CSH*MOV110 AND 2CSH*MOV112 prior to or after tripping the HPCS pump. If this is performed, it will not result in a JPM failure.		
7.	Recognizes that annunciator 601730, "HPCS Pump 1 Motor Overcurrent" is in alarm and that 2CSH*P1 current ammeter indicates upscale with 2CSH*P1 still running. References ARP 601730 and determines that the automatic response did not occur. Determines per operator action 3 that 2CSH*P1 control switch is required to be placed in P-T-L and places 2CSH*P1 control switch in P-T-L.	P	***PASS / FAIL** ***STD:*** *Rotates the control switch for 2CSH*P1, HPCS PUMP 1 on 2CEC*PNL601 counter clockwise to the Pull-To-Lock position and observes that both the green and red indicating lights are OFF and that pumps amps indicate zero.* ***Failure =*** *2CSH*P1 RED light OFF and GREEN light OFF not achieved.*
8.	Reports to the US that 2CSH*P1 experienced a motor overcurrent condition with a failure of 2CSH*P1 to automatically trip and that HPCS has been secured by placing the control switch in P-T-L.	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Evaluator Note:	Once 2CSH*P1, HPCS PUMP 1 control switch has been placed in Pull-To-Lock. Provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*HPCS pump has been secured by placing the pump control switch in P-T-L.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The plant operating at rated power. • The plant process computer is out of service for corrective maintenance. • Minor maintenance was conducted on the HPCS system. • HPCS has been declared inoperable. • The maintenance activity has been completed. • N2-OP-33, section E.1 Fill and Vent has been completed satisfactorily. • Breaker 2EHS*MCC201-6B, 2CSH*MOV110-COND STG TK TEST BP VALVE is ON and the ALARM CIRCUIT control switch is in ENABLE.

INITIATING CUE	**(Operator Name)**, *perform a HPCS Pump Run Following Maintenance in accordance with N2-OP-33, section H.17.0. The procedure is in progress up to step H.17.3.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant operating at rated power. • The plant process computer is out of service for corrective maintenance. • Minor maintenance was conducted on the HPCS system. • HPCS has been declared inoperable. • The maintenance activity has been completed. • N2-OP-33, section E.1 Fill and Vent has been completed satisfactorily. • Breaker 2EHS*MCC201-6B, 2CSH*MOV110-COND STG TK TEST BP VALVE is ON and the ALARM CIRCUIT control switch is in ENABLE. **Evaluator:** *Ask applicant if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Applicant Name)**, *perform a HPCS Pump Run Following Maintenance in accordance with N2-OP-33, section H.17.0. The procedure is in progress up to step H.17.3.*

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-33

HIGH PRESSURE CORE SPRAY SYSTEM

Revision 01900

Safety Related

CONTINUOUS USE

This procedure requires a 10 CFR 50.59

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
019	00	Minor Revision to Incorporate:

PCR-20-03156

Description: Added an enhancement to P&L D.2.4 to provide information in respect to minimizing time CSH pump is run on min flow. Also, updated embedded cautions for updated P&L. The P&L is updated to read “Minimize the amount of time the CSH pump is run at minimum flow condition to prevent pump damage. A total accumulated time that the CSH pump is run on minimum flow should not exceed 60 hours between overhauls. Reference Calculation Disposition No. A10.1 G 050 01A.”

Reason: AR 04324166-62 directed a review to eliminate vague guidance to perform actions to 'maximize', 'minimize', or 'limit' with no specifics on what it means.

- Section A.1, added a new reference.
- Step D.2.4, added additional information on running on minimum flow.
- Step H.3.3 proceeding caution, aligned to be identical with Step D.2.4.
- Step H.4.2 proceeding caution, aligned to be identical with Step D.2.4.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	5
	A.1. Developmental References	5
	A.2. Performance References	6
	A.3. Commitments	6
B.	SYSTEM DESCRIPTION	7
C.	OPERATING REQUIREMENTS	8
D.	PRECAUTIONS AND LIMITATIONS	9
E.	STARTUP	12
	E.1. Fill and Vent [SOP] [C3]	12
	E.2. Startup to Standby [SOP]	22
F.	NORMAL OPERATION	23
	F.1. Standby Condition Status Checks [EOP]	23
G.	SHUTDOWN	24
	G.1. Shutdown To Standby From Injection [SOP] [EOP] [C1]	24
	G.2. Shutdown to Standby From Test Lineup	25
	G.3. Shutdown To Inoperable	27
H.	OFF NORMAL PROCEDURES	28
	H.1. Automatic Initiation [SOP]	28
	H.2. Suppression Pool Fill By Gravity Drain [EOP]	29
	H.3. Suppression Pool Fill Utilizing CSH Pump [EOP]	30
	H.4. Securing Injection Flow Following Initiation	32
	H.5. Placing CSH in a Test Lineup	33
	H.6. Manual Transfer CSH Pump Suction from CST To Suppression Pool	36
	H.7. Manual Transfer of CSH Pump Suction from Suppression Pool To CST [EOP]	37
	H.8. Manual Initiation Using Initiation Pushbutton [EOP]	38
	H.9. Manual Injection to RPV [EOP]	39
	H.10. CSH Operation with WTR LEG PMP 2 Out of Service	40
	H.11. Defeating Injection Valve (2CSH*MOV107) High Vessel Level Interlock [EOP] [SOP]	41
	H.12. Drain CSH Discharge Header to Suppression Pool	42
	H.13. Placing CSH in Standby Following High Drywell Pressure Initiation	45
	H.14. Large Volume Flush of HPCS Injection Line in Mode 4 or 5	46
	H.15. HPCS Injection to Fill Reactor Refueling Cavity in Mode 5	52
	H.16. HPCS Fill and Vent using MWS Supply with System Keep Fill Pump Inoperable	64
	H.17. HPCS Pump Run Following Maintenance	67

TABLE OF CONTENTS (Continued)

SECTION TITLE PAGE

H.18. Maintaining HPCS Available using MWS Supply with 2CSH*P2 Inoperable..... 69

Attachment 1, Making 2CSH*MOV107, PMP 1 INJECTION VLV, Throttleable 71

Attachment 2, 2CEC*PNL601 (H.13.0) 73

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

- A10.1-G-050-01A, HPCS Pump Min. Flow Requirements
- T.S. 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation
- T.S. 3.5.1, ECCS Operating
- T.S. 3.5.2, Reactor Pressure Vessel (RPV) Water Inventory Control
- T.S. 3.6.1.3, Primary Containment Isolation Valves (PCIVs)
- T.S. 3.8.1, AC Sources - Operating
- T.S. 3.8.2, AC Sources - Shutdown
- USAR Section 6.3, Emergency Core Cooling Systems
- PID 33A High Pressure Core Spray
- PID 33B High Pressure Core Spray
- ESK 6CSH01, HPCS Water Leg Pump
- ESK 6CSH02, High Pressure Core Spray MOVs
- ESK 6CSH03, High Pressure Core Spray MOVs
- ESK 6CSH04, High Pressure Core Spray MOVs
- ESK 6CSH05, High Pressure Core Spray MOVs
- ESK 7CSH02, ERF Computer Input Isolation Circuits
- ESK 11IHS18, ERF Computer Output Isolation Circuits
- GE 807E172TY, HPCS Elementary Diagram and Device Test, Sheets 1 7
- GE 807E183TY, HPCS Power Supply System
- GE 731E950AF, HPCS Functional Control Diagrams
- Borg Warner Byron Jackson Pump Division, Manual No. 8020VMT/1F7820, GEK 64171, SWEC PO NMP2 P800A, Access No. 4300008888.
- Gould Pumps Technical Manual, SWEC P.O. NMP2 P222X 5, Access No. 430000719.
- Anchor/Darling Valve Co., Motor Operated Valve Instruction Manual VPF 3238 724 3, SWEC P.O. NMP2 800A, Access No. 430000178.
- GEK 83335 HPCS Operations and Maintenance Instructions
- FSAR Questions and Responses F430.35 ECCS QA Cat I MOV Duty
- SSM N064-022

A.1 (Continued)

- Safety Evaluation 89-029, 90-145
- NMPC File Code 73598 Dated March 19, 1991
- IR 02083078, RHS Suppression Pool Cooling Mode Inops LPCI mode
- IR 02094174, HPCS System Trip Unit Annunciator 601727 Alarmed After 2CSH*P1 Was Started During N2-OSP-CSH-Q@002
- N2-OSP-CSH-CS001, HPCS Cold Shutdown Valves Operability Test
- N2-OSP-CSH-M001, HPCS Discharge Piping Fill and Valve Line up Verification
- N2-OSP-CSH-Q001, High Pressure Core Spray Valve Operability Test
- N2-OSP-CSH-Q@001, HPCS Pump and Valve Operability and System Integrity Test
- N2-OSP-CSH-R001, High Pressure Core Spray System Functional and Response Time Test
- N2-OSP-CSH-R002, High Pressure Core Spray System Valve Position Indication Operability Test
- N2-OSP-CSH-R301, Logic System Functional Test of HPCS Injection Valve Closure on Reactor Vessel High Level 8 Trip
- N2-OSP-CSH-R@001, HPCS Pressure Isolation Valve Leakage Test

A.2. **Performance References**

- N2-VLU-33-LINEUPS
- N2-OP-4, Condensate Storage and Transfer

A.3. **Commitments**

Sequence Number	Commitment Number	Description
1	SOER 86-1	Verification of trips and valve positions after tests, maintenance and operations.
2	IR 02088103	Trip of RCIC Turbine
3	NRC-GL-2008-01 NCTS 504646	NRC Generic Letter 2008-01. Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems.

B. SYSTEM DESCRIPTION

The High Pressure Core Spray System (CSH) is designed to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. The CSH system is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than one-inch nominal diameter. CSH will provide spray cooling heat transfer during larger breaks in which the core is calculated to become uncovered. CSH will also supply makeup water to the reactor vessel in the event of reactor isolation and failure of the Reactor Core Isolation Cooling (ICS) system. The CSH system can also be used to add water to the Suppression Pool. The system is normally powered from off-site power, or from a dedicated Diesel Generator (EDG2), if off-site power is not available.

The CSH system consists of a single, motor-driven CSH pump (CSH*P1), a smaller motor-driven CSH System Pressure Pump (CSH*P2), and associated piping, valves, controls, and instrumentation to support system operation.

Normal suction for the system is from the "B" Condensate Storage Tank (CST), using the Suppression Pool (SP) as an alternate source. The CST suction piping for other systems is arranged so that a volume of 135,000 gallons is dedicated to the CSH system. The suction flowpath will automatically transfer from the CST to the SP upon either a low CST level or a high SP level signal.

Under normal plant conditions, the CSH system is filled and vented, in the "Standby" mode with the System Pressure Pump running to maintain pressure at approximately 75 psig.

To test the operability of the CSH system during plant operation, two test loops are provided to verify proper discharge pressures and flow. One test loop permits flow from/to the Condensate Storage Tank. The second test loop permits flow from/to the Suppression Pool. In the event the CSH pump is operated with the Injection Valve (CSH*MOV107) and all test valves
(*MOV110 -111, -112) close, the Minimum Flow Valve (*MOV105) opens to permit sufficient flow (500 gpm) to the Suppression pool to prevent pump overheating.

The CSH system initiates automatically on either high drywell pressure or low-low reactor vessel level, starting the CSH pump (*P1) and opening its injection valve (*MOV107). The CSH pump discharges into the vessel through a sparger mounted within the core shroud to direct spray over the fuel bundles. The system can also be manually initiated. If the system is in any mode other than standby, all valves will align to the injection mode. If the system is taking suction from the SP, the SP suction valve (*MOV118) will remain open.

When reactor vessel level is restored to the high level alarm point, the Injection Valve closes, and the Minimum Flow Valve opens, diverting flow to the SP. The CSH automatic initiation signal can be reset when the reactor vessel level is restored above 108.8". A high drywell pressure signal will not prevent resetting the initiation signal.

C. OPERATING REQUIREMENTS

C.1. **The following systems must be in operations to support HPCS system:**

- HPCS Diesel Generator N2-OP-100B
- Standby and Emergency AC Distribution N2-OP-72
- HPCS 125 VDC Distribution N2-OP-74B

D. PRECAUTIONS AND LIMITATIONS

D.1. System Operation

D.1.1. If WTR LEG PMP 2 is out of service, CSH will be inoperable but may remain available provided that the requirements of Subsection H.10, CSH Operation with WTR LEG PMP 2 Out of Service, are met.

D.1.2. Maintain CSH system in automatic operation, unless one of the following is confirmed, by at least two independent indications:

- Misoperation in AUTOMATIC mode is confirmed, or
- Adequate core cooling is assured.

D.1.3. Misoperation includes: inappropriate initiation of CSH system or continued operation of CSH system beyond automatic trip setpoints.

NOTE

When CSH system is in MANUAL, it will not restart automatically.

D.1.4. If CSH system is placed in MANUAL mode:

- Make frequent checks of the initiating or controlling parameter.
- When MANUAL mode is no longer required, restore the CSH system to AUTOMATIC or STANDBY Mode. (SER 90-145)

D.1.5. Only the Condensate Storage and Transfer System may be used to fill or flush the CSH System.

D.1.6. Condensate Storage and Transfer System shall never be used to keep CSH system charged due to the potential for over pressurizing the low pressure CNS piping with CSH pump discharge pressure.

D.1.7. If CSH system is discharging into the vessel under any conditions other than an actual loss-of-coolant accident, CSH suction should be aligned to CST.

D.1.8. If CSH system is operating in full flow test with suction and return aligned to CSTs, CST A level will rise. This may result in an overflow of CST A. ith 2CSH*P1 flow at 6350 gpm, CST levels will stabilize when a differential level of approximately 9 feet is established between CST A and CST B.

D.1.9. Due to concerns that CSH will not inject to the vessel within the analyzed time frame with the full flow test valves open. CSH will need to be declared inoperable with the CSH pump running and test return valves open.

D.1.10. Gas accumulation monitoring is required following maintenance or operational activities that have the potential to drain the system (or portions of the system) or that may cause gas accumulation. This will require Ultrasonic Testing with subsequent fill and vent (if required) prior to restoring system to operable. These actions will ensure piping is full of water

D.2. **CSH Pump and Motor Limitations**

D.2.1. Maintain CSH flow less than 7175 gpm in all operating modes.

D.2.2. CSH pump motor start limitations:

- Two starts in succession from ambient temperature after which a 60-minute wait is required prior to subsequent start attempts.
- One start from rated temperature (established after 30 minutes run time), after which a 60-minute wait is required prior to subsequent start attempts.

D.2.3. Maintain CSH Pump Motor stable full load running current less than 378 amps.

D.2.4. Minimize the amount of time the CSH pump is run at minimum flow condition to prevent pump damage. The NMPNS ASME XI Program implements vibration monitoring and performance monitoring, which will detect any significant effect on pump parameters prior to any significant degradation. No additional administrative actions are required for tracking time that the CSH pump is run at minimum flow. (See Calculation No. A10.1-G-050-01A.)

D.2.5. As an operator good practice, the High Pressure Core Spray pump supply breaker should be checked properly recharged following normal opening operation. Refer to N2-ELU-01 Subsection 3.4.

D.3. **CSH Valve Limitations**

D.3.1. The duty cycle of ECCS MOVs is five cycles per hour. If injection is required and MOV duty cycle limitation is exceeded, ICS or Low Pressure ECCS Systems should be used.

D.3.2. Do not open 2CSH*MOV107, Pump 1 Injection Vlv., while testing 2CSH*V108, Injection Check Valve.

D.3.3. 2CSH*MOV107, PMP 1 INJECTION VLV should not be manually opened with the CSH test return valves to the CST or suppression pool open. This would cause pump runout.

D.3.4. Do not use excessive force to achieve leak tight closure of valve 2CSH*HCV133, Test Bypass to Suppression Pool.

D.3.5. If the reactor is shutdown and depressurized, CSH Hi Water Level Seal in interlock will not reset. This is due to the wide range level instruments reading upscale because their calibration conditions are not met.

D.4. **CSH is a single train safety system:**

D.4.1. Unplanned inoperability of the CSH system must be reported to the NRC (10CFR50.72.b.3.v and 50.73.a.2.v) even though the plant Technical Specifications may allow such a condition to exist.

D.4.2. Inoperability of the CSH system where it is removed from service as part of a planned evolution (such as PM's, ST's or corrective maintenance) in accordance with an approved procedure, and in accordance with the plants Technical Specifications, does not require reporting to the NRC. However, if a condition or event is discovered which would have otherwise rendered CSH inoperable, then D.4.1 applies.

D.5. Prior to venting or draining evolutions, notify Radiation Protection.

D.6. If the CSH system is operated with suction from the suppression pool, it must be shut down prior to returning the suction to the CST's. Swapping the suction with the system in operation could cause a hydraulic transient that is outside the system design basis. **[C2]**

D.7. The CSH System is a designated closed system, which means that the system piping is part of the primary containment. The closed system boundary includes the main process lines from the suppression pool on the suction side of associated pumps through the containment penetration on the discharge side, out to and including the first normally closed branch boundary valve (i.e., drain/vent valves, interface boundary valves, relief valves, etc.). The closed system is credited as one of the two boundaries required for containment isolation under Appendix J and TS 3.6.1.3. Excessive leakage above the minimum suppression pool level (197’-8”) could impact the closed system primary containment boundary function. System leaks identified above elevation 197’-8” greater than 10cc/min (200 dpm) require entry into TS 3.6.1.3 and securing the associated primary containment penetration. Leaks identified less than 10cc/min need to be documented immediately on a work document and corrected on a priority basis (i.e. Pri-3 Necessary or greater).

Initials

H.17. HPCS Pump Run Following Maintenance

NOTE

- The purpose of this section is to provide a procedure to run the HPCS pump to perform dynamic flushing of the HPCS suction piping following maintenance which may have introduced air into the system.
- The HCPS system will be inoperable while the system is run in test lineup.

H.17.1. **VERIFY** section E.1 Fill **AND** vent has been completed following maintenance. OPE

H.17.2. At 2EHS*MCC201-6B, 2CSH*MOV110 COND STG TK TEST BP valve, **VERIFY** the following:

a. Breaker is ON. OPE

b. ALARM CIRCUIT control switch is in ENABLE. OPE

H.17.3. **VERIFY** 2CSH*MOV110, TEST RETURN TO CONDENSATE TK THROTTLE valve indicates 0%, CLOSED. _____

NOTE

Starting 2CSH*P1 with the suction from the CST can result in Gross Failure alarms on trip units E22-N652, HPCS PMP SUCT LO, and E22-N653, HPCS PMP SUCT HI.

H.17.4. **START** 2CSH*P1, HPCS PUMP 1. _____

H.17.5. **VERIFY** 2CSH*MOV107, PMP 1 INJECTION VLV, is closed. _____

H.17.6. **OBSERVE** 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, opens. _____

CAUTION

- Throttling HPCS return flow to CST with only one throttle valve can cause extensive valve damage.
- With HPCS operating in full flow test with suction and return to the CSTs, “A” CST will fill up and overflow if level is not monitored.

H.17.7. **THROTTLE OPEN** 2CSH*MOV110 **AND** 2CSH*MOV112, TEST RETURN TO CONDENSATE TK, in equal increments to establish 6350 gpm (6350 gpm to 6450 gpm) on E22-R603 (P601). _____

H.17.8. **OPERATE** 2CSH*P1 with flow at 6350 gpm (6350 gpm to 6450 gpm) for approximately 5 minutes. _____

Initials

H.17.9. **CLOSE** 2CSH*MOV110 **AND** 2CSH*MOV112, TEST RETURN TO CONDENSATE TK. _____

H.17.10. **OBSERVE** 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, opens. _____

H.17.11. **OPEN** 2CSH*MOV118, PMP SUCT FROM SUPPRESSION POOL. _____

H.17.12. **OBSERVE** 2CSH*MOV101, PMP SUCT FROM CNDS TK close. _____

H.17.13. **THROTTLE OPEN** 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL THROTTLE to establish a minimum of 6350 gpm (6350 gpm to 6450 gpm). _____

H.17.14. **OPERATE** 2CSH*P1 with flow at 6350 gpm (6350 gpm to 6450 gpm) for at least 5 minutes. _____

H.17.15. **THROTTLE CLOSE** 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL THROTTLE. _____

H.17.16. **OBSERVE** 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, opens. _____

H.17.17. **PLACE** 2CSH*P1, HPCS PMP 1 control switch to STOP. _____

H.17.18. **OBSERVE** 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, close. _____

H.17.19. **CLOSE** 2CSH*MOV118 PMP SUCT FROM SUPPRESSION POOL. _____

H.17.20. **OPEN** 2CSH*MOV101, PMP SUCT FROM CNDS TK. _____

H.17.21. At 2EHS*MCC201-6B, 2CSH*MOV110 COND STG TK TEST BP valve, **VERIFY** the following:

a. Breaker is OFF. _____

b. ALARM CIRCUIT control switch is in DISABLE. _____

Initials

H.18. **Maintaining HPCS Available using MWS Supply with 2CSH*P2 Inoperable**

NOTE

- 2MWS-V188 is located at Reactor Building elevation 183' north, just east of the north Aux bay.
- 2CSH*V61 is located in CSH Pump Room El 181'approx 12' from 2CSH*P1.
- Per OP-NM-108-101, a blue taped hose may be used to provide fill and vent water to a potentially contaminated system provided a check valve is installed to prevent reverse flow. Install the check valve at the end of the hose that is connected to the CSH system.
- A tee with a valve should be installed in the hose line to fill and vent the MWS hose prior to using it to pressurize CSH and to depressurize the hose prior to disconnecting from the CSH system.
- This section may be used to make CSH available. It may **NOT** be used to restore CSH to an OPERABLE status.
- CSH suction relief is set at 100 psig. MWS supply pressure must be controlled to prevent lifting suction relief.

H.18.1. **OBTAIN** Breech permit for HPCS Door R175-4. _____

H.18.2. **PREPARE** hose, capable of 150 psig **AND** containing a check valve, that will be connected from 2MWS-V188 to 2CSH*V61. _____

H.18.3. **UNCAP AND CONNECT** hose to2MWS-V188, MWS TO RHR A HEAT EXCHANGER HDR DRAIN. _____

H.18.4. **UNCAP AND CONNECT** hose to 2CSH*V61, SUPP POOL SUCT PIPE VENT. _____

H.18.5. **OPEN** 2MWS-V188. _____

H.18.6. **VERIFY NO OR** minimal leakage, **AND VENT** fill hose. _____

H.18.7. **VERIFY** 2CSH*P1, CONTROL SWITCH is in PULL-TO-LOCK. _____

NOTE

2CSH*V61 is located in the CSH pump room.

H.18.8. Station Operator in low dose area near 2CSH*V61. _____

Training ID: **Nov 2023 NRC JPM B** Revision: **0.0**

Title: **Override the Control Room Envelope ACU Cross-Divisional Operating Interlock**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A		Mike Carno	7/12/23
Facility Reviewer:	Signature on File		John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-53A, Rev. 02000, Control Building Ventilation System
2. NUREG 1123 K/A 290003 A4.02 (3.3)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the operator's ability to override the Control Room Envelope ACU Cross-Divisional Operating Interlock in accordance with N2-OP-53A section H.15.0.

   b. This JPM is **not** considered time critical because a note in the procedure states "…actuation of the override switches is required to be completed within 13 minutes after a fire that has disabled the previously running Control Room Envelope ACUs…". In this JPM, the ACUs are still running.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-288003-01051 perform override of control room envelope ACU cross-divisional operation interlock.

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
Safety Function:	9		Radioactivity Release	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	290003 Control Room Ventilation A4.02 - Ability to manually operate and/or monitor in the control room: Fans			
K/A Importance Rating:	RO	3.3	SRO	-

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) **FF16:012-001,** 332NW West Cable Chase CB 261 AA, Final = Alarm **Inserted**

2) Remotes

a) None

3) Overrides

a) None

4) Annunciators

a) None

5) Event Triggers

Event #	Event Action	Command
None	N/A	N/A

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-OP-53A, section H.15.0 with no steps placekept as complete.

8) Miscellaneous

a) IC-205 (For ILT 22-2, paired with JPM D)
-OR-

b) IC setup:

(1) Ensure that the simulator reflects the conditions stated in the initial conditions regarding Unit Coolers in service.

6. Strategy Code

a. None

7. Tools and Equipment

a. None

8. Commitments

a. None

9. Prerequisites

   a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

   a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • A fire has developed in fire zone 332NW - Cable Chase West, Control Bldg, 261' (Fire Area 16) • 2HVC*ACU1B, CONTROL ROOM AC FAN is available, but not running. • 2HVC*ACU2B, RELAY ROOM AC FAN is available and is running. • 2HVC*ACU1A, CONTROL ROOM AC FAN is available and is running • 2HVC*ACU2A, RELAY ROOM AC FAN is available, but not running. • The SM has directed that it is necessary to Override the Control Room Envelope ACU Cross-Divisional Operating Interlock. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *override the Control Room Envelope ACU Cross-Divisional Operating Interlock in accordance with N2-OP-53A section H.15.0.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	• The control logic for the Control Room Envelope air handling units 2HVC*ACU1A/B and *ACU2A/B includes a start permissive which precludes simultaneous operation the units and is intended to auto start the unit in the opposite division in the event of a trip of the running unit. A fire affecting one division has the potential to override this interlock such that the opposite division ACU may become disabled. This subsection will utilize override switches to ensure operability of the division without the fire. • Actuation of the override switches is required to be completed within 13 minutes after a fire that has disabled the previously running Control Room Envelope ACUs to ensure Control Room Envelope temperature remains <90°F. • This section is only used in response to a confirmed fire in Fire Areas 16, 17, 18, or 19. • The actions in this section are performed at 2CEC*PNL870(871) unless otherwise specified. • The Fire Zones listed in Step H.15.1 affect Division I components.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Evaluator Note:	In the above procedure note, a time limit of 13 minutes is discussed. For this JPM, the fire has not yet disabled the previously running Control Room Envelope ACUs, so the JPM is NOT time critical.		
2.	IF fire is in any of the following Fire Zones: • 321NW - Cable Chase West, Control Bldg, 237' (Fire Area 16) • 332NW - Cable Chase West, Control Bldg, 261' (Fire Area 16) • 352NW - Cable Chase West, Control Bldg, 288'6" (Fire Area 16) • 371NW - Cable Chase West, Control Bldg, 306' (Fire Area 16) • 333XL - Div I Switchgear Room, Control Bldg, 261' (Fire Area 17) THEN performs the following:	P (H.15.1)	SAT / UNSAT ***STD:*** *Recalls from the initial conditions that the fire is located in fire zone 332NW and placekeeps step.*
3.	Checks the following ACUs are available OR are in service: • 2HVC*ACU1B, CONTROL ROOM AC FAN.	P (H.15.1.1 first bullet)	SAT / UNSAT ***STD:*** *Observes GREEN light ON and RED light OFF for 2HVC*ACU1B, CONTROL ROOM AC FAN on 2CEC*PNL871 or recognizes available from JPM initial conditions.*
4.	• 2HVC*ACU2B, RELAY ROOM AC FAN	P (H.15.1.1 second bullet)	SAT / UNSAT ***STD:*** *Observes GREEN light OFF and RED light ON for 2HVC*ACU2B, RELAY ROOM AC FAN on 2CEC*PNL871 or recognizes available from JPM initial conditions.*
Procedure Note:	The following steps will cause an auto start of 2HVC*ACU1B and *ACU2B, if not already running.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
5.	Verifies control switch is in PULL-TO-LOCK for the following: • 2HVC*ACU1A, CONTROL ROOM AC FAN	P (H.15.1.2 first bullet)	SAT / UNSAT ***STD:*** *Rotates 2HVC*ACU1A, CONTROL ROOM AC FAN control switch counter clockwise and pulls. Verifies RED light OFF, GREEN light ON and that control switch handle is horizontal.*
6.	• 2HVC*ACU2A, RELAY ROOM AC FAN	P (H.15.1.2 second bullet)	SAT / UNSAT ***STD:*** *Rotates 2HVC*ACU2A, RELAY ROOM AC FAN control switch counter-clockwise and pulls. Verifies RED light OFF, GREEN light ON and that control switch handle is horizontal.*
7.	Notifies SM to declare 2HVC*ACU1A AND 2HVC*ACU2A inoperable AND to take actions per TS 3.7.2 AND 3.7.3 **Cue:** *If notified, report that 2HVC*ACU1A AND 2HVC*ACU2A have been declared inoperable AND that actions are being taken per TS 3.7.2 AND 3.7.3*	P (H.15.1.3)	SAT / UNSAT ***STD:*** *Proper communications used.*
8.	Obtains SM permission to override HVC ACU cross divisional operating interlock for 2HVC*ACU1A AND 2HVC*ACU2A	P (H.15.1.4)	SAT / UNSAT ***STD:*** *Recalls from the initial conditions that the SM has directed the task to be performed and place keeps step.*

Procedure Note:	If 2HVC*ACU1B is not running, the following step should cause an automatic start of 2HVC*ACU1B.
Evaluator Note:	• The next step requires an independent verification. If the operator requests an independent verification, repeat back the request for an independent verification only.

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
9.	At 2CEC*PNL871, places keylock switch APPENDIX R CR ACU1B OVERRIDE SW, to OVERRIDE **Cue:** *If asked, Independent Verification of this step will be performed later by an additional operator.*	P (H.15.1.5)	***PASS / FAIL** ***STD:*** *Obtains PA2235 key, inserts the key into the APPENDIX R CR ACU1B OVERRIDE SW on 2CEC*PNL871 and rotates the key clockwise to the OVERRIDE position.* ***Failure =*** *APPENDIX R CR ACU1B OVERRIDE SW in OVERRIDE position not achieved.*
10.	Verifies 2HVC*ACU1B running.	P (H.15.1.6)	SAT / UNSAT ***STD:*** *Observes RED light ON and GREEN light OFF for 2HVC*ACU1B on 2CEC*PNL871.*
11.	WHILE continuing in this subsection, waits approximately 2 minutes, THEN confirms proper operation of 2HVC*ACU1B **Cue:** *2 minutes has elapsed.*	P (H.15.1.7)	SAT / UNSAT ***STD:*** *Observes RED light ON and GREEN light OFF for 2HVC*ACU1B on 2CEC*PNL871. Observes 2CEC*PNL871 for any indications of a malfunction of 2HVC*ACU1B.*
12.	Places 2HVC*ACU1B control switch, CONTROL ROOM AC FAN, in Normal-After-START.	P (H.15.1.8)	SAT / UNSAT ***STD:*** *Rotates 2HVC*ACU1B control switch on 2CEC*PNL871 clockwise to the red flagged position and verifies RED light ON and GREEN light OFF.*

Procedure Note:	If 2HVC*ACU2B is not running, the following step should cause an automatic start of 2HVC*ACU2B.
Evaluator Note:	The next step requires an independent verification. If the operator requests an independent verification, repeat back the request for an independent verification only.

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
13.	At 2CEC*PNL871, places keylock switch APPENDIX R RR ACU2B OVERRIDE SW, to OVERRIDE **Cue:** *If asked, Independent Verification of this step will be performed later by an additional operator.*	P (H.15.1.9)	***PASS / FAIL** ***STD:*** *Obtains PA2235 key, inserts the key into the APPENDIX R RR ACU2B OVERRIDE SW on 2CEC*PNL871 and rotates the key clockwise to the OVERRIDE position.* ***Failure =*** *APPENDIX R RR ACU2B OVERRIDE SW in OVERRIDE position not achieved.*
Evaluator Note:	The initial conditions for this JPM will have 2HVC*ACU2B red flagged and running, so the next 3 steps will only require verification.		
14.	Verifies 2HVC*ACU2B running	P (H.15.1.10)	SAT / UNSAT ***STD:*** *Observes red light on and GREEN light OFF for 2HVC*ACU2B on 2CEC*PNL871.*
15.	WHILE continuing in this subsection, waits approximately 2 minutes, THEN confirms proper operation of 2HVC*ACU2B	P (H.15.1.11)	SAT / UNSAT ***STD:*** *Observes RED light ON and GREEN light OFF for 2HVC*ACU2B on 2CEC*PNL871. Observes 2CEC*PNL871 for any indications of a malfunction of 2HVC*ACU2B.*
16.	Places 2HVC*ACU2B control switch, RELAY ROOM AC FAN, in Normal-After-START.	P (H.15.1.12)	SAT / UNSAT ***STD:*** *Verifies 2HVC*ACU2B control switch on 2CEC*PNL871 is red flagged verifies red light on and green light off.*
Evaluator Note:	**Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*Both APPENDIX R CR ACU1B OVERRIDE SW and APPENDIX R RR ACU2B OVERRIDE SW have been placed in the OVERRIDE position.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • A fire has developed in fire zone 332NW - Cable Chase West, Control Bldg, 261' (Fire Area 16) • 2HVC*ACU1B, CONTROL ROOM AC FAN is available, but not running. • 2HVC*ACU2B, RELAY ROOM AC FAN is available and is running. • 2HVC*ACU1A, CONTROL ROOM AC FAN is available and is running • 2HVC*ACU2A, RELAY ROOM AC FAN is available, but not running. • The SM has directed that it is necessary to Override the Control Room Envelope ACU Cross-Divisional Operating Interlock.

INITIATING CUE	**(Operator Name)**, *override the Control Room Envelope ACU Cross-Divisional Operating Interlock in accordance with N2-OP-53A section H.15.0.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • A fire has developed in fire zone 332NW - Cable Chase West, Control Bldg, 261' (Fire Area 16) • 2HVC*ACU1B, CONTROL ROOM AC FAN is available, but not running. • 2HVC*ACU2B, RELAY ROOM AC FAN is available and is running. • 2HVC*ACU1A, CONTROL ROOM AC FAN is available and is running • 2HVC*ACU2A, RELAY ROOM AC FAN is available, but not running. • The SM has directed that it is necessary to Override the Control Room Envelope ACU Cross-Divisional Operating Interlock.

INITIATING CUE	**(Operator Name)**, *override the Control Room Envelope ACU Cross-Divisional Operating Interlock in accordance with N2-OP-53A section H.15.0.*



Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-53A

CONTROL BUILDING VENTILATION SYSTEM

Revision 02200

Safety Related

CONTINUOUS USE

10 CFR 50.59 REVIEW REQUIRED

Approval Authority: Director-Operations

Print In COLOR

SUMMARY OF ALTERATIONS

PCR-21-02423

Description: Added a new section to allow operations to shift the in-service HVK chiller oil filters.

Reason: Added a new section: H.23, Shifting Chiller Oil Filters. It to allows Operations to shift the in-service HVK chiller oil filters. Shifting oil filters is an operations activity and is required periodically based on rounds trending of oil pressure.

- Added Section H.23, Shifting Chiller Oil Filters.

PCR-22-00472

Description: Updated steps and added steps to ensure the chillers are manually loaded when started.

Reason: Updated steps and added steps to ensure the chillers are manually loaded when started. The change is driven by lessons learned from IR 04462597, 2HVK*CHL1A Control Building Water Chiller Low Oil Pressure. A learning from the IR was that manual starting the chillers is needed to ensure the chillers do not trip on low oil pressure due to equipment damage when system disturbances occur after a fast start of the chillers.

- Added P&L D.30.
- Step E.2.7.k, added to establish control room communication.
- Added a bullet to the NOTE before step E.2.7.l to match the content on the ‘B’ train in the note before step E.2.8.l.
- Steps E.2.7.l - E.2.7.o, added to manually control the start of the chiller.
- Step E.2.7.q.2) was updated with “OR approximately 50 °F IF operating with High Service Water temperature conditions.” This aligns the step with the same directions found in step H.8.2.b.2).
- Added a CAUTION before step E.2.8.j.
- Step E.2.8.k, added to establish control room communication.
- Steps E.2.8.l - E.2.8.o, added to manually control the start of the chiller.
- Made E.2.8.q.1) and E.2.8.q.2) substeps per Operator feedback.
- Deleted the NOTE before step F.3.8. It stated, “Aux Oil Pressure is read on 2HVK*PI1000A(B) AND is only available for approximately 20 seconds from the time the Chiller switch is placed in PULL-TO-LOCK until the time it shuts down.” No revision bars used.
- Step F.3.8, moved the substeps to F.3.13.
- Moved last step of F.3.13 to F.3.18 since it is administrative and does not need to be done until the end of operational action steps. Performed the same for steps E.2.18, F.2.11, F.3.18, G.1.17, and H.22.27.

Procedure Writer Changes

- Step E.2.7.i and E.2.7.j, moved the D.18 bloat to a preceding CAUTION.
- Deleted P&L 29, “Use of N/A may only be used as directed in this procedure, except as allowed by HU-AA-104-101, Procedure Use and Adherence.” It is an outdated legacy precaution requiring removal per IR 04172818, PIIM 2018-0380 Operations Procedure Excellence. Steps automatically renumbered. No revision bar used.
- Updated CAUTION statements to align with P&L D.30. The P&L was written to the procedure writing standards where the hazard is listed first and then explanatory information is provided. Also removed the condition statement to require a manual start of the chiller.
- Step F.3.15, removed bloat reference to P&L 18 and moved it to a proceeding CAUTION. Repeated for all identical steps in the procedure.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	5
B.	SYSTEM DESCRIPTION	8
C.	OPERATING REQUIREMENTS	12
D.	PRECAUTIONS AND LIMITATIONS	13
E.	STARTUP	16
E.1.	Fill and Vent of Control Building Chilled Water System	16
E.2.	System Startup	23
F.	NORMAL OPERATIONS	36
F.1.	Operational Checks	36
F.2.	Shifting From 2 Chiller to Single Chiller Operation [SOP]	37
F.3.	Shifting from Single Chiller to 2 Chiller Operation	41
F.4.	Swapping Control Room Envelope Air Conditioning Units	48
F.5.	Swapping Computer Room A/C Units	51
G.	SHUTDOWN	52
G.1.	System Shutdown	52
G.2.	Shutdown of the Control Building Special Filter Train	56
H.	OFF-NORMAL PROCEDURES	57
H.1.	Automatic Initiation of Control Building Special Filter Train [EOP] [SOP]	57
H.2.	Control Room Smoke Removal	59
H.3.	Control Room Odor Removal	62
H.4.	Relay Room and Computer Room Smoke Removal	63
H.5.	Lining Up Service Water to Chilled Water Loop A (B)	67
H.6.	Manual Initiation of the Control Building Special Filter Train [EOP] [SOP]	75
H.7.	Fire in Special Filter Charcoal Absorbers [C1]	76
H.8.	Operation of 2HVK*CHL1A(B) with 2SWP*V240A(B) Inop and/or 2SWP*P2A(B) Inop	78
H.9.	Restoring Normal Level in Chilled Water Expansion Tanks	81
H.10.	Overriding Temperature Control Valves Open for ACUs for Corrective Maintenance	84
H.11.	Response to Alarms on 2HVC-ACU4A(B)	86
H.12.	Shutdown of 2HVC*ACU1A and 2HVC*ACU1B	87
H.13.	Shutdown of 2HVC*ACU2A and 2HVC*ACU2B	89
H.14.	Bypassing SWP Temperature Control Valves for 2HVK*CHL1A(B)	91
H.15.	Overriding the Control Room Envelope ACU Cross-Divisional Operating Interlock	93
H.16.	Voluntary Entry Into Tech Spec Condition 3.7.2.B [C14]	98
H.17.	Responding to a Control Room Envelope Door Alarm [C15]	99

TABLE OF CONTENTS

SECTION TITLE PAGE

H.18. Manually Purging and/or Monitoring Non-Condensable Gas at 2HVK*CHL1A/B 101
H.19. Inoperable Control Room Envelope Boundary 102
H.20. Adding Oil to 2HVK*CHL1A(B) to Support Chiller Start-up 103
H.21. Restart 2HVC-ACU4A(B) 104
H.22. Starting and Operating HVK Chiller with 2SWP*TV35A(B) INOPERABLE 105
Attachment 1, Control Room Envelope Doors 112

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

- 3.3.3.2, Remote Shutdown System
- 3.7.2, Control Room Envelope Filtration (CREF) System
- 3.7.3, Control Room Envelope Air Conditioning (AC) System
- 5.5.7, Ventilation Filter Testing Program (VFTP)
- 5.5.13, Control Room Envelope Habitability Program

A.2. Licensee Documentation

Updated Safety Analysis Report

- Section 9.4.1, Control Building Heating, Ventilating, and Air Conditioning System
- Section 9.4.10.1, Control Building Chilled Water Subsystem
- Table 9.4-2, Design Data of Principal Equipment-Control Building and Normal Switchgear Building HVAC System
- Table 9.4-11, Design Data of Principal Equipment-Plant Chilled Water System
- Figure 9.4-4 sheets 1 through 9, Control Building HVAC System-Main Control Room Logic Diagram
- Figure 9.4-5 sheets 1 through 7, Control Building HVAC System-Relay and Computer Rooms Logic Diagram
- Figure 9.4-20 sheets 1 through 5, Control Building Chilled Water Systems Logic Diagram

Technical Requirements Manual (TRM)

- Section 3.3.3.2, Remote Shutdown System

A.3. Technical Information

A.3.1. Electrical Diagrams

- ESK-62HVC01 through 62HVC24, Control Building Air Conditioning
- ESK-6HVK01 through 6HVK04, Control Building Chill Water
- ESK-72HVC01 through 72HVC36, Control Building Air Conditioning
- ESK-7HVK01 through 7HVK04, Control Building Chill Water

A.3.2. Logic Diagrams

- LSK-9-10J, Service Water
- LSK-22-9.1A through 22-9.4G, Control Bldg. Air Conditioning
- LSK-22-12A through 22-12E, Control Bldg. Chill Water

A.3.3. Vendor Manuals

- 2HVC*FLT2A(2B), Spec: NMP2-P243U, Mine Safety Appliance Co. Filter Division, Doc. No. B-553560, NMPC File Sequence No. N20969
- 2HVK*CHL1A(B), Spec: NMP2-P413H, Borg Warner, York Division Y01800, HTE1B1-BACS York Centrifugal Turbopak, Doc. No. SI3326 (1078), NMPC File Sequence No. N20416

A.3.4. Vendor Drawings

- York Elementary 076-07927E (SWEC 10.440-104-001)
- Liebert Electrical Schematic 2E-4560 (S&W File No. 10.450-949-093)

A.4. **Supplemental References**

- SER 06-87, Fire Suppression System Effects on Equipment
- NMP38379, OEA Response to SER 06-87 and IN 83-41
- Licensee Event Report 88-046, An Engineered Safety Feature not Single Failure Proof due to a Design Deficiency - Inadequate Design Review
- Safety Evaluation Report 88-062, Closure of Instrument Air Root Valve to 2HVC*AOD120
- Safety Evaluation Report 88-066, Closure of Instrument Air Root Valves to 2HVC*AOD145 & 148
- Safety Evaluation Report 95-027, Control Room Smoke Removal
- Engineering Specification NMP2-5207M (Application of Protective Coating Material
- Safety Evaluation Report 97-054, Removal of Trip Function of Chiller Low Service Water Flow 2SWP*FSL29A and *FSL29B
- Safety Evaluation Report 97-065, Inlet Service Water Temperature Control for Control Building Chillers
- Safety Evaluation 98-067, Logic Changes for Control Building Ventilation fans
- IR 02020531, Unplanned LCO, 2HVK*CHL1A
- CC-NM-201-1001, Plant Barrier Control Program Implementation
- NEP-CRH-01, Control Room Integrity Program Review
- CC-AA-102, Design Impact and Configuration Change Impact Screening
- IR 02338177, Control Room Habitability Tech Spec Amendment 126
- Safety Evaluation 11-001, Safety Classification Determination for 2HVK*V95 & 2HVK*V327
- IR 04425875, Enhance fire resp proc to address impact on charcoal filters

A.5. **Commitments**

Sequence Number	Commitment Number	Description
1	NCTS 003387	Include procedure for operating the Control Room Outdoor Air Special Filter Trains and manually initiating the deluge system including plant impact during high temperatures or fire.
2	NCTS 003364	Amend procedure to include precaution statements warning personnel of the hazards of carbon dioxide and halon.
3	NCTS 503312	Amend procedure to isolate control air to 2HVC*AOD120, 145, and 148.
4	IR 02068849	Procedural guidance for fires outside the Control Room.
5	LER 90-01	2HVC Special Filter Train Actuation Due to Breaker Cycling
6	IR 02069413	TSI #25 Rev 13 issued without procedure changes
7	NCTS 503911-00	Deleted (Refer to TSA #125)
8	IR 02076986	Run Control Room/Relay Room smoke removal mode every other cycle to verify system will function when needed (PMCR P-NMP-015891).
9	IR 02082686	Trip of 2HVK*CHL1A on Low Oil Pressure
10	S.E. 98-067	Logic Changes for Control Building Ventilation Fans
11	IR 02089610	Discrepancies between N2-OP-53A and USAR Section 9.4.10.
12	IR 02091302	Division 2 Chiller failed to operate following maintenance
13	IR 02091993	Low Pressure in Relay Room
14	NCTS 504437, 504438, 504439, 504440	NRC Required Actions for Implementation of Tech Spec Amendment 97
15	IR 02105666 LER 02-003	Open Door Between the Control Building and the Auxiliary Building Results in Breach of the Control Room Envelope
16	IR 02022633	HVK Relief Valve Leakby

B. SYSTEM DESCRIPTION

Control Building Ventilation supplies tempered recirculated and outdoor air to all areas of the Control Building. The system can be divided into the following subsystems:

- Control Room Ventilation
- Relay Room Ventilation
- Control Building Special Filter Trains
- Control Building Chilled Water
- Standby Switchgear/Battery Room Ventilation

Operating instructions for Control Building Ventilation on elevations 261' and below is covered in N2-OP-53E, Standby Switchgear/Battery Room Ventilation. Operating instructions for Control Building Ventilation on elevations 288' and 306' are covered by this procedure.

The Control Room and Relay Room ACUs serve two functions. One is to maintain a positive 1/8" pressure in the Control Room Envelope under normal and accident conditions. The second function is to provide a suitable environment for normal operations, and to remove the design heat load and maintain a controlled environment following a DBA.

With respect to Technical Specifications, the safety function of the Control Room Envelope Air Conditioning (AC) System consists of ensuring that the Control Room Envelope temperature will not exceed equipment OPERABILITY limits (<90 °F) following Control Room Envelope isolation.

Control Room Ventilation

The Control Room Ventilation subsystem supplies tempered recirculated and outdoor air to all areas located on elevation 306' of the Control Building. Air is recirculated from all spaces except the kitchen and toilet room where the air is exhausted through fans to the atmosphere. The supply capacity exceeds the exhaust capacity to ensure a positive pressure is maintained in the Control Room.

The subsystem is served by two full capacity, redundant air conditioning units (2HVC*ACU1A and 1B), each consisting of a fan, cooling coil, and filters. Normally, flow passes through a glass fiber cartridge type filter before entering the ACUs. In the event removal of odors is required, manual valves in the filter unit allow for placing a charcoal filter on-line.

During normal operation, one Control Room ACU will be operating with the other in standby. The standby ACU will auto start after a time delay on low air-flow in the running ACU provided that the running unit cooler breaker has opened (i.e., loss of power, control switch to pull to lock, etc.).

Additionally, the ACUs have time delays for sequential starting in case of loss of power to both divisions of AC emergency power. This time delay function is to prevent operation of both 100% redundant unit coolers at the same time. Heaters in the duct autostart upon ACU startup and cycle to maintain desired temperature.

B SYSTEM DESCRIPTION (Continued)

Normal cooling to *ACU1A and 1B is provided by the Control Building Chilled Water System. In the event of a failure of both chiller compressors, ACUs may be manually lined up to the Service Water System for backup cooling.

Heaters are provided in the Control Room air recirculation ducts (2HVC*CH11A/B) and Relay Room air recirculation ducts (2HVC*CH12A/B) to temper air being returned from the Control and Relay Rooms to the ACU suction plenums. These heaters are not required to be operable to consider the ACU and ventilation systems operable.

The Makeup Air Fan 2HVC-HVU1 which supplies outside air for smoke removal, has two safety-related disconnect switches (normally in Off) to prevent inadvertent operation. This is to preclude the possibility of allowing unmonitored, unfiltered air to enter the Control Room area, based upon single failure criteria requirements of 2HVC*AOD142.

Control Room smoke removal is also provided for. In order to operate in this mode, *ACU1A and 1B must first be secured. A switch at fire panel 2CEC-P849 is provided with a NORMAL and SMOKE REM position. A similar switch is located at 2HVC-IPNL270 at elevation 288' in the Control Building. When either switch is taken to the SMOKE REM position, the smoke removal fan inlet and outlet dampers will open, smoke removal fan 2HVC-FN3 will start, makeup air fan 2HVC-HVU1 will autostart (REF P&ID53C, ESK62HVC14) and its discharge damper will open.

Relay Room Ventilation

The Relay Room ventilation subsystem supplies tempered recirculated and outside air to all areas of Control Building elevation 288'. The supply capacity exceeds the exhaust capacity to ensure a positive pressure is maintained in the Relay Room.

The subsystem is served by two full capacity, redundant air conditioning units (2HVC*ACU2A and 2B), each consisting of a fan, cooling coil, and filters. In addition, the Computer Room is provided with two air conditioning units (2HVC-ACU4A and 4B) which recirculate internal air plus makeup air from outdoor air booster fans 2HVC-FN17 and FN18.

During normal operation, one Relay Room ACU will be operating with the other in standby. The standby ACU will auto start after a time delay on low air-flow in the running ACU provided that the running unit cooler breaker has opened (i.e., loss of power, control switch to PULL−TO−LOCK, etc.).

Additionally, the ACUs have time delays for sequential starting in case of loss of power to both divisions of AC emergency power. This time delay function is to prevent operation of both 100% redundant unit coolers at the same time.

Normal cooling for *ACU2A (2B) and ACU4A (4B) is provided by the Control Building Chilled Water System. In the event of a failure of both chiller compressors, ACUs may be manually lined up to the Service Water System for backup cooling.

B SYSTEM DESCRIPTION (Continued)

Relay room smoke removal is also provided for. In order to operate in this mode, *ACU2A, 2B, ACU4A, 4B and FN17, 18 must first be secured. A switch at fire panel 2CEC-P849 is provided with a NORMAL and SMOKE REM position. A similar switch is located on 2HVC-IPNL270 at elevation 288' in Control Building. When either switch is taken to the SMOKE REM position, the smoke removal fan inlet and outlet dampers will open, smoke removal fan 2HVC-FN3 will start, makeup air fan 2HVC-HVU1 will autostart (REF P&ID53C, ESK−62HVC14) and its discharge damper will open.

Control Building Special Filter Trains

Outside air for makeup is provided to the Control Room, Relay Room and Computer Rooms via two normally open intake dampers, one located on the west side of the Control Building (2HVC*AOD61B) and one located on the east side of the Control Building (2HVC*AOD61A). Both air intakes are normally open, but either is sized and balanced to provide sufficient outside air to maintain positive pressure in the Control and Relay Rooms.

Outside air from the building intake dampers is routed through two normally open special filter train bypass valves (2HVC*MOV1A and 1B), four radiation elements (2HVC*RE18A through D), and two ventilation chases. Control Room ACUs, Relay Room ACUs and the Computer Room outdoor air booster fans then take air from these chases.

The Control Building special filter trains are normally bypassed and automatically come on line on either a supply air radiation monitor trip system signal or a LOCA signal. The filter train would then provide filtered air to the Control, Relay and Computer Rooms.

The system design is such that both trains of 2HVC special filter trains will simultaneously autostart on a LOCA signal or supply air radiation monitor trip signal. Both fans will continue to run until manually secured. One of the two operating fans is required to be shut down within 20 minutes of initiation on a valid LOCA/Hi radiation signal. This action is required to comply with Engineering analysis of Control Room dose during accident conditions. That fan once placed in standby will autostart if the running fan trips.

On a radiation monitor trip system signal from both elements *RE18A and 18C (18B and 18D) or Div. 1 (2) LOCA signal, 2HVC*MOV1A (1B) will close to divert supply air to special filter train *FLT2A (2B). The corresponding booster fan will auto start. A high-high charcoal absorber temperature will trip the operating train and the standby train will come on-line.

The subsystem is comprised of two redundant filter trains, 2HVC*FLT2A and 2B. Each train has an electric heating element, pre-filter, HEPA filter, charcoal filter, another HEPA filter, and a booster fan. The special filter trains are physically and electrically separated.

With respect to Technical Specifications, the safety function of the Control Room Envelope Filtration (CREF) System consists of maintaining the Control Room Envelope at a 1/8" positive pressure and a suitable/habitable environment following a DBA. The CREF System safety function is maintained when the CREF System components equivalent to one CREF subsystem are OPERABLE (TS 3.7.2 Bases).

B SYSTEM DESCRIPTION (Continued)

Control Room Habitability

The Control Room Habitability Program was added to the Technical Specifications by Amendment 126 and is defined in Technical Specification 5.5.13, Control Room Envelope Habitability Program. Technical Specification 3.7.2 was also modified to initiate actions to implement mitigating actions immediately (3.7.2, Action B.1) and within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical and smoke hazards will not exceed limits (3.7.2, Action B.2). Completion time to restore the Control Room Envelope boundary to OPERABLE status was changed from 24 hours to 90 days (3.7.2, Action B.3).

In Regulatory Guide 1.196, Rev 01, Section 2.7.3, the NRC endorses the use of the guidance in NEI 99-03 Appendix F while corrective actions are being taken to resolve Control Room Habitability Systems that do not meet their licensing bases. The use of mitigating actions described in NEI 99-03, Appendix F is subject to the limitations stated in RG 1.196, Section 2.7.3. NEI 99-03, Appendix F discusses the use of self-contained breathing apparatus (SCBA) or potassium iodide (KI) as acceptable compensatory measures.

NEP-CRH-01, Control Room Integrity Program, contains mitigating actions based on the use of potassium iodide (KI) tablets. Additional compensatory measures, such as the use of temporary sealing materials will reduce the potential dose to the control room occupants during radiological emergencies. However, temporary sealing materials are not sufficient by themselves to satisfy the requirements of mitigating actions required by Technical Specifications (3.7.2, Action B.1).

The Control Building Chilled Water Subsystem provides chilled water (approximately 45 °F or approximately 50 °F if operating with High Service Water temperature) to each of the following ACUs:

- Control Room ACUs 2HVC*ACU1A and 1B
- Relay Room ACUs 2HVC*ACU2A and 2B
- Computer Room ACUs 2HVC-ACU4A and 4B
- Remote Shutdown Room ACUs 2HVC*ACU3A and 3B

The subsystem is a closed loop piping system consisting of two independent, redundant chilled water loops. Each loop consists of a chilled water pump, a chiller compressor, an expansion tank, and the cooling coils of the four ACUs served.

The chiller compressors are York Hermetic Centrifugal R11 units. Each is capable of meeting total chilled water demand.

In the event of failure of both chiller units (2HVK*CHL1A and B) and both circulating pumps (2HVK*P1A and B), Service Water can be manually valved in-service to provide cooling directly to the ACUs cooling coils. The chilled water supply to the Control and Relay Room ACUs are provided with temperature control valves. Chilled water can either flow directly to the cooling coils or be bypassed based upon a temperature controller setpoint.

During High Service Water temperature conditions (>82 °F) Chill Water Outlet Temperature Controller setpoint must be changed from 45 °F to 50 °F. This action will reduce the heat load from the chiller allowing the available Service Water to adequately remove design heat load.

C. OPERATING REQUIREMENTS

The following systems must be in operation in accordance with their respective Operating Procedures to support the Control Room Ventilation System.

- N2-OP-11, Service Water
- N2-OP-16, Makeup Water Storage and Transfer,
- N2-OP-19, Instrument and Service Air,
- N2-OP-71,13.8KV/4160V/600V AC Power Distribution,
- N2 OP 72, Standby and Emergency AC Distribution,
- N2-OP-79, Radiation Monitoring

D. PRECAUTIONS AND LIMITATIONS

D.1. Air temperature in the Control Room Envelope, and Remote Shutdown Rooms must be maintained less than 90 °F to ensure proper operation of electrical and electronic equipment. **[C6]**

- Failure to maintain the Control Room Envelope less than 90 °F will result in the Control Room Envelope Air Conditioning System being INOPERABLE {TS 3.7.3}.
- Failure to maintain the Remote Shutdown Rooms less than 90 °F will result in the Remote Shutdown System being INOPERABLE {TS 3.3.3.2}.

D.2. All pistol grip switches must be reset following an auto trip by going to the STOP, then Normal-After-STOP position.

D.3. Ensure Control Building chiller compressor oil heaters are energized a minimum of 12-hours before starting the compressor, by verifying control power to the chiller is energized, except immediately following compressor oil replacement and with System Engineering concurrence.

D.4. For normal operation, ensure the Control and Relay Room smoke removal fan inlet AODs (2HVC*AOD145 and 148) collared switches on 2CEC*PNL870 are not in the TURN TO CLOSE position and have been reset.

D.5. Do not use the smoke removal mode of operation if an airborne radiation hazard is present.

D.6. Control Room Special Filter Train Booster Fan 2HVC*FN2A (2B) flow rate must be set at 2250CFM (2025-2475). If adjustment is required, fan discharge damper 2HVC*DMPV504 (505) should be opened or closed as necessary to achieve proper flow.

D.7. Control Building ventilation distribution dampers have been set per the system flow balance; these dampers should not normally have their positions changed. If required to change position for maintenance, record the initial position and restore when possible.

D.8. Ensure Technical Specification-required radiation monitors (2HVC*RE18A, B, C, D) are in operation prior to placing the system in service. Refer to Section 3.3.7.1 for Limiting Conditions For Operation if any radiation monitor is found inoperable.

D.9. Ensure the chiller compressor oil pump also marked up when applying any markup on the chiller compressor motor breaker.

D.10. 2HVC*AOD120, 145, and 148 have their instrument air root valves shut to fail the dampers closed. The smoke removal plenum door is normally open to prevent the vent duct from negative pressure conditions. These actions will prevent entry of unfiltered outside air into the Control/Relay rooms during an accident condition. Should the smoke removal function be required, it will be accomplished manually. **[C3]**

D.11. Exposure to high concentrations of Halon 1301 gas may create a health hazard to personnel. Halon activation in the Control Room is preceded by a warning siren and revolving red light and is accompanied by a wintergreen scent. **[C2]**

D.12. The CO_2 System will normally be in standby readiness during all modes of station operation. Care must be exercised during CO_2 System operation. Any discharge may result in hazards to personnel due to oxygen deficiency and reduced visibility. Thirty seconds prior to CO_2 release, horns and revolving red lights are activated near the hazard areas, and also in adjacent areas where CO_2 can collect. **[C2]**

D.13. The chiller manufacturers recommended minimum steady state Service Water inlet temperature is 53 °F. If 2SWP*TV35A(B) is placed in manual/open for an extended period of time, service water inlet temperature may drop below 53 °F, and the chiller may trip on low refrigerant pressure or low oil pressure. **[C9]**

D.14. The following guidelines shall be observed when applying coatings/paint within the Control Building ventilation envelope:

- All painting/coating within the Control Building envelope shall be per an approved Work Order.
- If during or within 8 hours after painting/coating application the Control Building Special Filter train is started, the painting shall be stopped and the daily coating logs (maintained by Mech Maintenance) are to be forwarded to the responsible engineer. The special filter train will remain operable and may remain in operation. Coating thereafter will proceed upon written approval from Engineering and be documented on the Work Document.

D.15. When checking the operability of redundant units or shifting the operation of units, the 2HVC*ACUs may be placed in PULL-TO-LOCK for 5 minutes without declaring the 2HVC inoperable. The following two conditions must be met:

- Cooling can be restored to the affected space upon demand, and
- Control Room or Relay Room temperature prior to this event is 75 °F or less.

D.16. A fire in either division of Control/Relay Room Ventilation, Control Building Special Filter Train, or Control Building Chilled Water, could cause loss of affected division and the auto start feature of the other division. Control Room evacuation may be required per N2-SOP-78. **[C4]**

D.17. With Control Room Ventilation in a normal lineup and 2HVC*FN2A secured, a small amount of flow can be indicated on 2HVC*FR10A. This situation has been evaluated and determined to be due to turbulence in the duct near the flow element (2HVC*FR10A) for 2HVC*FN2A, which is a false reading. Testing has shown that when either special filter train is run, the flow indication is accurate.

D.18. To ensure adequate heat load, one primary air conditioning unit (2HVC*ACU1A/B or 2HVC*ACU2A/B) should be loaded on each chiller unit at all times.

D.19. If both special filters start due to a valid initiation signal, one of the running special filter trains must be secured within 20 minutes after the booster fan starts. This is to prevent exceeding maximum dose to Control Room operators. **[C10]**

D.20. Air conditioning units 2HVC*ACU1A, ACU1B, ACU2A, ACU2B have a time delay in the start circuit to prevent both ACUs (which utilize the same duct) from running at the same time. There will be a time delay when shifting ACUs per this procedure.

D.21. The standby ACU (2HVC*ACU1A, 1B, 2A, 2B) will not auto start on low flow in the running ACU alone. The running ACUs breaker must be open in addition to a low flow condition prior to the standby fan starting.

D.22. Service water backup to HVK is required in Modes 1, 2, 3. Refer to N2-OP-78, Attachment 11, prior to disabling SWP to HVK as compensatory action may be required.

D.23. On a loss of power, the discharge dampers for the Control Room and Relay Room Air Conditioning Units will fail open. Under these conditions, the operating ACUs will be supplying air back through the non-operating ACUs and possibly out of the ductwork via access door seals etc. This could result in higher than normal operating D/p in the area of the non-running ACUs. **[C13]**

D.24. Removal of a Control Room/Relay Room Air Conditioning Unit (ACU) fan from service impacts the operability of the associated Control Room Envelope Filtration (CREF) and Control Room Envelope Air Conditioning (AC) Subsystem. Refer to TS 3.7.2 and 3.7.3.

D.25. Removal of a Control Room/Relay Room cooling coil (e.g. shutdown of associated chiller) from service only affects the operability of the associated Control Room Envelope Air Conditioning Subsystem as long as the ACU fan is intact and can pressurize the Control Room Envelope. Refer to TS 3.7.3.

D.26. Computer Room Supply Fans 2HVC-FN17 and 2HVC-FN18 will trip if both 2HVC-ACU4A and ACU4B are off or if a Control Building halon discharge signal occurs. Downpowering 2FPM-PNL129 will generate a Control Building halon discharge signal.

D.27. Due to the location of the temperature sensing units, it may be necessary to secure the Control Room or Relay Room duct heaters in order to restore automatic control to 2HVK*TV21A(B) or 2HVK*TV22A(B). A high return air temperature sensed downstream of the heaters will prevent the valves from being restored from max cooling. See ARP 870310(311) or 871310(311) for required actions.

D.28. The Recirc Flow Control System is sensitive to sustained temperature changes in the Control Room. The Reactor Recirculation Flow Control valve, 2RCS*HYV17A, will move gradually over a 3-4 hour period with the change in temperature (the valve opens a fraction of 1% when temperature decreases greater than or equal to 4 °F and closes a fraction of 1% when temperature increases greater than or equal to 4 °F). This is due to temperature sensitivity of control circuit components. As such, when performing changes in alignment of the Control Room Chillers, Operations should take positive control of reactivity by closely monitoring Reactor Thermal Power and the "A" Flow Control Valve position as Control Room temperature is likely to change during chiller swaps. Refer to CR 2006-5063 for details.

D.29. Operations may remove from service non-essential “business” equipment in order to maintain control room temperature less than 75 °F or to maintain chilled water temperature between 40 °F to 45 °F or when chiller operating temperatures are outside of expected ranges.

D.30. A chiller trip may occur when exceeding 180 amps. When manually controlling chiller load, closely monitor chiller amps with a target of 90 amps.

H.15. **Overriding the Control Room Envelope ACU Cross-Divisional Operating Interlock**

NOTE

- The control logic for the Control Room Envelope air handling units 2HVC*ACU1A/B **AND** *ACU2A/B includes a start permissive which precludes simultaneous operation the units **AND** is intended to auto start the unit in the opposite division in the event of a trip of the running unit. A fire affecting one division has the potential to override this interlock such that the opposite division ACU may become disabled. This subsection will utilize override switches to ensure operability of the division without the fire.
- Actuation of the override switches is required to be completed within 13 minutes after a fire that has disabled the previously running Control Room Envelope ACUs to ensure Control Room Envelope temperature remains <90 °F.
- This section is only used in response to a confirmed fire in Fire Areas 16, 17, 18, **OR** 19.
- The actions in this section are performed at 2CEC*PNL870(871) unless otherwise specified.
- The Fire Zones listed in Step H.15.1 affect Division 1 components.

H.15.1. **IF** fire is in any of the following Fire Zones:

- 321NW **-** Cable Chase West, Control Bldg, 237' (Fire Area 16) ..... ☐
- 332NW **-** Cable Chase West, Control Bldg, 261' (Fire Area 16) ..... ☐
- 352NW **-** Cable Chase West, Control Bldg, 288'6" (Fire Area 16) ..... ☐
- 371NW **-** Cable Chase West, Control Bldg, 306' (Fire Area 16) ..... ☐
- 333NL **-** Div I Switchgear Room, Control Bldg, 261' (Fire Area 17) ..... ☐ _____

THEN PERFORM the following,
OTHERWISE MARK the following substeps N/A.

a. **CHECK** the following ACUs are available **OR** are in service:

- 2HVC*ACU1B, CONTROL ROOM AC FAN ..... ☐
- 2HVC*ACU2B, RELAY ROOM AC FAN ..... ☐ _____

NOTE

The following steps will cause an auto start of 2HVC*ACU1B **AND** *ACU2B, if **not** already running.

b. **VERIFY** control switch is in PULL**-**TO**-**LOCK for the following:

- 2HVC*ACU1A, CONTROL ROOM AC FAN ..... ☐
- 2HVC*ACU2A, RELAY ROOM AC FAN ..... ☐ _____

H.15.1 (Continued)

c. **NOTIFY** SM to declare 2HVC*ACU1A **AND** 2HVC*ACU2A inoperable **AND** to **TAKE** actions per TS 3.7.2 **AND** 3.7.3. _____

d. **OBTAIN** SM permission to override HVC ACU cross divisional operating interlock for 2HVC*ACU1A **AND** 2HVC*ACU2A. _____
SM

NOTE

If 2HVC*ACU1B is **not** running, the following step should cause an automatic start of 2HVC*ACU1B.

e. At 2CEC*PNL871, **PLACE** keylock switch APPENDIX R CR OVERRIDE SW HVC*ACU1B, to OVERRD. _____

_____
IV

f. **VERIFY** 2HVC*ACU1B running. _____

g. **WHILE** continuing in this subsection, wait approximately two minutes,
THEN CONFIRM proper operation of 2HVC*ACU1B. _____

h. **PLACE** 2HVC*ACU1B control switch, CONTROL ROOM AC FAN, in Normal-After-START. _____

NOTE

If 2HVC*ACU2B is **not** running, the following step should cause an automatic start of 2HVC*ACU2B.

i. At 2CEC*PNL871, **PLACE** keylock switch APPENDIX R RR OVERRIDE SW HVC*ACU2B, to OVERRD. _____

_____
IV

j. **VERIFY** 2HVC*ACU2B running. _____

k. **WHILE** continuing in this subsection, wait approximately two minutes,
THEN CONFIRM proper operation of 2HVC*ACU2B. _____

l. **PLACE** 2HVC*ACU2B control switch, RELAY ROOM AC FAN, in Normal-After-START. _____

H.15.1 (Continued)

m. **WHEN** override is **no** longer required, (system configuration back to normal),
THEN PERFORM the following:

1) **OBTAIN** SM permission to restore HVC ACU cross divisional operating interlock for 2HVC*ACU1A **AND** 2HVC*ACU2A. _____ SM

2) At 2CEC*PNL871, **PLACE** keylock switch APPENDIX R CR OVERRIDE SW HVC*ACU1B, to NORMAL. _____

_____ IV

3) At 2CEC*PNL871, **PLACE** keylock switch APPENDIX R RR OVERRIDE SW HVC*ACU2B, to NORMAL. _____

_____ IV

4) **NOTIFY** SM that HVC cross divisional interlocks are restored. _____

NOTE

The Fire Zones listed in Step H.15.2 affect Division II components.

H.15.2. **IF** fire is in any of the following Fire Zones:

- 304NW **-** Electrical Tunnel, 230°, 214'6" (Fire Area 18)..... ☐
- 324NW **-** Cable Chase East, Control Bldg, 237' (Fire Area 18) ..... ☐
- 337NW **-** Cable Chase East, Control Bldg, 261' (Fire Area 18) ..... ☐
- 359NW **-** Cable Chase East, Control Bldg, 288'6" (Fire Area 18) ..... ☐
- 377NW **-** Cable Chase East, Control Bldg, 306' (Fire Area 18) ..... ☐
- 336NL **-** Div II Switchgear Room, Control Bldg, 261' (Fire Area 19) ..... ☐ _____

THEN PERFORM the following,
OTHERWISE MARK the following substeps N/A.

a. **CHECK** the following ACUs are available **OR** are in service:

- 2HVC*ACU1A, CONTROL ROOM AC FAN ..... ☐
- 2HVC*ACU2A, RELAY ROOM AC FAN ..... ☐ _____

H.15.2 (Continued)

NOTE

The following steps will cause an auto start of 2HVC*ACU1A **AND** *ACU2A, if **not** already running.

b. **VERIFY** control switch is in PULL-TO-LOCK for the following:

- 2HVC*ACU1B, CONTROL ROOM AC FAN ..... ☐
- 2HVC*ACU2B, RELAY ROOM AC FAN ..... ☐ _____

c. **NOTIFY** SM to declare 2HVC*ACU1B **AND** 2HVC*ACU2B inoperable **AND** to **TAKE** actions per TS 3.7.2 **AND** 3.7.3. _____

d. **OBTAIN** SM permission to override HVC ACU cross divisional operating interlock for 2HVC*ACU1B **AND** 2HVC*ACU2B. _____
SM

NOTE

If 2HVC*ACU1A is **not** running, the following step should cause an automatic start of 2HVC*ACU1A.

e. At 2CEC*PNL870, **PLACE** keylock switch APPENDIX R CR OVERRIDE SW HVC*ACU1A, to OVERRD. _____

_____
IV

f. **VERIFY** 2HVC*ACU1A running. _____

g. **WHILE** continuing in this subsection, wait approximately two minutes,
THEN CONFIRM proper operation of 2HVC*ACU1A. _____

h. **PLACE** 2HVC*ACU1A control switch, CONTROL ROOM AC FAN, in Normal-After-START. _____

H.15.2 (Continued)

NOTE

If 2HVC*ACU2A is **not** running, the following step should cause an automatic start of 2HVC*ACU2A.

i. At 2CEC*PNL870, **PLACE** keylock switch APPENDIX R RR OVERRIDE SW HVC*ACU2A, to OVERRD. _____

_____
IV

j. **VERIFY** 2HVC*ACU2A running. _____

k. **WHILE** continuing in this subsection, wait approximately two minutes,
THEN CONFIRM proper operation of 2HVC*ACU2A. _____

l. **PLACE** 2HVC*ACU2A control switch, RELAY ROOM AC FAN, in Normal-After-START.

m. **WHEN** override is **no** longer required (system configuration back to normal),
THEN PERFORM the following:

1) **OBTAIN** SM permission to restore HVC ACU cross divisional operating interlock for 2HVC*ACU1B **AND** 2HVC*ACU2B _____
SM

2) At 2CEC*PNL870, **PLACE** keylock switch APPENDIX R CR OVERRIDE SW HVC*ACU1A, to NORMAL _____

_____
IV

3) At 2CEC*PNL870, **PLACE** keylock switch APPENDIX R RR OVERRIDE SW HVC*ACU2A, to NORMAL. _____

_____
IV

4) **NOTIFY** SM that HVC cross divisional interlocks are restored. _____

H.16. **Voluntary Entry Into Tech Spec Condition 3.7.2.B [C14]**

H.16.1. **IF** voluntary entry into Tech Spec Condition 3.7.2.B is required,
THEN PERFORM the following:

- **IF** determined necessary, in unlikely event that a hazardous environment arises,
  THEN VERIFY Self Contained Breathing Apparatus (SCBA) equipment is available for operating crew to use. _____
- **NOTIFY** Security Department of any impairment affecting security access control **AND VERIFY** applicable sections of Security Plan are implemented. _____
- **IF** an impairment exists that precludes maintaining acceptable temperature in Control Room areas (Main Control Room **OR** Relay Room),
  THEN VERIFY that Tech Spec Conditions 3.7.3.C **AND** 3.7.3.D, as applicable, are entered for inoperable Control Room chillers. _____
- **IF** impairment exists that affects fire protection features,
  THEN VERIFY that applicable requirements of Fire Protection Program are implemented. _____

Training ID: **Nov 2023 NRC JPM C** Revision: **0.0**

Title: **Control Rod Exercising (Alternate Path)**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mike Carno	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OSP-RMC-W@001, Control Rod Movement and Position Indicator Verification
2. N2-ARP-603400, 2CEC*PNL603 Series 400 Alarm Response Procedures
3. N2-OP-30, Control Rod Drive
4. NUREG 1123 K/A 201002 A4.01 (4.1)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to manipulate controls associated with the Reactor Manual Control. The operator will perform N2-OSP-RMC-W@001, Control Rod Movement and Position Indicator Verification.

   b. This JPM is considered alternate path because, when the coupling check is performed for control rod XX-YY, the control rod will be uncoupled. The Operator will recouple the control rod using N2-OP-30 section H.2.0.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-201003-01004, Couple a Control Rod

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☒ Yes ☐ No			
Safety Function:	1		Reactivity Control	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	201002 Reactor Manual Control System A4.01 - Ability to manually operate and/or monitor in the control room: Rod movement control switch			
K/A Importance Rating:	RO	4.1	SRO	-

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) **RD08-18-59**, Control Rod Failure – Uncoupled, Final = true **Inserted**

2) Remotes

a) None

3) Overrides

a) None

4) Annunciators

a) None

5) Event Triggers

Event #	Event Action	Command
1	rdvposb(179) < 740 .AND. anntbl(644) >0 (control rod 18-59 at position 46 or below with overtravel alarm in)	dmf rd08-18-59

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-OSP-RMC-W@001 complete up to step 6.1.1

8) Miscellaneous

a) IC-206 (For ILT 22-2, paired with JPM E)

-OR-

b) IC setup:

(1) Reactor power should be at 98%

6. Strategy Code

a. None

7. Tools and Equipment

a. None

8. Commitments

a. None

9. Prerequisites

   a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

   a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant operating at approximately 98% power. • N2-OSP-RMC-W@001, Control Rod Movement and Position Indicator Verification, is to be performed for control rod 18-59. • Control rod 18-59 is NOT known to double-notch. • Prerequisites have been verified. • Stroke time trending is NOT to be performed. • There is NO known fuel failure. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *perform N2-OSP-RMC-W@001 for control rod 18-59. The procedure is in progress up to step 6.1.1.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
2.	**SELECT** control rod to be tested.	P (6.1.1)	***PASS / FAIL** ***STD:*** *Depresses select pushbutton for control rod 18-59.* ***Failure =*** *Control rod 18-59 not selected*
3.	**VERIFY** there are **no** Rod Insert Blocks present.	P (6.1.2)	SAT / UNSAT ***STD:*** *Observes that there are no rod insert blocks present.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
4.	**RECORD** control rod's initial position on Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet **OR** Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect.	P (6.1.3)	SAT / UNSAT ***STD:*** *Records initial position for control rod 18-59 as "48" on Attachment 1.*
5.	For control rods on Attachment 2, **RECORD** computer indicated rod position on Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer) **Cue:** Computer indicated rod position is "48".	P (6.1.4)	SAT / UNSAT ***STD:*** *Records computer indicated rod position for control rod 18-59 as "48" on Attachment 2.*
Procedure Notes	• For insert rod timing, the stopwatch should be started when the initial position extinguishes and stopped when the target position extinguishes. • If the target position does not extinguish on the insert, the CRDM is too slow. This is to be noted on the Attachment 4 data sheet in the comments. • Insert rod speed should be within 1.8 to 2.2 seconds, with an optimum 2.0 second target for the single notch insertion; which corresponds to a full stroke time of 43.2 to 52.8 seconds with a target value of 48 seconds. • To have optimum performance, the insert time should be less than the withdraw time. • Any double-notch events need to be documented in an IR.		
6.	**PERFORM** one of the following to insert control rod: Using normal Drive Water differential pressure of 250 to 265 psid, **INSERT** control rod one notch.	P (6.1.5)	***PASS / FAIL** ***STD:*** *Depresses INSERT pushbutton.* ***Failure*** *= Control rod 18-59 does NOT reach position 46.*
7.	**IF** control rod inserts but does **not** reach next notch due to being slow…	P (6.1.6)	SAT / UNSAT ***STD****: Determines control rod reaches next notch and marks step N/A.*
8.	**IF** stroke timing was performed…	P (6.1.7)	SAT / UNSAT ***STD****: Determines stroke timing was not performed and marks step N/A.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
9.	**IF** control rod double-notches…	P (6.1.8)	SAT / UNSAT ***STD**: Determines control rod did not double notch and marks steps N/A.*
10.	**IF** control rod does **not INSERT** with normal drive pressure…	P (6.1.9)	SAT / UNSAT ***STD**: Determines control rod inserted with normal drive pressure and marks steps N/A.*
11.	**IF** position indication is **not** operable at one notch inserted…	P (6.1.10)	SAT / UNSAT ***STD**: Determines position indication is operable and marks steps N/A.*
12.	**RECORD** new control rod position on Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet **OR** Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect.	P (6.1.11)	SAT / UNSAT ***STD**: Records new position for control rod 18-59 as "46" on Attachment 1.*
13.	For control rods on Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer), **RECORD** computer indicated rod position on Attachment 2. **Cue:** Computer indicated rod position is "46".	P (6.1.12)	SAT / UNSAT ***STD**: Records computer indicated rod position for control rod 18-59 as "46" on Attachment 2.*
14.	**MONITOR** nuclear instrumentation response.	P (6.1.13)	SAT / UNSAT ***STD**: Observes APRMs indicate stable.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
15.	**CONFIRM** 4-Rod Display indicates proper rod position, **AND IF** so, **THEN CHECK-OFF** this step on Control Rod Movement and Position Indication Verification Data Sheet.	P (6.1.14)	SAT / UNSAT ***STD**: Observes 4-rod display indicates position 46 for control rod 18-59. Checks off step on Attachment 1.*
Procedure Notes	• "Almost double-notching" is described as rod speed observed to be too fast when compared to other normal rod movements or the rod moves too far past the intended notch but does not latch into the next notch. • The intent of identifying these rods is that this rod has the potential to double-notch and, therefore, should be adjusted before it does. • Adjustments for almost double-notching do not have to be immediately retested but should be entered into the U2ESL/CRDDL for observance during the next surveillance performance.		
16.	**IF** control rod withdraw almost double-notches…	P (6.1.15)	SAT / UNSAT ***STD**: Determines control rod does not almost double notch.*
Procedure Notes:	• When performing withdraw notch timing, perform withdrawal from 46 to 48 and coupling check in one manipulation to prevent erroneous data due to the rod settle function. • For withdraw rod timing, the stopwatch should be started when the initial position is re-illuminated after the double dash appears AND stopped when the target position is illuminated. • If control rod does not withdraw to the target position, then the CRDM is too slow. • Withdraw rod speed should be within 1.8 to 2.2 seconds, with an optimum 2.0 second target for the single notch withdraw. This corresponds to a full-stroke time of 43.2 to 52.8 seconds with a target value of 48 seconds. To have optimum performance, the insert time should be less than the withdraw time. However, since the withdraw timer relay energizes the DCV 2RDS*SOV122 for only 1.5 seconds, the withdraw time only provides a trend information of CRDM speed. • If control rod being tested initial position is 48, then coupling check per step 6.1.21 may be performed concurrently with step 6.1.16. • Any double-notch events need to be documented in an IR.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Procedure Caution:	• Withdrawing a control rod past its initial position could cause fuel damage due to preconditioning violations.		
17.	**PERFORM** one of the following to withdraw control rod: Using normal Drive Water Differential Pressure of 250 to 265 psid, **WITHDRAW** control rod to its pre-test position.	P (6.1.16)	***PASS / FAIL** ***STD:*** *Depresses WITHDRAW pushbutton.* ***Failure =*** *Control rod 18-59 does NOT reach position 48.*
18.	**IF** stroke timing was performed…	P (6.1.17)	SAT / UNSAT ***STD****: Determines stroke timing was not performed and marks step N/A.*
19.	**IF** control rod double-notches…	P (6.1.18)	SAT / UNSAT ***STD****: Determines control rod did not double notch and marks step N/A.*
20.	**IF** control rod will **not** withdraw with normal drive pressure of 250 to 265 psid…	P (6.1.19)	SAT / UNSAT ***STD****: Determines control rod withdrew with normal drive pressure and marks step N/A.*
21.	**IF** control rod withdraw almost double-notches…	P (6.1.20)	SAT / UNSAT ***STD****: Determines control rod does not almost double notch.*
Alternate Path:	During performance of the following step, the control rod will overtravel due to being uncoupled. The Operator will reference ARP 603444 and proceed to N2-OP-30 to re-couple the control rod.		
Procedure Note:	• Step 6.1.21 shall be performed only for control rods that have final positions of 48.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
22.	**IF** Control Rod's final position is 48, **THEN PERFORM** the following, **OTHERWISE MARK** these steps N/A: **ATTEMPT** to withdraw past position 48, by **DEPRESSING AND HOLDING BOTH** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons.	P (6.1.21.1)	***PASS / FAIL** ***STD:*** *Depresses and holds both WITHDRAW and CONTINUOUS WITHDRAW pushbuttons.* ***Failure*** *= Control rod 18-59 does NOT move past position 48.*
23.	**CONFIRM AND RECORD** red Full Out position light for control rod is lit. **CONFIRM AND RECORD** four rod display shows position 48 **CONFIRM AND RECORD** 603444 CONTROL ROD OVERTRAVEL does **not** alarm **IF** Control Rod Drive withdraw stall flow on C12-R604 (2RDS-FIY112) is less than or equal to 3.5 gpm, **THEN RECORD** on Attachment 1 **OR** Attachment 3.	P (6.1.21.2-5)	SAT / UNSAT ***STD****: Determines control rod 18-59 is uncoupled.*
24.	**RELEASE** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons	P (6.1.21.6)	***PASS / FAIL** ***STD:*** *Releases both WITHDRAW and CONTINUOUS WITHDRAW pushbuttons.* ***Failure =*** *WITHDRAW and CONTINUOUS WITHDRAW pushbuttons NOT released.*
25.	References ARP 603444.	P	SAT / UNSAT ***STD****: References ARP 60344.*
26.	Obtains N2-OP-30 Section H.2.0.	P	SAT / UNSAT ***STD****: Obtains N2-OP-30 Section H.2.0.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Procedure Notes:	• While performing rod coupling check, any rod which becomes uncoupled from its drive mechanism will energize annunciator 603444, CONTROL ROD OVERTRAVEL. • While drive is being moved, rod uncoupling could also be indicated by lack of a noticeable change in neutron monitoring indication. • Exact location of rod (vertically) may be determined by a TIP trace.		
27.	Declare control rod INOPERABLE. **Role Play:** Acknowledge need to declare control rod 18-59 inoperable.	P (N2-OP-30 H.2.1)	SAT / UNSAT ***STD**: Notifies SRO of need to declare control rod 18-59 inoperable, proper communications used.*
28.	Perform the following within three hours per Tech Spec 3.1.3: IF permitted by Rod Worth Minimizer (RWM may be bypassed as allowed by TS 3.3.2.1), THEN perform the following: Insert control rod two notches in an attempt to recouple rod. **Examiner Note:** Based on the wording of this procedure step, with the control rod initially in the overtravel position, the operator may consider inserting 2 notches either going to position 46 or 44.	P (N2-OP-30 H.2.2.1.a)	***PASS / FAIL** ***STD:** Depresses INSERT pushbutton as needed to insert control rod 18-59 to position 44 or 46.* ***Failure** = Control rod 18-59 does NOT reach position 44 or 46.*
Booth Operator:	When control rod 18-59 reaches position 46, verify **TRG1** automatically initiates and **deletes malfunction RD08-18-59**. This re-couples the control rod.		
29.	Consult with Reactor Engineering before control rod withdrawal. **Role Play:** As Reactor Engineering, inform the Operator that you concur with continuing in the procedure and withdrawing the control rod.	P (N2-OP-30 H.2.2.1.b)	SAT / UNSAT ***STD**: Contacts Reactor Engineering about withdrawing control rod, proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
30.	Withdraw control rod to position 48 with Reactor Engineering concurrence AND observe nuclear instrumentation response during rod movement.	P (N2-OP-30 H.2.2.1.c)	***PASS / FAIL** ***STD:*** *Depresses WITHDRAW pushbutton.* ***Failure =*** *Control rod 18-59 does NOT reach position 48.*
31.	Apply a continuous withdraw signal at position 48 AND verify 603444, CONTROL ROD OVERTRAVEL, does NOT alarm. **Note:** To apply a continuous withdraw signal, the Operator must depress and hold both WITHDRAW and CONTINUOUS WITHDRAW pushbuttons.	P (N2-OP-30 H.2.2.1.d)	***PASS / FAIL** ***STD:*** *Depresses and holds both WITHDRAW and CONTINUOUS WITHDRAW pushbuttons.* ***Failure*** *= WITHDRAW and CONTINUOUS WITHDRAW pushbuttons NOT both depressed.* SAT / UNSAT ***STD****: Observes annunciator 603444 does NOT alarm.*
Evaluator Note:	Once the coupling check is successfully performed: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*Control rod 18-59 tested per N2-OSP-RMC-W@001. Control rod un-coupling detected. Control rod re-coupled per N2-OP-30 section H.2.0. Control rod coupling check completed.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The plant operating at approximately 98% power. • N2-OSP-RMC-W@001, Control Rod Movement and Position Indicator Verification, is to be performed for control rod 18-59. • Control rod 18-59 is NOT known to double-notch. • Prerequisites have been verified. • Stroke time trending is NOT to be performed. • There is NO known fuel failure.

INITIATING CUE	**(Operator Name)**, *perform N2-OSP-RMC-W@001 for control rod 18-59. The procedure is in progress up to step 6.1.1.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant operating at approximately 98% power. • N2-OSP-RMC-W@001, Control Rod Movement and Position Indicator Verification, is to be performed for control rod 18-59. • Control rod 18-59 is NOT known to double-notch. • Prerequisites have been verified. • Stroke time trending is NOT to be performed. • There is NO known fuel failure.

INITIATING CUE	**(Operator Name)**, *perform N2-OSP-RMC-W@001 for control rod 18-59. The procedure is in progress up to step 6.1.1.*

Nine Mile Point Nuclear Station Unit 2
TECHNICAL PROCEDURE

N2-OSP-RMC-W@001

CONTROL ROD MOVEMENT AND POSITION INDICATOR VERIFICATION

Revision 01100

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-18-05352:

- Description: Precaution 4.8 regarding a previous timing problem associated with the NSSS Graphics Video Control Rod Display is **no** longer applicable, and may be deleted.
- Reason: With the new R*Time Plant Process Computer (PPC), this precaution is **no** longer relevant. The “NSSS Graphics Video CR Display” and OD-7 refer to old Honeywell PPC features. The procedure later tells the Operator to get the Attachment 2 data from the computer which is sufficient.

PCR-18-06159:

- Description: Delete previous revision outdated legacy precautions, reduced steps to one action per step.
- Reason: Per IR 04172818, PIIM 2018-0380 Operations Procedure Excellence:
- 2.0, 4.0, and 5.0 deleted:
    - Use of N/A may only be used as directed in this procedure, except as allowed per HU-AA-104-101, Procedure Use and Adherence.
    - The Shift Manager (SM) shall be notified immediately whenever a procedural step **cannot** be completed as stated, or if other problem develops during the test.
    - Personnel responsible for the performance of this test must read and thoroughly understand its contents prior to participating.
    - Prior to initialing any step in this procedure, all individuals shall place their initials, signatures, and printed names on Attachment 1.
- 6.5.3.2, 6.5.4.2, 6.6.4.1, and 6.6.6.1 revised to one action per step to allow placekeeping removal of the needle valve cap.

PCR-20-01693:

- Description: Update directional control valve wrench precaution and sections for Control Rod Speed Adjustments.
- Reason: Modification to select directional control valve needle valves during N2R17, now using different sizes wrenches to make control rod speed adjustments.
- 4.10 clarified that two wrenches are required for both the needle valve and lock nut to prevent inadvertent adjustment of the needle valve.
- 6.3.1 preceding note, corrected misspelling of “stop watch” to stopwatch.
- 6.5.3.3, 6.5.3.4, and 6.5.3.5 removed specific wrench sizes.
- 6.5.4.3, 6.5.4.4, and 6.5.4.5 removed specific wrench sizes.
- 6.6.4.2, 6.6.4.3, 6.6.4.4, and 6.6.4.5 removed specific wrench sizes.
- 6.6.6.2, 6.6.6.3, 6.6.6.4, and 6.6.6.5 removed specific wrench sizes.

SUMMARY OF ALTERATIONS (Continued)

PCR-21-02113:

- Description: Changed frequency to per SFCP with recent change from monthly to quarterly.
- Reason: Implementation of N2-STI-017, Extend the Frequency for the Nine Mile Point Unit 2 Control Rod Movement and Position Indicator Verification Test from 31 days to a Quarterly (92 days) Frequency.
- changed Frequency to: per SFCP2, Nine Mile Point, Unit 2 Surveillance Frequency Control Program, List of Surveillance Frequencies.
- 3.1 added N2-STI-017.

Procedure Writer Editorial Changes:

- Updated N/A statements to comply with AD-NM-101-1005 5.4.C.13 and in preparation for the transition to Exelon Smart Procedures, by removing N/A check-boxes, made into **IF/THEN** statements and "**OTHERWISE MARK** this step (or these steps) N/A", **no** revision bars used.
- Updated fonts for **ACTION VERBS** and **LOGIC WORDS**, bold and lowercase underline for **not**, **cannot** and **no** per AD-NM-101-1005 and AD-AA-101-1002, **no** revision bars used.
- 2.2 Special Tools, and Equipment Recommended moved from previous revision prerequisite step 5.1, where it existed as **non**-actionable text, and **not** a numbered, placekept step.
- 3.2 added NF-AB-440-1001, Fuel Conditioning For 3D Monicore Plants.
- 4.12 added new Limitation that: “Test equipment may be substituted with approval of Engineering” per IR 02616500, MA-AA-716-040 (Step 4.7.3 on Rev. 12) and AD-AA-102-1001.
- 5.3 revised previous revision prerequisite to **OBTAIN** a stopwatch, instead of recording the calibration due date of stopwatches as digital electronic stopwatches are **no** longer calibrated per MA-AA-716-040 Control of Portable Measurement and Test Equipment Program, Exelon Corporate IR# 00344792.
- 5.18, 5.19, and 5.20 moved into prerequisites from section 6.1, as they are actually prerequisites, and do **not** belong in the PERFORMANCE section of the procedure. This change also standardizes placement of PLANT IMPACT steps where they are most appropriate in the procedure, **no** revision bars used.
- 6.1.16 moved **CAUTION** to before step and after Notes per AD-NM-101-1005.
- 7.3.1 revised IST Programs Engineer review and signature to:
  EMAIL a completed procedure copy to IST Programs Engineer for review.
- Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet and Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect:
  - Standardized various micro fonts to Arial 11 pt.
  - Equalized column widths.
  - Un-abbreviated terms in headers.

SUMMARY OF ALTERATIONS (Continued)

Procedure Writer Editorial Changes (continued):

- For the upcoming transition to Exelon Smart Procedures (ESP), and for proper step numbering and placekeeping, moved the following previous revision attachments into the body of the procedure:
  - 6.4 Adjustment of Almost Double-Notching Rods, Continuous IN, High Drive Pressure to Withdraw, or Double-Notching Rods,
    previously Attachment 4 Adjustment of Almost Double Notching Rods, Continuous IN, High Drive Pressure to Withdraw, or Double Notching Rods.
  - 6.6 Adjustment of Single-Notch Timed Control Rods,
    previously Attachment 6 Adjustment of Single Notch Timed Control Rods.

Operator Review:

- 2.3.1 2nd bullet, improved “V122” to 2RDS*SOV122.
- 3.3.1 added new definition for **U2ESL/CRDDL**: Unit 2 Control Rod Drive Deficiency Log.
- 5.15 preceding note, added 3rd bullet from previous incorrect placement after steps 5.15.1, 5.15.2 and 5.15.3.
- 5.15.1 added “page 1”, deleted “sheet 1” that doesn’t exist, and added IV of step.
- 5.15.2 added “page 2”, deleted “sheet 2” that doesn’t exist, and added IV of step.
- 5.15.3 added “page 3”, reworded action to “**LIST** all control rods which are” deleted “sheet 3” that doesn’t exist.
- 6.1 preceding note 2nd bullet defined abbreviation U2ESL/CRDDL.
  Throughout the rest of the procedure, standardized abbreviation of U2ESL/CRDDL.
- 6.1.3 corrected incorrect reference to Attachment 2 to Attachment 1 **OR** 3.
- 6.1.9.2 added: **RECORD** failure to insert in attachment remarks.
- 6.1.11 updated wording to reference to Attachment 1 **OR** 3.
- 6.1.15 updated to mimic layout of step 6.1.20 and break up multi-action steps.
- 6.1.16 preceding note, 4th bullet improved “DCV 122” to: DCV 2RDS*SOV122.
- 6.1.18 broke up multi-action steps, directed repeat test to determine if adjustment was successful.
- 6.1.21.7 placed before step 6.1.21.8 for improved procedure flow.
- 6.1.21.7.d placed before step 6.1.21.7.e for improved procedure flow.
- 6.1.22.5 placed before step 6.1.22.6 for improved procedure flow.
- 6.1.22.5.d placed before step 6.1.22.5.e for improved procedure flow.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
1.0	PURPOSE	6
2.0	APPLICABILITY/SCOPE	6
3.0	REFERENCES AND DEFINITIONS	9
4.0	PRECAUTIONS AND LIMITATIONS	10
5.0	PREREQUISITES	12
6.0	PERFORMANCE	17
6.1.	Control Rod Exercise Test	17
6.2.	Control Rod Exercise Test for Attachment 3 Control Rods (Failed Fuel)	31
6.3.	Adjustment and Verification of Single Notch Timed Control Rods	32
6.4.	Adjustment of Almost Double-Notching Rods, Continuous IN, High Drive Pressure to Withdraw, or Double-Notching Rods	33
6.5.	Adjustment of Almost Double-Notching Rods	34
6.6.	Adjustment of Single-Notch Timed Control Rods	36
6.7.	Return to Normal	39
6.8.	Acceptance Criteria	41
7.0	POST-PERFORMANCE ACTIVITIES	42
8.0	BASES	43
9.0	RECORDS	43
Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet		44
Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer)		60
Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect		61
Attachment 4, CRDM Stroke Time Trending		64

1.0 PURPOSE

To perform control rod operability testing per Technical Specifications.

2.0 APPLICABILITY/SCOPE

2.1. Frequency and Condition

2.1.1. Operational Conditions 1 and 2.

2.1.2. This procedure shall be performed as follows:

- Per SFCP2, Nine Mile Point, Unit 2 Surveillance Frequency Control Program, List of Surveillance Frequencies for control rods at position 48, though **not** required to be performed until 31 days after the control rod is withdrawn and Thermal Power is greater than the LPSP of the RWM.
- Per SFCP2, Nine Mile Point, Unit 2 Surveillance Frequency Control Program, List of Surveillance Frequencies for control rods **not** at position 00, though **not** required to be performed until 31 days after the control rod is withdrawn and Thermal Power is greater than the LPSP of the RWM.
- Within 24 hours from discovery of a stuck withdrawn control rod with RWM greater than LPSP.
- Prior to declaring control rod operable after work on control rod or CRD system that could affect coupling.

2.1.3. Stroke time measurement trending of CRDM speeds for insert and withdraw will normally be done:

- During the first exercising after an outage.
- During the first exercising after a sequence exchange/scram time measurement.
- For CRDM performance trending as recommended by System Engineering.

2.2. Special Tools, and Equipment Recommended

2.2.1. Stopwatch

2.3. **Discussion**

2.3.1. Demonstrate the operability of each withdrawn operable control rod (02 to 48), by inserting the respective control rod at least one notch.

2.3.2. Demonstrate the integrity of the Cooling Water Orifice installation, by verifying that the control rods insert properly with normal Drive Water pressure.

2.3.3. Demonstrate the operability of each control rod's position indication by verifying that the control rod movement is properly indicated.

2.3.4. Demonstrate that each applicable control rod is coupled to its drive mechanism by verifying that the control rod drive mechanism does **not** go to the overtravel position.

2.3.5. Demonstrate that each applicable control rod position indicator corresponds to the control rod "Full Out" position when performing the control rod coupling integrity verification.

2.3.6. Performance of this procedure satisfies the following Surveillance Requirements:

- 5.5.6, Inservice Testing Program
- SR 3.1.3.3, Control Rod Operability
- SR 3.1.3.5, Control Rod Operability

2.3.7. Proper control rod motion also satisfies reverse flow test requirements for the HCU Cooling Water Check Valves, 2RDS*V138 valves. **[IST]**

2.3.8. Proper control rod motion also satisfies test requirements for HCU Drive Water Check Valves, 2RDS*V137 valves. **[IST]**

2.3.9. Control rods that are fully inserted will be tested to ensure proper operation of their respective HCU Cooling Water Check Valves, 2RDS*V138 valves.

2.3.10. Obtain Insert Stall Flow for all rods that are fully inserted, for monitoring Cooling Water Orifice installation integrity and drive-in seal condition trending.

2.3.11. Control rod insert speed and withdraw speed are to be verified during the performance of this procedure as stated in section 2.1 Frequency and Condition; the purpose of this is to obtain trending information of CRDM performance.

- The insert single notch stroke time is to be normally 2 seconds $\pm$ 0.2 seconds.
- The withdraw time will need to be trended because the Directional Control Valve (DCV) 2RDS*SOV122 is open for approximately 1.5 seconds, which is less than the nominal 2 second stroke time. This trending information will provide data to Reactor Engineering and System Engineering to determine which CRDMs will need to have stroke time adjustments made.
- An IR will be generated by System Engineering or Reactor Engineering for stroke time adjustments that may be necessary.
- Operations is to verify control rod insert speed by performing single notch insertion stroke timing and control rod withdraw speed by performing single notch withdraw timing. This will be done at the same time normal weekly exercising is performed.

2.3.12. Control rods that are **not** withdrawn and have their directional control valves disarmed electrically or hydraulically, are **not** required to be tested by the implementation of this procedure, and shall have N/A placed in the appropriate signoffs in the Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet or Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect, as applicable.

2.3.13. Any control rod that has an initial position of 48 shall be demonstrated to be coupled to its drive mechanism by verification that the control rod does **not** go into the OVERTRAVEL position, if its post-test position is also 48. This verification shall be recorded in the Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet or Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect, as applicable.

2.3.14. Control rod selection and surveillance may be performed in any order as best facilitates implementation of the procedure.

2.3.15. Control rods **not** at position 00 will be inserted one notch and then will be withdrawn to their original position.

2.3.16. Control rods at position 00 will be given an insert signal, and movement will be verified.

2.3.17. Each time a control rod is re-positioned per this procedure, the position indication for each prescribed position shall be verified to be operable.

2.3.18. This procedure may be used for Post Maintenance Testing and/or verification of operability for any of the equipment covered by this procedure. When used for this purpose, all sections shall be performed. All IST tests applicable to a particular component are required to be performed to verify operability.

3.0 REFERENCES AND DEFINITIONS

3.1. Developmental References

- USAR Section 4.6.1, Control Rod Drive System
- USAR Section 4.6.3, Testing and Verification of the CRDS
- LCO 3.1.3, Control Rod Operability
- NMP2 Inservice Pump and Valve Testing Program Plan
- GEK-39469B, Hydraulic Control Unit
- GEK-83317 VOL III, Parts 3 and 4, Control Rod Drive System
- 12177-PID-30A through 30C, Control Rod Drive Hydraulic System
- GE SIL 538, Cooling Water Orifices, dated November 4, 1991
- IR 02062099 (legacy CR-1991-001558), Response to GE SIL 538, Cooling Water Orifices
- IR 02028746 (legacy CR-2007-000101), INPO AFI – ER.1-2, Engineering has **not** developed consistent and aggressive strategies to improve the reliability of the Unit 2 control rod drive (CRD) system
- DCP N2-07-005, ARTS
- NER NC-18-021-YELLOW, Aggregate Effects of Control Rod Drive Mechanism Degradation and Channel Distortion
- GEH, TP16-2-205, Revision 0, Shaker Test Project – Updated Fuel Channel Bow Surveillance Recommendations and Friction Limits For BWR/2-5.
- N2-STI-017, Extend the Frequency for the Nine Mile Point Unit 2 Control Rod Movement and Position Indicator Verification Test from 31 days to a Quarterly (92 days) Frequency.

3.2. Performance References

- N2-OP-30, Control Rod Drive System
- N2-OP-95A, Rod Worth Minimizer System
- N2-OP-96, Reactor Manual Control and Rod Position Indication System
- NF-AB-440-1001, Fuel Conditioning For 3D Monicore Plants

3.3. Definitions

3.3.1. **U2ESL/CRDDL**: Unit 2 Control Rod Drive Deficiency Log.

4.0 PRECAUTIONS AND LIMITATIONS

4.1. Neutron monitoring instrumentation shall be observed for proper response whenever a control rod is being moved.

4.2. Recovery from all control rod malfunctions, such as mispositioned control rod, or stuck control rod, shall be per the direction given in Off-Normal section of Operating Procedures, and by the direction of the SM.

4.3. The operator exercising control rods should **not** exercise more than 50 rods without taking a break.

4.4. Do **not** remove a lens cover on the Rod Select Module to replace a lightbulb for a rod that is selected, as damage could occur to the Reactor Manual Control System if an object falls onto the exposed contacts. A different rod shall be selected prior to removing a lens cover for a rod on the select module.

4.5. The steps in the body of this procedure provide instructions for the surveillance of each control rod. All applicable control rods shall be tested by the implementation of these steps, and signoffs for the individual control rods will be on Attachment 1 or on Attachment 3, as applicable.

4.6. When performing control rod exercising with a known fuel failure, the reactor shall be operated within the guidance provided in NF-AB-440-1001, Fuel Conditioning for 3D Monicore Plants

4.7. When reactor power is reduced for this surveillance test to comply with NF-AB-440-1001, return to rated power may occur at an unrestricted (but reasonable) rate if all of the following are met:

- **No** change in control rod notch positions has been made.
- The duration of the reduced power operation is less than 4 hours.

4.8. If reactor power is reduced for this surveillance test to comply with NF-AB-440-1001 and the duration of reduced power operation is greater than 4 hours, the failed fuel envelope must be deconditioned per NF-AB-440-1001.

4.9. If reactor power is required to be reduced for this surveillance test to comply with NF-AB-440-1001 the control rod testing requiring reduced reactor power will be scheduled and communicated to Generation Dispatch.

4.10. When making adjustments to Directional Control Valve needle valve position, two wrenches are required for both the needle valve and lock nut to prevent inadvertent adjustment of the needle valve.

4.11. During Noble Metals injection, the plant prerequisite for core flow is greater than or equal to 85% flow and reactor power greater than or equal to 70%. Notify Chemistry Department if outside of these parameters during Noble Metals injection.

4.12. Test equipment may be substituted with approval of Engineering.

Initials

5.0 **PREREQUISITES**

5.1 **VERIFY** all pages of this procedure are included. OPE

5.2 **ALL** test personnel **REVIEW** procedure **AND RECORD**:

Printed Name	Signature	Initials
Operator P. Examination	Operator P. Examination	OPE
Independent E. Verifier	Independent E. Verifier	IEV
Shift M. Manager	Shift M. Manager	SMM
Chief R. Operator	Chief R. Operator	CRO

Initials

5.3. **OBTAIN** a stopwatch for control rod timing. OPE

5.4. **IF** rod stroke timing is to be performed,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

5.4.1. **DETERMINE** which Attachment 4, CRDM Stroke Time Trending, sheets need to be completed
AND RECORD these sheets in step 7.1.1 Remarks. OPE

5.4.2. **ENSURE** installed plant instrumentation is available for use:

- C12-R604, Drive Wtr Flow, 2CEC*PNL603 (2RDS FIY112) ..... ☑
- C12-R602, Drive Wtr Diff Press, 2CEC*PNL603 (2RDS PDIY114) ..... ☑ OPE

5.5. **VERIFY** Reactor Manual Control
AND Rod Position Indication systems are aligned for normal operation per N2-OP-96, Reactor Manual Control and Rod Position Indication System. OPE

5.6. **VERIFY** Reactor Power is above the Rod Worth Minimizer (RWM) Low Power Setpoint
AND at the level directed by the Reactor Engineer for implementation of this procedure. OPE

5.7. **VERIFY** Reactor Power is within limits stated in the Core Reactivity Control Book. OPE

5.8. **VERIFY** **no** control rod withdrawal blocks present. OPE

5.9. **RECORD AND VERIFY** C12-R602 (2RDS-PDIY114) Drive Water Differential Pressure is set at 250 to 265 psid:

Drive Water Differential Pressure: 260 psid OPE

5.10 **IF** a partially inserted control rod has potential to double-notch as determined from a review of U2ESL/CRDDL **OR** IRs,
THEN RUN predictors which model the potential condition,
OTHERWISE MARK this step N/A. OPE

5.11 **IF** predictor runs show PCMARG greater than 0.20
OR any thermal limit greater than 0.99,
THEN CONSULT with Reactor Engineering to develop appropriate action,
OTHERWISE MARK this step N/A. OPE

5.12 **OBTAIN** a Control Rod Position printout via OD-7
OR 3D-Monicore. OPE

5.13 **ENSURE** **no** other testing is in progress that affects this procedure. OPE

Initials

5.14 **IF** any control rods have their position substituted in the Rod Worth Minimizer,
THEN RECORD their number **AND** position below,
OTHERWISE MARK this step N/A:

CONTROL ROD NUMBER	POSITION

N/A

Initials

NOTE

- Those control rods listed on Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect will be tested at reduced power. See Precaution and Limitation 4.7.
- If operating with failed fuel and the location is unknown then all Control Rods are to be considered within the 3x3 array.
- Additional copies of Attachment 3 may be used.

N/A 5.15 **IF** performing control rod exercising with a known fuel failure,
AND Reactor Engineering recommends a power change,
THEN PERFORM the following,
OTHERWISE MARK this step N/A:

N/A 5.15.1. Using Attachment 3 page 1, **LIST** all control rods fully inserted, deactivated and declared inoperable to suppress a fuel defect.

N/A
N/A
IV

N/A 5.15.2. Using Attachment 3 page 2, **LIST** all control rods at position 48 within a 3 x 3 array of the suppression rod(s).

N/A
N/A
IV

N/A 5.15.3. Using Attachment 3 page 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect, **LIST** all control rod which are at a partially withdrawn position within a 3 x 3 array of the suppression rod(s).

N/A
N/A
IV

NOTE

The control rods remaining on Attachment 1 may be tested with the reactor at full rated power.

N/A 5.15.4. Using Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet,
MARK all control rods listed on Attachment 3 as N/A.

N/A
N/A
IV

Initials

NOTE

- Attachment 4 may be completed in whole, in part, or **not** at all.
- Only sheets for control rods to be timed must be completed.

N/A 5.16. **IF** stroke timing is to be performed,
THEN using Attachment 4, CRDM Stroke Time Trending,
MARK all control rods at position 00 as N/A on sheets to be performed,
OTHERWISE MARK this step N/A. N/A

N/A 5.17. **IF** failed fuel exists,
AND Reactor Engineering recommends a power change
THEN VERIFY Reactor Engineering has provided ReMa to establish power levels to meet test requirements of NF-AB-440-1001, Fuel Conditioning For 3D Monicore Plants,
OTHERWISE MARK this step N/A. N/A

PLANT IMPACT **1. CONTROL ROD INSERT BLOCKS AND CONTROL ROD WITHDRAW BLOCKS MAY OCCUR DUE TO THE IMPLEMENTATION OF THIS PROCEDURE.**

5.18. SM **PERFORM** the following:

- **REVIEW** the Plant Impact. ..... ☑
- **ACKNOWLEDGE** that temporary alterations are to be used. ..... ☑
- **INDICATE** permission to perform procedure. ..... ☑

SMM
SM

5.19. CRO **PERFORM** the following:

- **REVIEW** the Plant Impact. ..... ☑
- **ACKNOWLEDGE** that procedure is to be performed **AND** that temporary alterations are to be used. ..... ☑

CRO
CRO

5.20. **RECORD** start time and date: Now / Today OPE

6.0 PERFORMANCE

NOTE

- Section 6.2, Control Rod Exercise Test for Attachment 3 Control Rods (Failed Fuel) may be performed prior to section 6.1.
- For any reference to Unit 2 Control Rod Drive Deficiency Log (U2ESL/CRDDL) entries required by this procedure, they may be captured as a remark on the appropriate attachment and entered after testing is complete.

6.1. Control Rod Exercise Test

NOTE

- Control Rods can be tested in any order.
- More than one Attachment 1 can be used to clarify the intended order.
- For procedure placekeeping, per OP-AA-300, on Attachment 1 or Attachment 3 the Additional Qualified Individual will record data for Steps 6.1.3 through 6.1.4 and initial block 6.1.26.1.
- The performer then initials block 6.1.26.2 signifying that all data was recorded correctly.

6.1.1. **SELECT** control rod to be tested.

6.1.2. **VERIFY** there are **no** Rod Insert Blocks present.

6.1.3. **RECORD** control rod's initial position on Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet **OR** Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect. **[T/S] [IST]**

6.1.4. For control rods on Attachment 2,
RECORD computer indicated rod position on Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer).

NOTE

- For insert rod timing, the stopwatch should be started when the initial position extinguishes and stopped when the target position extinguishes.
- If the target position does **not** extinguish on the insert, the CRDM is too slow. This is to be noted on the Attachment 4 data sheet in the comments.
- Insert rod speed should be within 1.8 to 2.2 seconds, with an optimum 2.0 second target for the single notch insertion; which corresponds to a full stroke time of 43.2 to 52.8 seconds with a target value of 48 seconds.
- To have optimum performance, the insert time should be less than the withdraw time.
- Any double-notch events need to be documented in an IR.

6.1.5. **PERFORM** one of the following to insert control rod: **[IST][T/S]**

☐ **IF** control rod exhibits double-notching after drive flow has been adjusted as far as possible with 2RDS*SOV123 as indicated in U2ESL/CRDDL
AND rod will only be inserted one notch,
THEN INSERT rod per N2-OP-96, section H, Preventing Control Rod Insert Double Notching.

☐ **IF** performing in conjunction with Control Rod movements,
THEN MOVE control rod as directed by Reactor Engineer, **OR** by Reactor Engineering procedures.

☐ Using normal Drive Water differential pressure of 250 to 265 psid,
INSERT control rod one notch.

6.1.6. **IF** control rod inserts but does **not** reach next notch due to being slow,
THEN RE-PERFORM INSERT by
HOLDING INSERT pushbutton
UNTIL desired position is observed,
AND VERIFY U2ESL/CRDDL gets updated with an IR,
OTHERWISE MARK this step N/A.

6.1.7. **IF** stroke timing was performed,
THEN DOCUMENT CRDM insert stroke time on Attachment 4,
OTHERWISE MARK this step N/A. **[T/S]**

6.1.8. **IF** control rod double-notches
AND 2RDS*SOV123 is **not** full closed,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **ADJUST** drive water flow per N2-OP-96, section H, Preventing Control Rod Insert Double Notching. ..... ☐

2. **RE-PERFORM** control rod exercise test
AND DOCUMENT manipulations at end of Control Rod Movement and Position Indication Verification Data Sheet. ..... ☐

3. **VERIFY** U2ESL/CRDDL gets updated with IR number. . ☐

6.1.9. **IF** control rod does **not INSERT** with normal drive pressure,
THEN PERFORM the following
OTHERWISE MARK these steps N/A:

1. **PERFORM** N2-OP-30, section H, Failure to Insert/Withdraw Control Rod ..... ☐

2. **RECORD** failure to insert in attachment remarks. ..... ☐

3. **RECORD** failure to insert with IR number in U2ESL/CRDDL. ..... ☐

NOTE

Failure of a control rod to insert with normal Drive Water Differential Pressure may be an indication of a missing rod drive mechanism Cooling Water Orifice.

6.1.10. **IF** position indication is **not** operable at one notch inserted,
THEN IMMEDIATELY PERFORM the following
OTHERWISE MARK these steps N/A:

1. **NOTIFY** SM. ..... ☐

2. **CONSULT** Reactor Engineer for instructions as to inserting rod one additional notch
UNTIL position indication is obtained. ..... ☐

6.1.11. **RECORD** new control rod position on Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet **OR** Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect. **[T/S] [IST]**

6.1.12. For control rods on Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer), **RECORD** computer indicated rod position on Attachment 2.

6.1.13. **MONITOR** nuclear instrumentation response.

6.1.14. **CONFIRM** 4-Rod Display indicates proper rod position,
AND IF so,
THEN CHECK-OFF this step on Control Rod Movement and Position Indication Verification Data Sheet. **[T/S]**

NOTE

- "Almost double-notching" is described as rod speed observed to be too fast when compared to other normal rod movements or the rod moves too far past the intended notch but does **not** latch into the next notch.
- The intent of identifying these rods is that this rod has the potential to double-notch and, therefore, should be adjusted before it does.
- Adjustments for almost double-notching do **not** have to be immediately retested but should be entered into the U2ESL/CRDDL for observance during the next surveillance performance.

6.1.15. **IF** control rod withdraw almost double-notches,
THEN PERFORM the following:

1. **PERFORM** a copy of section 6.5 Adjustment of Almost Double-Notching Rods..... ☐
2. **NOTE** in Remarks section so that withdraw drive water flow can be adjusted later in section 6.3 Adjustment and Verification of Single Notch Timed Control Rods..... ☐

NOTE

Capturing Remarks in attachments and U2ESL/CRDDL with IR may be performed later.

3. **VERIFY** U2ESL/CRDDL has been updated with IR number. ..... ☐

NOTE

- When performing withdraw notch timing, perform withdrawal from 46 to 48 and coupling check in one manipulation to prevent erroneous data due to the rod settle function.
- For withdraw rod timing, the stopwatch should be started when the initial position is re-illuminated after the double dash appears AND stopped when the target position is illuminated.
- If control rod does **not** withdraw to the target position, then the CRDM is too slow.
- Withdraw rod speed should be within 1.8 to 2.2 seconds, with an optimum 2.0 second target for the single notch withdraw. This corresponds to a full-stroke time of 43.2 to 52.8 seconds with a target value of 48 seconds. To have optimum performance, the insert time should be less than the withdraw time. However, since the withdraw timer relay energizes the DCV 2RDS*SOV122 for only 1.5 seconds, the withdraw time only provides a trend information of CRDM speed.
- If control rod being tested initial position is 48, then coupling check per step 6.1.21 may be performed concurrently with step 6.1.16.
- Any double-notch events need to be documented in an IR.

CAUTION

Withdrawing a control rod past its initial position could cause fuel damage due to preconditioning violations.

6.1.16. **PERFORM** one of the following to withdraw control rod: **[T/S] [IST]**

☐ **IF** control rod exhibits double-notching after drive flow has been adjusted as far as possible with 2RDS*SOV120 as indicated in U2ESL/CRDDL
AND rod will only be withdrawn one notch,
THEN WITHDRAW rod per N2-OP-96, section H, Preventing Control Rod Withdraw Double Notching.

☐ **IF** performing in conjunction with control rod movements,
THEN MOVE control rod as directed by Reactor Engineer,
OR by Reactor Engineering procedures.

☐ Using normal Drive Water Differential Pressure of 250 to 265 psid,
WITHDRAW control rod to its pre-test position.

6.1.17. **IF** stroke timing was performed,
THEN DOCUMENT CRDM withdraw stroke time for select rods on Attachment 4.

6.1.18. **IF** control rod double-notches
AND 2RDS*SOV120 is **not** full closed,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **ADJUST** drive water flow per N2-OP-96, section H, Preventing Control Rod Withdraw Double Notching. ..... ☐

2. **RE-PERFORM** control rod exercise test
AND DOCUMENT manipulations at end of Control Rod Movement and Position Indication Verification Data Sheet. ..... ☐

3. **VERIFY** U2ESL/CRDDL gets updated with IR number. . ☐

NOTE

Capturing Remarks in attachments and U2ESL/CRDDL with IR may be performed later.

6.1.19. **IF** control rod will **not** withdraw with normal drive pressure of 250 to 265 psid,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **PERFORM** N2-OP-30, section H, Failure to Insert/Withdraw Control Rod ..... ☐

2. **RECORD** failure to withdraw in attachment remarks. ..... ☐

3. **RECORD** failure to withdraw with IR number in U2ESL/CRDDL. ..... ☐

NOTE

- "Almost double-notching" is described as rod speed observed to be too fast when compared to other normal rod movements or the rod moves too far past the intended notch but does **not** latch into the next notch.
- The intent of identifying these rods is that this rod has the potential to double-notch and, therefore, should be adjusted before it does.
- Adjustments for almost double-notching do **not** have to be immediately retested but should be entered into the U2ESL/CRDDL for observance during the next surveillance performance.

6.1.20. **IF** control rod withdraw almost double-notches,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **PERFORM** a copy of section 6.5 Adjustment of Almost Double-Notching Rods. ..... ☐

2. **NOTE** in Remarks section so that withdraw drive water flow can be adjusted later in section 6.3 Adjustment and Verification of Single Notch Timed Control Rods..... ☐

NOTE

Capturing Remarks in attachments and U2ESL/CRDDL with IR may be performed later.

3. **VERIFY** U2ESL/CRDDL has been updated with IR number. ..... ☐

NOTE

Step 6.1.21 shall be performed only for control rods that have final positions of 48.

6.1.21. **IF** Control Rod's final position is 48,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **ATTEMPT** to withdraw past position 48, by **DEPRESSING AND HOLDING BOTH** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons. **[T/S]** ..... ☐

2. **CONFIRM AND RECORD** red Full Out position light for control rod is lit. **[T/S]**..... ☐

3. **CONFIRM AND RECORD** four rod display shows position 48. ..... ☐

4. **CONFIRM AND RECORD** 603444 CONTROL ROD OVERTRAVEL does **not** alarm. **[T/S]**..... ☐

5. **IF** Control Rod Drive withdraw stall flow on C12-R604 (2RDS-FIY112) is less than or equal to 3.5 gpm, **THEN RECORD** on Attachment 1 **OR** Attachment 3. **[T/S]** ..... ☐

6. **RELEASE** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons. ..... ☐

6.1.21 (Continued)

NOTE

Procedure may continue with CRDDL entry made later.

7. **IF** Control Rod Drive withdraw stall flow on C12-R604 (2RDS-FIY112) is greater than 3.5 gpm
AND flushing is **not** waived by SM/US,
THEN PERFORM the following,
OTHERWISE MARK these step N/A:

   a. **DEPRESS AND HOLD**
   BOTH WITHDRAW
   AND CONTINUOUS WITHDRAW pushbuttons
   for one to five seconds to flush CRDM seals. ..... ☐

   b. **RELEASE** WITHDRAW
   AND CONTINUOUS WITHDRAW pushbuttons. ☐

   c. **REPEAT** steps 6.1.21.7.a
   AND 6.1.21.7.b up to 4 more times. ..... ☐ ☐ ☐ ☐

   d. **IF** Control Rod Drive withdraw stall flow on C12-R604 (2RDS-FIY112) is greater than 3.5 gpm
   AND showing a lowering trend,
   THEN REPEAT Steps 6.1.21.7.a
   AND 6.1.21.7.b up to five more times,
   OTHERWISE MARK this step N/A. ..... ☐ ☐ ☐ ☐ ☐

   e. **RECORD** the following:

      (1) Final withdraw stall flow in Attachment 1
      OR Attachment 3. ..... ☐

      (2) **H**igh withdraw stall flow condition in Remarks of Attachment 1
      OR Attachment 3. ..... ☐

      (3) Final withdraw stall flow in U2ESL/CRDDL. ..... ☐

6.1.21 (Continued)

8. **IF** stall flow is greater than 3.5 gpm
AND has degraded **OR** **not** improved with flushing performed greater than or equal to four tests,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

   a. **RECORD** observed withdraw stall flow on Attachment 1 **OR** Attachment 3. ..... ☐

   b. **RECORD** high withdraw stall flow condition in Remarks of Attachment 1 **OR** Attachment 3. ..... ☐

NOTE

Procedure may continue with CRDDL entry made later.

9. **IF** SM/US waives step 6.1.21.7 flushing,
THEN PEFORM the following,
OTHERWISE MARK these steps N/A:

   - **RECORD** on Attachment 1 **OR** Attachment 3:
     - High withdraw stall flow..... ☐
     - IR number..... ☐
     - Flushing waived by SM/US. ..... ☐
   - **RECORD** in U2ESL/CRDDL:
     - High withdraw stall flow..... ☐
     - IR number..... ☐
     - Flushing waived by SM/US. ..... ☐

NOTE

Successful completion of Step 6.1.22 satisfies the reverse flow exercise test of 2RDS*V138 for that rod.

6.1.22. **ATTEMPT** to insert each control rod that is fully inserted (Position 00) as follows:

NOTE

Indication should go from 00 to blank on the Four Rod Display.

1. **PUSH AND HOLD** depressed INSERT pushbutton on rod select matrix. ..... ☐

2. **CONFIRM AND RECORD** that 00 indication is lost. **[IST]** ..... ☐

3. **IF** Control Rod Drive insert stall flow on C12-R604 (2RDS-FIY112) is less than or equal to 3.5 gpm, **THEN RECORD** on Attachment 1 **OR** Attachment 3..... ☐

4. **RELEASE** INSERT pushbutton. ..... ☐

6.1.22 (Continued)

5. **IF** Control Rod Drive insert stall flow on C12-R604 (2RDS-FIY112) is greater than 3.5 gpm
AND flushing is **not** waived by SM/US,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

   a. **DEPRESS AND HOLD** INSERT pushbutton for one to five seconds to flush CRDM seals. ..... ☐

   b. **RELEASE** INSERT pushbutton. ..... ☐

   c. **REPEAT** Steps 6.1.22.5.a
   AND 6.1.22.5.b up to four more times. ..... ☐ ☐ ☐ ☐

   d. **IF** Control Rod Drive insert stall flow on C12-R604 (2RDS-FIY112) greater than 3.5 gpm **AND** showing a lowering trend,
   THEN repeat Steps 6.1.22.5.a
   AND 6.1.22.5.b up to five more times. ..... ☐ ☐ ☐ ☐ ☐

   e. **RECORD** the following:

      (1) Final withdraw stall flow in Attachment 1
      OR Attachment 3. ..... ☐

      (2) **H**igh withdraw stall flow condition in Remarks of Attachment 1
      OR Attachment 3. ..... ☐

      (3) Final withdraw stall flow in U2ESL/CRDDL. ..... ☐

   f. **IF** Control Rod Drive insert stall flow on C12-R604 (2RDS-FIY112) is less than or equal to 3.5 gpm,
   THEN RECORD in Attachment 1
   OR Attachment 3. ..... ☐

   g. **IF** insert stall flows are upscale following repeated insert flushing attempts,
   THEN GENERATE an IR to include this control rod in the next scheduled maneuver performing Full Stroke Insertion Testing with notch-to-notch timing and settle time testing, to evaluate if channel friction exists,
   OTHERWISE MARK this step N/A. ..... ☐

6.1.22 (Continued)

6. **IF** insert stall flow is greater than 3.5 gpm
AND has degraded
OR <u>not</u> IMPROVED with flushing performed greater than or equal to four tests,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

   a. **RECORD** observed insert stall flow on Attachment 1 **OR** Attachment 3. ..... ☐

   b. **ENTER** in Remarks of Attachment 1 **OR** Attachment 3 high insert stall flow condition. ☐

NOTE

Procedure may continue with CRDDL entry made later.

7. **IF** SM/US waived step 6.1.22.5 flushing,
THEN PEFORM the following,
OTHERWISE MARK these steps N/A:

   - **RECORD** on Attachment 1 **OR** Attachment 3:
     - High withdraw stall flow..... ☐
     - IR number..... ☐
     - Flushing waived by SM/US. ..... ☐
   - **RECORD** in U2ESL/CRDDL:
     - High withdraw stall flow..... ☐
     - IR number..... ☐
     - Flushing waived by SM/US. ..... ☐

NOTE

Recording rod position verifies position indication operability for prescribed rod.

6.1.23. **RECORD** final control rod position on Attachment 1 **OR** Attachment 3. **[T/S] [IST]**

6.1.24. For control rods on Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer), **RECORD** computer indicated rod position on Attachment 2.

6.1.25. **CONFIRM** final position is same as initial position,
THEN indicate results on Attachment 1 **OR** Attachment 3.

6.1.26. **PERFORM** the following for rod being tested:

1. Additional qualified individual initials that all data is accurate **AND** complete. ..... ☐

2. Performer initials that all data is accurate **AND** complete. ..... ☐

6.1.27. **REPEAT** Step 6.1.1 through 6.1.26 for each rod to be tested.

6.2. **Control Rod Exercise Test for Attachment 3 Control Rods (Failed Fuel)**

6.2.1. **IF no** failed fuel exists
OR Reactor Engineering does **not** recommend a power change,
THEN MARK all steps in this section N/A,
OTHERWISE MARK this step N/A. _____

NOTE

- Control rods can be tested in any order.
- More than one Attachment 3 can be used to clarify the intended order.
- This section is intended for conducting Exercise Testing for control rods when a failed fuel bundle has been identified by flux suppression testing and Reactor Engineering recommends a power reduction.
- On Attachment 3, the Additional Qualified Individual (per OP-AA-300) will record data for Steps 6.1.3 through 6.1.25 and initial block 6.1.26.1, the performer then initials block 6.1.26.2 signifying that all data was recorded correctly.
- Control rods **not** associated with flux suppression will be exercised using Attachment 1.

6.2.2. **VERIFY** reactor power is lowered to a value consistent with Precaution and Limitation 4.6. _____

6.2.3. **PERFORM** flux suppression control rod exercising per section 6.1 **AND DOCUMENT** step completion on Attachment 3. _____

6.3. **Adjustment and Verification of Single Notch Timed Control Rods**

NOTE

It is recognized that when using a stopwatch to measure 2.0 seconds, human reaction time may **not** be consistent and therefore judgment may be used to assure rod movement is satisfactory.

6.3.1. For all control rods that were single-notch timed
AND single-notch insert
OR withdraw time is **not** within 1.8 to 2.2 seconds,
COMPLETE section 6.6 Adjustment of Single-Notch Timed Control Rods,
OTHERWISE MARK this step N/A. _____

6.3.2. For all control rods that were single-notch stroke timed
AND were adjusted per section 6.6 Adjustment of Single-Notch Timed Control Rods,
VERIFY single-notch reliability in
BOTH insert **AND** withdraw directions,
OTHERWISE MARK these steps N/A:

1. **INSERT** control rod to be timed one notch from 48 to 46
AND CAPTURE timing data on Attachment 4..... ☐

NOTE

When performing withdraw notch timing, perform withdrawal from 46 to 48 and coupling check in one manipulation to prevent erroneous data due to the rod settle function.

2. **WITHDRAW** control rod to be timed one notch from 46 to 48
AND capture timing data on Attachment 4..... ☐ _____

6.3.3. **REPEAT** Steps 6.3.1 **AND** 6.3.2
UNTIL insert **AND** withdraw times are acceptable
OR UNTIL no Directional Control Valve adjustments are available. _____

6.4. **Adjustment of Almost Double-Notching Rods, Continuous IN, High Drive Pressure to Withdraw, or Double-Notching Rods**

6.4.1. **IF** rods have had section 6.5 Adjustment of Almost Double-Notching Rods copies initiated
OR have been noted in Remarks section to have almost double-notched,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **VERIFY** a copy of section 6.5 Adjustment of Almost Double-Notching Rods is completed for each rod that requires adjustment. ..... ☐

2. **ATTACH** copies of section 6.5 Adjustment of Almost Double-Notching Rods to this surveillance. ..... ☐

3. **VERIFY** U2ESL/CRDDL has been updated with IR number. ..... ☐ _____

6.5. **Adjustment of Almost Double-Notching Rods**

> **NOTE**
>
> Complete a copy of section 6.5 Adjustment of Almost Double-Notching Rods for each rod that requires adjustment.

6.5.1. **RECORD** Control Rod Number _____ _____

6.5.2. **CHECK** direction of almost double-notching control rod:

☐ Inserting

☐ Withdrawing _____

6.5.3. **IF** control rod almost double-notched when inserting,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **IF** U2ESL/CRDDL indicates control rod drive flow has been adjusted to point where **no** further adjustment is available (2RDS*SOV123 is full closed),
THEN EXIT this section by
MARKING all remaining steps N/A,
OTHERWISE MARK this step N/A. ..... ☐

2. At Hydraulic Control Unit (HCU),
REMOVE needle valve cap on 2RDS*SOV123..... ☐

3. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
LOOSEN locknut on 2RDS*SOV123 needle valve..... ☐

4. **USING** a wrench,
THREAD needle valve **IN** 1/4 of a turn..... ☐

5. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
TIGHTEN 2RDS*SOV123 needle valve locknut..... ☐

6. **REINSTALL** cap on 2RDS*SOV123
AND torque closed 100 to 115 in-lbs. ..... ☐

7. **IF** 1/4 turn adjustment resulted in full closure of 2RDS*SOV123,
THEN INITIATE an IR to replace CRDM,
OTHERWISE MARK this step N/A. ..... ☐ _____

6.5.4. **IF** control rod almost double-notched when withdrawing,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. **IF** U2ESL/CRDDL indicates control rod drive flow has been adjusted to the point where **no** further adjustment is available (2RDS*SOV120 is full closed),
THEN EXIT this section by
MARKING all remaining steps N/A,
OTHERWISE MARK this step N/A. ..... ☐

2. At Hydraulic Control Unit (HCU),
REMOVE needle valve cap on 2RDS*SOV120..... ☐

3. **WHILE MAINTAINING** needle value position with a wrench,
USE a wrench to
LOOSEN locknut on 2RDS*SOV120 needle valve..... ☐

4. **USING** a wrench,
THREAD needle valve **IN** 1/4 of a turn..... ☐

5. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to tighten 2RDS*SOV120 needle valve locknut. ..... ☐

6. **REINSTALL** cap on 2RDS*SOV120
AND torque closed 100 to 115 in-lbs. ..... ☐

7. **IF** 1/4 turn adjustment resulted in full closure of 2RDS*SOV120,
THEN INITIATE an IR to replace CRDM,
OTHERWISE MARK this step N/A. ..... ☐ _____

NOTE

If multiple section 6.5 Adjustment of Almost Double-Notching Rods are being performed, U2ESL/CRDDL entry may be made later.

6.5.5. **DOCUMENT** adjustments, attempted adjustments,
OR full closures of 2RDS*SOV123
OR 2RDS*SOV120 in U2ESL/CRDDL
AND step 7.1.1 Remarks. _____

6.6. Adjustment of Single-Notch Timed Control Rods

NOTE

Complete a copy of section 6.6 Adjustment of Single-Notch Timed Control Rods for each rod that requires adjustment.

6.6.1. **RECORD** Control Rod Number: _____ _____

6.6.2. **CHECK** direction of required adjustment:

☐ Inserting

☐ Withdrawing _____

6.6.3. **IF** U2ESL/CRDDL indicates control rod drive flow has been adjusted to point where **no** further adjustment is available (2RDS*SOV123 is full closed),
THEN EXIT this section by
MARKING all remaining steps N/A,
OTHERWISE MARK this step N/A. _____

6.6.4. **IF** control rod adjustment is required for insert direction,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. At Hydraulic Control Unit (HCU),
REMOVE needle valve cap on 2RDS*SOV123..... ☐

2. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
LOOSEN locknut on 2RDS*SOV123 needle valve..... ☐

3. **IF** timing is too fast,
THEN USE a wrench to
THREAD Needle Valve **IN** a maximum of 1/2 turn,
OTHERWISE MARK this step N/A. ..... ☐

4. **IF** timing is too slow,
THEN USE a wrench to
THREAD Needle Valve **OUT** a maximum of 1/4 turn,
OTHERWISE MARK this step N/A. ..... ☐

5. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
TIGHTEN 2RDS*SOV123 needle valve locknut..... ☐

6. **REINSTALL** cap on 2RDS*SOV123
AND TORQUE closed 100 to 115 in-lbs. ..... ☐

7. **IF** adjustment resulted in full closure of 2RDS*SOV123,
THEN INITIATE an IR to replace CRDM,
OTHERWISE MARK this step N/A. ..... ☐ _____

6.6.5. **IF** U2ESL/CRDDL indicates control rod drive flow has been adjusted to the point where **no** further adjustment is available (2RDS*SOV120 is full closed),
THEN EXIT THEN EXIT this section by
MARKING all remaining steps N/A,
OTHERWISE MARK this step N/A. _____

6.6.6. **IF** control rod adjustment is required for withdraw direction,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1. At Hydraulic Control Unit (HCU),
REMOVE needle valve cap on 2RDS*SOV120..... ☐

2. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
LOOSEN locknut on 2RDS*SOV120 needle valve..... ☐

3. **IF** timing is too fast,
THEN USE a wrench to
THREAD Needle Valve **IN** a maximum of 1/2 turn,
OTHERWISE MARK this step N/A. ..... ☐

4. **IF** timing is too slow,
THEN USE a wrench to
THREAD Needle Valve **OUT** a maximum of 1/4 turn,
OTHERWISE MARK this step N/A. ..... ☐

5. **WHILE MAINTAINING** needle valve position with a wrench,
USE a wrench to
TIGHTEN 2RDS*SOV120 needle valve locknut..... ☐

6. **REINSTALL** cap on 2RDS*SOV120
AND TORQUE closed 100 to 115 in-lbs. ..... ☐

7. **IF** adjustment resulted in full closure of 2RDS*SOV120,
THEN INITIATE an IR to replace CRDM,
OTHERWISE MARK this step N/A. ..... ☐ _____

NOTE

If multiple section 6.6 Adjustment of Single-Notch Timed Control Rods are being performed, entry in U2ESL/CRDDL may be done later.

6.6.7. **DOCUMENT** adjustments, attempted adjustments,
OR full closures of 2RDS*SOV123 **OR** 2RDS*SOV120 in U2ESL/CRDDL **AND** surveillance step 7.1.1 Remarks. _____

6.7. **Return to Normal**

6.7.1. **VERIFY** Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet, is complete,
AND Not Applicable signoffs are denoted with N/A. _____

6.7.2. **IF** performed,
THEN VERIFY Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect, is complete,
AND Not Applicable signoffs are denoted with N/A,
OTHERWISE MARK this step N/A. _____

6.7.3. **OBTAIN** a Control Rod Position printout,
AND PERFORM the following:

- Except for rods repositioned per ReMa,
  VERIFY all control rods are at their pre-test positions. ... ☐
- **IF** rods were repositioned per ReMa,
  THEN ATTACH ReMa,
  OTHERWISE MARK this step N/A. ..... ☐
- **IF** Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet was performed,
  THEN VERIFY positions recorded in steps 6.1.3, 6.1.11 **AND** 6.1.23 on Attachment 1, are same as those recorded in 6.1.4, 6.1.11 and 6.1.24 on Attachment 2,
  OTHERWISE MARK this step N/A. ..... ☐
- **IF** Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect was performed,
  THEN VERIFY positions recorded in steps 6.1.3, 6.1.11 **AND** 6.1.23 on Attachment 3, are same as those recorded in 6.1.4, 6.1.11 and 6.1.24 on Attachment 2,
  OTHERWISE MARK this step N/A. ..... ☐
- **ATTACH** Control Rod Position printouts to this procedure. ..... ☐

_____

6.7.4. **IF** rods were listed in step 5.14,
THEN SUBSTITUTE position in Rod Worth Minimizer for control rod(s) listed in Step 5.14 per N2-OP-95A,
OTHERWISE MARK this step N/A. _____

6.7.5. **VERIFY** applicable attachments are complete,
AND all abnormal results are clearly identified in step 7.1.1 Remarks. _____

6.7.6. **VERIFY** CRAM Rod Hard Card matches Core Reactivity Control Book CRAM Rods Listing. _____

6.7.7. **NOTIFY** SM and CRO that procedure is completed **AND** temporary alterations have been returned to normal. _____

6.7.8. **RECORD** test stop time and date: _____ _____

6.8. **Acceptance Criteria**

6.8.1. **VERIFY** all control rod movement verifications performed per requirements of Plant Technical Specification SR 3.1.3.2 **AND** SR 3.1.3.3 were successfully completed. _____

6.8.2. **VERIFY** all Control Rod Coupling Integrity Verifications performed per requirements of Technical Specification SR 3.1.3.5 were successfully completed. _____

6.8.3. **VERIFY** satisfactory control rod movement verification which proves reverse flow requirements for corresponding HCU Cooling Water check valves, 2RDS*V138 per IST criteria. _____

6.8.4. **VERIFY** satisfactory control rod movement verification which proves test requirements for corresponding HCU Drive Water Check Valves, 2RDS*V137 per IST criteria. _____

6.8.5. **VERIFY** Cooling Water Orifice Installation Integrity was verified by ensuring all control rods inserted, moved properly with normal Drive Water Differential Pressure of 250 to 265 psid. _____

7.0 POST-PERFORMANCE ACTIVITIES

7.1. SM Review

☐ Satisfactory, **no** corrective action required.

☐ Satisfactory, corrective action required,
PERFORM the following,
OTHERWISE MARK these steps N/A:

- **EXPLAIN** in step 7.1.1 Remarks as necessary..... ☐
- **INITIATE** an IR..... ☐

☐ Unsatisfactory,
PERFORM the following,
OTHERWISE MARK these steps N/A:

- **EXPLAIN** in step 7.1.1 Remarks necessary ..... ☐
- **INITIATE** an IR..... ☐
- **IMMEDIATELY NOTIFY** Shift Operations Superintendent or designee:..... ☐

_____
Name of Person Notified

7.1.1. Remarks: _____
_____
_____
_____
_____
_____
_____
_____
_____
_____
_____
_____
_____

_____ _____
Shift Manager Signature Date/Time

7.2. Second Operations Review

7.2.1. **VERIFY** all pages of this procedure are included. _____

_____ _____
Second Ops Review Signature Date

7.3. **Engineering Reviews**

7.3.1. **EMAIL** a completed procedure copy to IST Programs Engineer for review:

Sent to: _____ _____

8.0 BASES

None.

9.0 RECORDS

The records generated by this procedure are Quality Records and shall be processed per RM-AA-101, Records Management Programs.

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
18-59													
22-59													
26-59													
30-59													
34-59													
38-59													
42-59													
14-55													
18-55													
22-55													
26-55													
30-55													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
34-55													
38-55													
42-55													
46-55													
10-51													
14-51													
18-51													
22-51													
26-51													
30-51													
34-51													
38-51													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
42-51													
46-51													
50-51													
06-47													
10-47													
14-47													
18-47													
22-47													
26-47													
30-47													
34-47													
38-47													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
42-47													
46-47													
50-47													
54-47													
02-43													
06-43													
10-43													
14-43													
18-43													
22-43													
26-43													
30-43													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
34-43													
38-43													
42-43													
46-43													
50-43													
54-43													
58-43													
02-39													
06-39													
10-39													
14-39													
18-39													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
22-39													
26-39													
30-39													
34-39													
38-39													
42-39													
46-39													
50-39													
54-39													
58-39													
02-35													
06-35													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
10-35													
14-35													
18-35													
22-35													
26-35													
30-35													
34-35													
38-35													
42-35													
46-35													
50-35													
54-35													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
58-35													
02-31													
06-31													
10-31													
14-31													
18-31													
22-31													
26-31													
30-31													
34-31													
38-31													
42-31													

Remarks: _____

_____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
46-31													
50-31													
54-31													
58-31													
02-27													
06-27													
10-27													
14-27													
18-27													
22-27													
26-27													
30-27													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
34-27													
38-27													
42-27													
46-27													
50-27													
54-27													
58-27													
02-23													
06-23													
10-23													
14-23													
18-23													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
22-23													
26-23													
30-23													
34-23													
38-23													
42-23													
46-23													
50-23													
54-23													
58-23													
02-19													
06-19													

Remarks: _____

_____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
10-19													
14-19													
18-19													
22-19													
26-19													
30-19													
34-19													
38-19													
42-19													
46-19													
50-19													
54-19													

Remarks: _____

_____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
58-19													
06-15													
10-15													
14-15													
18-15													
22-15													
26-15													
30-15													
34-15													
38-15													
42-15													
46-15													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
50-15													
54-15													
10-11													
14-11													
18-11													
22-11													
26-11													
30-11													
34-11													
38-11													
42-11													
46-11													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
50-11													
14-07													
18-07													
22-07													
26-07													
30-07													
34-07													
38-07													
42-07													
46-07													
18-03													
22-03													

Remarks: _____

Attachment 1, Control Rod Movement and Position Indication Verification Data Sheet (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
26-03													
30-03													
34-03													
38-03													
42-03													

Remarks: _____

_____

Attachment 2, Control Rod Movement and Position Indication Verification Data Sheet (Computer)

NOTE

- The following demonstrates that the Process Computer control rod position matches the control rod position indicated by the four-rod display for a sampling of control rods.
- Verification of all rods is **not** required.

STEP NUMBER	6.1.4	6.1.11	6.1.24
CONTROL ROD NUMBER	COMPUTER INITIAL ROD POSITION	COMPUTER INSERT POSITION	COMPUTER FINAL ROD POSITION
18-59			
22-59			
26-59			
30-59			
34-59			
38-59			
42-59			
14-55			
18-55			
22-55			
26-55			
30-55			

Initials

_____

Remarks: _____

_____

_____

_____

Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			
	00	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	00			

Remarks: _____

Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials

Remarks: _____

_____

Attachment 3, Control Rod Movement and Position Indication Verification Data Sheet with Suppressed Fuel Defect (Continued)

NOTE

De-select Control Rod prior to changing light bulbs.

Step Number	6.1.3 **[T/S] [IST]**	6.1.11 **[T/S] [IST]**	6.1.14 **[T/S]**	6.1.21.2 **[T/S]**	6.1.21.3 **[T/S]**	6.1.21.4 **[T/S]**	6.1.21.5 or 6.1.21.8 or 6.1.21.7 **[T/S]**	6.1.22.2 **[IST]**	6.1.22.3 or 6.1.22.6 or 6.1.22.5 **[T/S]**	6.1.23 **[T/S] [IST]**	6.1.25	6.1.26.1	6.1.26.2
Control Rod Number	Initial Rod Position	Insert Rod Position	Position Indication SAT (✓)	Full Out Light Lit (✓)	Position 48 Indication SAT (✓)	**No** Over-travel (✓)	Withdraw Stall Flow (gpm)	Insert Beyond 00 (✓)	Insert Stall Flow (gpm)	Final Rod Position	Final Position is Same as Initial (✓)	Additional Qualified Individual Initials	Performer Initials
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				
				N/A	N/A	N/A	N/A	N/A	N/A				

Remarks: _____

Attachment 4, CRDM Stroke Time Trending

NOTE

- It is recognized that when using a stopwatch to measure 2.0 seconds, human reaction times may **not** be consistent and therefore judgment may be used to assure rod movement is satisfactory.
- For insert rod timing, the stopwatch should be started when the initial position extinguishes and stopped when the target position extinguishes.
- For withdraw rod timing, the stopwatch should be started when the initial position is re-illuminated after the double dash appears and stopped when the target position is illuminated.
- Rod speed should be within 1.8 to 2.2 seconds, with an optimum 2.0 second target for the single notch movement. To have optimum performance, the insert time should be less than the withdraw time.

Control Rod Number: _____

STEP NUMBER	6.1.7, 6.1.17, 6.3.2.1, 6.3.2.2	
Movement Positions	Notch Time (Sec)	Drive Water DP (psid)

Remarks:

_____

_____

_____

_____

Training ID: **Nov 2023 NRC JPM D** Revision: **0.0**

Title: **Depressurize the Reactor Using Turbine Bypass Valves, Main Steam Leak in Turbine Building**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mike Carno	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-SOP-101C, Reactor Scram
2. N2-ARP-602200, 2CEC*PNL602 Series 200 Alarm Response Procedures
3. NUREG 1123 K/A 241000 A4.06 (4.2)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to depressurize the Reactor using Turbine Bypass Valves following a scram. The Operator will use N2-SOP-101C Detail 3 to lower Reactor pressure to 800 psig.

   b. This JPM is considered alternate path because, during the depressurization, a Main Steam Leak will develop into the Turbine Building and MSIVs will fail to automatically close. The Operator will take manual action to close MSIVs to isolate the steam leak.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-SOP-101C-01001, Perform or Respond to SOP-101C, Reactor Scram

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☒ Yes ☐ No			
Safety Function:	3		Reactor Pressure Control	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	241000 Reactor/Turbine Pressure Regulating System A4.06 - Ability to manually operate and/or monitor in the control room: Bypass valves			
K/A Importance Rating:	RO	4.2	SRO	-

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) **MS01**, Steam Leakage Outside Primary Containment, Final = 50 **TRG1**

b) **MS13**, MSIV Isolation Failure, Final = True **Inserted**

2) Remotes

a) None

3) Overrides

a) None

4) Annunciators

a) None

5) Event Triggers

Event #	Event Action	Command
1	rrpdome<920	Left Blank

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-SOP-101C Detail 3 and provide to the Operator.

8) Miscellaneous

a) IC-205 (For ILT 22-2, paired with JPM B)

-OR-

b) IC setup:

(1) The Reactor is scrammed

(2) Reactor pressure is approximately 940 psig on Turbine Bypass Valves

(3) Verify proper DW/SC pressures

(4) Verify HMI is **NOT** selected to PRESSURE CONTROL screen (5532) (including between IC resets)

6. Strategy Code

a. None

7. Tools and Equipment

    a. None

8. Commitments

    a. None

9. Prerequisites

    a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • A Reactor scram has occurred. • Reactor pressure is approximately 940 psig and stable on Turbine Bypass Valves. • N2-SOP-101C, Reactor Scram, is being executed. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *lower Reactor pressure to 800 psig using N2-SOP-101C Detail 3, Adjusting Pressure Set and Establishing Automatic Cooldown. Use a COOLDOWN TEMPERATURE RATE of 75°F/hr.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	• The AUTO RATE button is selectable when the difference between PRESSURE SET and PAM PRESSURE is less than 5 psig or when PRESSURE SET is less than PAM PRESSURE. • To exit automatic cooldown, select MANUAL RATE. • The AUTO RATE or MANUAL RATE button will be highlighted in blue indicating it is controlling and was the last method used for plant heat up or cooldown.		
2.	Navigate to the PRESSURE CONTROL screen (5532)	P (1)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL851, pushes PRESSURE CONTROL button on HMI.* ***Failure = Does not achieve view of PRESSURE CONTROL screen (5532)***

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.	Select MANUAL RATE (unless already highlighted blue)	P (2)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes MANUAL RATE button on HMI.*
4.	Select 1.5 PSIG/SEC	P (3)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes 1.5 PSIG/SEC button on HMI.*
5.	Enter a PRESSURE SETPOINT value to match Reactor Pressure (PAM PRESSURE)	P (4)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, enters a PRESSURE SETPOINT value to match Reactor pressure (approximately 940 psig) on HMI.*
6.	Select ENTER **AND** verify PRESS SET value at desired value	P (5)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes ENTER button on HMI. Observes PRESS SET value is at desired value.*
7.	Select GO	P (6)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes GO button on HMI.*
8.	Select AUTO RATE **WHEN** PAM PRESSURE AND PRESS SET are within 5 psig of each other AND AUTO RATE button is enabled (NOT gray)	P (7)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes AUTO RATE button on HMI.*
9.	Enter desired COOLDOWN TEMPERATURE RATE	P (8)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, enters COOLDOWN TEMPERATURE RATE of 75°F/hr on HMI.*
10.	Select ENTER	P (9)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes ENTER button on HMI.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
11.	Select CONFIRM **AND** verify RATE is correct	P (10)	SAT / UNSAT ***STD:*** *At 2CEC*PNL851, pushes CONFIRM button on HMI. Observes RATE is 75°F/hr*
12.	Enter a target PRESSURE SETPOINT value	P (11)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL851, enters target PRESSURE SETPOINT value of 800 psig on HMI.* ***Failure = PRESSURE SETPOINT value of 800 psig not achieved.***
13.	Select ENTER **AND** verify PRESSURE CONTROL TARGET is correct	P (12)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL851, pushes ENTER button on HMI. Observes PRESSURE CONTROL TARGET is 800 psig.* ***Failure = PRESSURE SETPOINT value of 800 psig not achieved.***
14.	Select GO **AND** verify PRESS SET lowering	P (13)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL851, pushes GO button on HMI.* ***Failure = Lowering Reactor pressure not achieved.***
Alternate Path:	When Reactor pressure lowers below approximately 920 psig, a leak will develop from Main Steam into the Turbine Building. The MSIVs will fail to automatically close. The Operator will isolate the leak by closing at least one MSIV in each Main Steam Line.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
15.	Recognize Main Steam leak into Turbine Building **Evaluator Note:** The Operator may execute ARPs 602228, 602233, and/or 602234. Alternatively, given the failure of an automatic action to occur, the Operator may close MSIVs without reference to a procedure as a backup to the failed automatic action. **Role Play:** If dispatched to investigate in Turbine Building, report that there is a large amount of steam in the Turbine Building and you are exiting the building.	P	SAT / UNSAT **STD:** Recognizes Main Steam leak into Turbine Building
16.	Isolate Main Steam leak by closing MSIVs **Evaluator Note:** It is only critical to close enough MSIVs to isolate the steam leak (ie. one MSIV in each steam line).	P	***PASS / FAIL** **STD:** *At 2CEC*PNL602, rotates each MSIV control switch (at least 4, up to 8) counter clockwise from the AUTO position to the CLOSE position. Observes the following for each MSIV:* • *PILOT SOV A ENERGIZE light not lit* • *PILOT SOV B ENERGIZE light not lit* • *GREEN light lit* • *RED light not lit* ***Failure = Main Steam leak into Turbine Building NOT isolated (ie any Main Steam Line still has both MSIVs NOT fully closed).***
Evaluator Note:	Once the steam leak has been isolated by closure of MSIVs, provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*A Reactor depressurization has been commenced using Turbine Bypass Valves by adjusting pressure set and establishing automatic cooldown. A steam leak into the Turbine Building has been isolated by closing MSIVs.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • A Reactor scram has occurred. • Reactor pressure is approximately 940 psig and stable on Turbine Bypass Valves. • N2-SOP-101C, Reactor Scram, is being executed.

INITIATING CUE	**(Operator Name)**, *lower Reactor pressure to 800 psig using N2-SOP-101C Detail 3, Adjusting Pressure Set and Establishing Automatic Cooldown. Use a COOLDOWN TEMPERATURE RATE of 75°F/hr.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • A Reactor scram has occurred. • Reactor pressure is approximately 940 psig and stable on Turbine Bypass Valves. • N2-SOP-101C, Reactor Scram, is being executed. **Evaluator:** *Ask applicant if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Applicant Name)**, *lower Reactor pressure to 800 psig using N2-SOP-101C Detail 3, Adjusting Pressure Set and Establishing Automatic Cooldown. Use a COOLDOWN TEMPERATURE RATE of 75°F/hr.*

PRESSURE CONTROL DETAILS

NOTE

- HOLD/GO/CANCEL buttons may be selected at any time during pressure adjustments to pause/resume/cancel the pressure change (for example, to stabilize the plant, allow for control rod insertion, etc.).
- Turbine must be tripped prior to initiating pressure control with Turbine Bypass Valves.
- During normal pressure reduction or cooldown following a SCRAM, Detail 3 should be used.

Detail 3: Adjusting Pressure Set and Establishing Automatic Cooldown

NOTE

- The AUTO RATE button is selectable when the difference between PRESSURE SET and PAM PRESSURE is less than 5 psig or when PRESSURE SET is less than PAM PRESSURE.
- To exit automatic cooldown, select MANUAL RATE.
- The AUTO RATE or MANUAL RATE button will be highlighted in blue indicating it is controlling and was the last method used for plant heat up or cooldown.

1. **NAVIGATE** to the PRESSURE CONTROL screen (5532).
2. **SELECT** MANUAL RATE (unless already highlighted blue).
3. **SELECT** 1.5 PSIG/SEC.
4. **ENTER** a PRESSURE SETPOINT value to match Reactor Pressure (PAM PRESSURE).
5. **SELECT** ENTER
   AND VERIFY PRESS SET value at desired value.
6. **SELECT** GO.
7. **WHEN** PAM PRESSURE
   AND PRES SET are within 5 psig of each other
   AND AUTO RATE button is enabled (**not** gray),
   THEN SELECT AUTO RATE.
8. **ENTER** desired COOLDOWN TEMPERATURE RATE.
9. **SELECT** ENTER.
10. **SELECT** CONFIRM
    AND VERIFY RATE is correct.
11. **ENTER** a target PRESSURE SETPOINT value.
12. **SELECT** ENTER
    AND VERIFY PRESSURE CONTROL TARGET is correct.
13. **SELECT** GO **AND VERIFY** PRESS SET lowering.
14. To change COOLDOWN TEMPERATURE RATE, **PERFORM** the following:
    a. **ENTER** desired COOLDOWN TEMPERATURE RATE.
    b. **SELECT** ENTER.
    c. **SELECT** CONFIRM **AND VERIFY** RATE is correct.
    d. **VERIFY** new rate is processing
    AND BPVs are controlling.

Detail 4: Opening all TBVs

NOTE

Press HOLD / CANCEL button as required to stop depressurization.

1. **NAVIGATE** to the BPV Jack screen (5537).
2. **SELECT** FULL OPEN.
3. **SELECT** CONFIRM.
4. **VERIFY** TBVs open.

Detail 5: Manual Pressure Control with TBV Jack

NOTE

A TBV full open is about 20% demand.

1. **NAVIGATE** to BPV Jack screen (5537).
2. **DETERMINE** approximate TBV position desired.
3. **ENTER** Target (0-100%).
4. **ENTER** Rate (.01-10%).
5. **SELECT** GO.
6. **VERIFY** TBVs open to desired position.
7. **REPEAT** as needed to adjust TBV position.

Detail 6: Select 550 psig Control

NOTE

Press HOLD / CANCEL button as required to stop depressurization.

1. **NAVIGATE** to PRESSURE CONTROL screen (5532).
2. **SELECT** 550 psig.
3. **SELECT** CONFIRM.
4. **VERIFY** Pressure Set Lowering.
5. **VERIFY** TBV response.

Detail 7: Adjusting Pressure Set and Establishing Manual Depressurization Rate

NOTE

- Press HOLD / CANCEL button as required to stop depressurization.
- Target rate may be adjusted while pressure set is lowering.

1. **NAVIGATE** to the PRESSURE CONTROL screen (5532).
2. **SELECT** MANUAL RATE.
3. **SELECT** 1.5 PSIG/SEC
   OR ENTER TARGET RATE (0.1 to 1.5 psig/sec).
4. **VERIFY** TARGET RATE correct.
5. **ENTER** PRESSURE SETPOINT
   AND VERIFY target value correct.
6. **SELECT** GO
7. **VERIFY** PRESS SET lowering

Training ID: **Nov 2023 NRC JPM E** Revision: **0.0**

Title: **Transfer House Loads from Normal Station Service to the Reserve Station Transformers**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mike Carno	7/13/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 10 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-71A, 13.8KV AC Power Distribution
2. NUREG 1123 K/A 262001 A4.01 (3.7)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to Transfer House Loads from Normal Station Service to the Reserve Station Transformers in accordance with N2-OP-71A, Section F.3.3.

   b. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-262001-01008, Transfer Station Service From Normal to Reserve

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☒ Perform ☐ Simulate			
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
Safety Function:	6		Electrical	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)	3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	262001 AC Electrical Distribution A4.01 - Ability to manually operate and/or monitor in the control room: Breakers and disconnects			
K/A Importance Rating:	RO	3.7	SRO	-

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021 or equivalent

b. Presets / With Triggers

1) Malfunctions

a) None

2) Remotes

a) None

3) Overrides

a) None

4) Annunciators

a) None

5) Event Triggers

Event #	Event Action	Command
None	N/A	N/A

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-OP-71A with steps F.3.1, F.3.2, and F.3.3.a placekept as complete. Next step to be performed is F.3.3.b.

8) Miscellaneous

a) IC-206 (For ILT 22-2, paired with JPM C)

-OR-

a) IC setup:

(1) House Loads being supplied by the Normal Station Transformer.

(2) Ensure that MWe has been reduced to < 1343 MWe by reducing reactor power. (P&L 7.0)

6. Strategy Code

   a. None

7. Tools and Equipment

   a. None

8. Commitments

   a. None

9. Prerequisites

   a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The Main Generator is on line and the load is stable. • All notifications required by OP-NM-108-107-1002 have been made. • All lock out relays and relay flags are reset in the relay room CB 288'. • 2RTX-SW001 Neutral Switch is open. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *transfer house loads to the Reserve Station Transformers in accordance with N2-OP-71A Sections F.3.3 and F.3.5. The procedure is in progress up to step F.3.3.b.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue. **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used*
2.	Places SYNCHRONIZE RESERVE A Sync Switch across Breaker 1-1 to NPS 13.8KV Bus 001 in ON.	P (F.3.3.b)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, places the sync key switch in the SYNCHRONIZE RESERVE A slot and rotates SYNCHRONIZE RESERVE A synch switch clockwise to the ON position.* ***Failure =*** *SYNCHRONIZE RESERVE A synch switch in the ON position not achieved.*
3.	Verifies frequency is in sync as indicated by a stable SYNCHROSCOPE indicator.	P (F.3.3.c)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852 visually observes SYNCHROSCOPE indicating needle is stable.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
4.	Observes voltage indication on 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter.	P (F.3.3.d)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, visually observes transformer and bus voltages are matched as read on 13.8 KV BUS NPS 001/003 INCOMING VOLTS meter and 13.8 KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter.*
Evaluator Note:	The following 2 steps may be required to be performed in order to match voltages.		
5.	IF 2RTX-XSR1A Transformer Voltage needs to be adjusted, performs the following: • Places RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT Switch in MAN.	P (F.3.3.e.1)	SAT / UNSAT / NA ***STD:*** *At 2CEC*PNL852, rotates RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT switch counter clockwise to the MAN position..*
6.	• Raises OR lowers Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1A LOAD TAP CHANGER Control Switch.	P (F.3.3.e.2)	SAT / UNSAT / NA ***STD:*** *At 2CEC*PNL852, adjusts RES STA SVCE XFMR 1A LOAD TAP CHANGER control switch to modify transformer voltage to match transformer and bus voltages.*
7.	Observes amps indication on RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT ammeters.	P (F.3.3.f)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes transformer amps at zero as read on RES STA SVCE XFMR 1A PRIM CURRENT phase 1, 2 and 3 meters.*
Procedure Note:	Indication of load transferred to 2RTX-XSR1A by observation of a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT ammeters may not be evident depending on incoming and running voltages and on bus loading.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
8.	Closes Breaker 1-1.	P (F.3.3.g)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, rotates control switch for BREAKER 1-1 clockwise to the NORMAL-AFTER-START position and observes RED light lit and green light not lit.* ***Failure =*** *BREAKER 1-1 RED light ON and GREEN light OFF not achieved.*
9.	Places SYNCHRONIZE RESERVE A Synch Switch across BREAKER 1-1 to NPS 13.8KV Bus 001 in OFF.	P (F.3.3.h)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, rotates SYNCHRONIZE RESERVE A synch switch counter clockwise to the OFF position.*
10.	Opens BREAKER 1-3, 2STX-XNS1 Normal Station Service Transformer Supply to NPS 13.8KV Bus 001.	P (F.3.3.i)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, rotates control switch for BREAKER 1-3 counter clockwise to the NORMAL-AFTER-STOP position and observes GREEN light lit and RED light not lit.* ***Failure =*** *BREAKER 1-3 GREEN light ON and RED light OFF not achieved.*
11.	Observes load transferred to 2RTX-XSR1A transformer as indicated by a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT Ammeters.	P (F.3.3.j)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes transformer amps at zero as read on RES STA SVCE XFMR 1A PRIM CURRENT phase 1, 2, and 3 meters.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
12.	Verifies RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT Switch in AUTO.	P (F.3.3.k)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes/places RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT switch in the AUTO position.*
13.	Observes voltage indication at 13.8KV NORM BUS NPS 001 KILOVOLTS Voltmeter.	P (F.3.3.l)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, visually observes voltage to be approximately 13.8 KV as read on 13.8 KV NORM BUS NPS 001 KILOVOLTS meter.*
14.	Section F.3.5 of OP-71A referenced. Transfer 2NPS-SWG003 to 2RTX-XSR1B.	P (F.3.5)	SAT / UNSAT ***STD:*** *Step referenced IAW Procedure Use and Adherence Requirements.*
15.	Places SYNCHRONIZE RES STA SVCE XFMR 1B Synch Switch across Breaker 3-1 to NPS 13.8KV Bus 003 in ON.	P (F.3.5.a)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, places the sync key switch in the SYNCHRONIZE RES STA SVCE XFMR 1B slot and rotates SYNCHRONIZE RES STA SVCE XFMR 1B synch switch clockwise to the ON position.* ***Failure =*** *SYNCHRONIZE RES STA SVCE XFMR 1B synch switch in the ON position not achieved.*
16.	Verify frequency is in sync as indicated by a stable SYNCHROSCOPE indicator.	P (F.3.5.b)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852 visually observes SYNCHROSCOPE indicating needle is stable.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
17.	Observe voltage indication of 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter.	P (F.3.5.c)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, visually observes transformer and bus voltages are matched as read on 13.8 KV BUS NPS 001/003 INCOMING VOLTS meter and 13.8 KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter.*
Evaluator Note:	The following 2 steps may be required to be performed in order to match voltages.		
18.	IF 2RTX-XSR1B Transformer Voltage needs to be adjusted, performs the following: • Places RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in MAN.	P (F.3.5.d.1)	SAT / UNSAT / NA ***STD:*** *At 2CEC*PNL852, rotates RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT switch counter clockwise to the MAN position..*
19.	• Raises OR lowers Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1B LOAD TAP CHANGER Control Switch.	P (F.3.5.d.2)	SAT / UNSAT / NA ***STD:*** *At 2CEC*PNL852, adjusts RES STA SVCE XFMR 1B LOAD TAP CHANGER control switch to modify transformer voltage to match transformer and bus voltages.*
20.	Observes amps indication on RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters.	P (F.3.5.e)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes transformer amps at zero as read on RES STA SVCE XFMR 1B PRIM CURRENT phase 1, 2 and 3 meters.*
Procedure Note:	Indication of load transferred to 2RTX-XSR1B by observation of a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT ammeters may not be evident depending on incoming and running voltages and on bus loading.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
21.	Closes Breaker 3-1.	P (F.3.5.f)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, rotates control switch for BREAKER 3-1 clockwise to the NORMAL-AFTER-START position and observes RED light lit and green light not lit.* ***Failure =*** *BREAKER 3-1 RED light ON and GREEN light OFF not achieved.*
22.	Places SYNCHRONIZE RES STA SVCE XFMR 1B Synch Switch across BREAKER 3-1 to NPS 13.8KV BUS 003 in OFF.	P (F.3.5.g)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, rotates SYNCHRONIZE RES STA SVCE XFMR 1B synch switch counter clockwise to the OFF position.*
23.	Opens Breaker 3-14, 2STX-XNS1 Normal Station Service Transformer Supply to NPS 13.8KV BUS 003.	P (F.3.5.h)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL852, rotates control switch for BREAKER 3-14 counter clockwise to the NORMAL-AFTER-STOP position and observes GREEN light lit and RED light not lit.* ***Failure =*** *BREAKER 3-14 GREEN light ON and RED light OFF not achieved.*
24.	Observes load transferred to 2RTX-XSR1B transformer as indicated by a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters.	P (F.3.5.i)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes transformer amps at zero as read on RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT phase 1, 2, and 3 meters.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
25.	Verifies RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in AUTO.	P (F.3.5.j)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, observes/places RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT switch in the AUTO position.*
26.	Observes voltage indication at 13.8KV NORM BUS NPS 003 KILOVOLTS Voltmeter.	P (F.3.5.k)	SAT / UNSAT ***STD:*** *At 2CEC*PNL852, visually observes voltage to be approximately 13.8 KV as read on 13.8 KV NORM BUS NPS 003 KILOVOLTS meter.*
Evaluator Note:	Once voltage indication at 13.8KV NORM BUS NPS 003 KILOVOLTS Voltmeter has been observed, then provide the following cue: **Cue:** *Your task is complete. Another operator will complete any remaining actions.*		

TASK STANDARD	*House loads have been transferred to the Reserve Station Transformers in accordance with N2-OP-71A Sections F.3.3 and F.3.5.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The Main Generator is on line and the load is stable. • All notifications required by OP-NM-108-107-1002 have been made. • All lock out relays and relay flags are reset in the relay room CB 288'. • 2RTX-SW001 Neutral Switch is open.

INITIATING CUE	**(Operator Name)**, *transfer house loads to the Reserve Station Transformers in accordance with N2-OP-71A Sections F.3.3 and F.3.5. The procedure is in progress up to step F.3.3.b.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The Main Generator is on line and the load is stable. • All notifications required by OP-NM-108-107-1002 have been made. • All lock out relays and relay flags are reset in the relay room CB 288'. • 2RTX-SW001 Neutral Switch is open. **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *transfer house loads to the Reserve Station Transformers in accordance with N2-OP-71A Sections F.3.3 and F.3.5. The procedure is in progress up to step F.3.3.b.*

Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-71A

13.8KV AC POWER DISTRIBUTION

Revision 01000

This procedure requires a 10 CFR 50.59 Review

Safety Related

CONTINUOUS USE

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
010	00	Minor Revision to incorporate:

PCR-19-00831:

- Editorially changed a bullet from step to it's own numbered step for improved Human Performance visibility, added linked procedure number for reference.
- Reason: Bullet lacked visibility and link to procedure to restore TSC Ventilation.
- H.14.5.c created, previously was 5th bullet of Step 14.5.b.
- H.14.5.c revised to add reference to N1-OP-66 and text for clarification of TSC Ventilation response.

PCR-20-00997:

- Allow breakers to be installed in 2NPS-SWG001-16 and 2NPS-SWG003-16.
- Reason: ECP-19-000363, ESR-16-000017 - Install Breakers in Alternate Feed Cubicles for 2NPS-SWG001 and 2NPS-SWG003.
- A.5.2, added ECP-19-000363.
- D.5 and D.6, revised to include provisions required by ECP-19-000363 and 10CFR50.59 Screen 5059-2019-124.
- H.1 through H.8 revised to reflect maintaining the Alternate Supply to NPS Bus 001 and 003 RACKED IN and OPEN, with CLOSING FUSES removed, and CONTROL SWITCH in Pull-to-Lock.

Writer Review:

- H.1 through H.8, changed instances of "as quickly as possible" to "efficiently".
- H.5 through H.8, placed new note to permit preparatory steps to be performed concurrently.
- H.5 through H.8, moved LCO checks, review of scheduled work and Diesel operability determinations to beginning of section from inappropriate previous revision placement within "as quickly as possible" steps.
- Applied Exelon Operating Procedure Template, moved procedure information from page footer to header, no revision bars used.
- Updated N/A statements throughout to comply with AD-NM-101-1005 5.4.C.13 by removing N/A check-boxes, made into IF/THEN statements and "OTHERWISE MARK this step N/A". No revision bars used.
- Updated CAPITALIZATION and bold font for action verbs, logic words, and "**not**" per AD-NM-101-1005 and AD-AA-101-1002.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 5
B. SYSTEM DESCRIPTION 9
C. OPERATING REQUIREMENTS 10
D. PRECAUTIONS AND LIMITATIONS 11
E. STARTUP 12
E.1. Placing 2NPS-SWG001 In Service 12
E.2. Placing 2NPS-SWG003 In Service 14
E.3. Placing 2NPS-SWG002 in Service 16
E.4. Placing 2ATX-XS1 in Service 18
E.5. Placing 2ATX-XS3 in Service 19
F. NORMAL OPERATIONS 20
F.1. Preparing 2STX-XNS1, Normal Station Service Transformer for Service 20
F.2. Transfer Station Service from Reserve to Normal 22
F.3. Transfer Station Service from Normal to Reserve 25
F.4. Security Building Switchgear Power Transfer 32
F.5. Placing Station Service Transformer Fan/Coolers 3 and 4 into Service and Fan/Coolers 1 and 2 to Standby 34
F.6. Placing Station Service Transformer Fan/Coolers 1 and 2 into Service and Fan/Coolers 3 and 4 to Standby 35
F.7. Placing 2ATX-XS1 Fans from Automatic to Manual Operation 36
F.8. Placing 2ATX-XS1 Fans from Manual to Automatic Operation 36
F.9. Placing 2ATX-XS3 Fans from Automatic to Manual Operation 37
F.10. Placing 2ATX-XS3 Fans from Manual to Automatic Operation 37
G. SHUTDOWN 38
H. OFF-NORMAL PROCEDURES 39
H.1. Transfer 2NPS-SWG001 Available Off-Site Feed from 2RTX-XSR1A to 2RTX-XSR1B with Main Generator On-Line 39
H.2. Transfer 2NPS-SWG001 Available Off-Site Feed from 2RTX-XSR1B to 2RTX-XSR1A with Main Generator On-Line 42
H.3. Transfer 2NPS-SWG003 Available Off-Site Feed from 2RTX-XSR1B to 2RTX-XSR1A with Main Generator On-Line 45
H.4. Transfer 2NPS-SWG003 Available Off-Site Feed from 2RTX-XSR1A to 2RTX-XSR1B with Main Generator On-Line 49
H.5. Transfer 2NPS-SWG001 Off-Site Feed from 2RTX-XSR1A to 2RTX-XSR1B with Main Generator Off-Line 52

TABLE OF CONTENTS (CONT)

SECTION PAGE

H.6. Transfer 2NPS-SWG001 Off-Site Feed from 2RTX-XSR1B to 2RTX-XSR1A with Main Generator Off-Line 59

H.7. Transfer 2NPS-SWG003 Off-Site Feed from 2RTX-XSR1B to 2RTX-XSR1A with Main Generator Off-Line 65

H.8. Transfer 2NPS-SWG003 Off-Site Feed from 2RTX-XSR1A to 2RTX-XSR1B with Main Generator Off-Line 72

H.9. Transfer 2NPS-SWG002 from 2ABS-X1 to 2RTX-XSR1A 79

H.10. Transfer 2NPS-SWG002 from 2RTX-XRS1A to 2ABS-X1 80

H.11. Removing 2NPS-SWG002 from Service 81

H.12. Transfer of 13.8KV Construction Power Loop from 29952 Feeder to 29951 Feeder 82

H.13. Restoration of 13.8KV Construction Power Loop from 29951 Feeder to 29952 Feeder 84

H.14. Loss of Power to Construction Power Loop..... 86

H.15. Maintaining 13.8 KV Switchgear 2NPS-SWG001 (003) Voltage with 2STX-XNS1 Load Tap Changer in Manual..... 89

ATTACHMENT 1: AUXILIARY TRANSFORMER XS1 PRIMARY LOCKOUT RELAY..... 90

ATTACHMENT 2: AUXILIARY TRANSFORMER XS3 PRIMARY LOCKOUT RELAY..... 91

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

- 3.8.1, AC Sources – Operating
- 3.8.2, AC Sources - Shutdown

A.2. Licensee Documentation

USAR Chapter 8.0, Electric Power

Section 8.3.1 A.C. Power Systems

A.3. Policies, Programs, and Procedures

- N2-OP-54A, Normal Switchgear Building Ventilation
- N2-OP-70, Station Electrical Feed And 115KV Switchyard
- N2-OP-73A, Normal DC Distribution
- N2-OP-74A, Emergency DC Distribution

A.4. Technical Information

A.4.1. Flow Diagrams

None

A.4.2. Electrical Diagrams

- LSK 24-8.2A, B, D, E, G through 8.2V, Normal Station Service (13.8KV) Breaker Controls and Auto Transfer
- LSK 24-8.2C and F, Normal Station Service Auto Transfer Logic
- ESK 5NJS01, FDR to XFMR 2NJS-X1C & X1D ACB 1-5
- ESK 5NJS02, FDR to XFMR 2NJS-X1A, X1B & X1G ACB 1-14
- ESK 5NJS03, FDR to XFMR 2NJS-X3C & X3D ACB 3-3
- ESK 5NJS04, FDR to XFMR 2NJS-X3A, X3B & X3G ACB 3-13
- ESK 5NJS05, FDR to XFMR 2NJS-X1F ACB 14-4
- ESK 5NJS06, FDR to XFMR 2NJS-X1E ACB 14-8
- ESK 5NJS07, FDR to XFMR 2NJS-X3F ACB 15-1
- ESK 5NJS08, FDR to XFMR 2NJS-X3E ACB 15-7
- ESK 5NJS09, FDR to XFMR 2NJS-X1H, X1J & X1K ACB1-1A

A. REFERENCES AND COMMITMENTS (Continued)

A.4.2 (Continued)

- ESK 5NJS10, FDR to XFMR 2NJS-X3H, J, K ACB 3-1A
- ESK 5NNS01, Feeder to Aux XFMR 2ATX-XS1 ACB 1-4
- ESK 5NNS02, Feeder to Aux XFMR 2ATX-XS3 ACB 3-6
- ESK 5NNS03, Bus 2NNS-SWG011 Sta Svce Sply ACB11-3
- ESK 5NNS06, Bus 2NNS-SWG012 Bus Prot
- ESK 5NNS08, Bus 2NNS-SWG011 & 012 UV Prot
- ESK 5NNS09, Bus 2NNS-SWG013 Sta Svce Sply ACB 13-6
- ESK 5NNS10, Bus 2NNS-SWG011 & 013 Prim Bus Prot
- ESK 5NNS11, Bus 2NNS-SWG012 & 2NNS-SWG013 Tie ACB 13-10
- ESK 5NNS12, Bus 2NNS-SWG013 U.V. Prot
- ESK 5NNS13, Bus 2NNS-SWG012 Backup Bus Prot
- ESK 5NNS14, Bus 2NNS-SWG011 & SWG012 Tie ACB 11-1
- ESK 5NNS16, Bus 2NNS-SWG018 Supply ACB 18-2
- ESK 5NNS17, Bus 2NNS-SWG016 Supply ACB 16-2
- ESK 5NNS18, Bus 2NNS-SWG017 Supply ACB 17-2
- ESK 5NNS19, Bus 2NNS-SWG014 Supply ACB 14-2
- ESK 5NNS20, Bus 2NNS-SWG014 Supply ACB 14-1
- ESK 5NNS21, Bus 2NNS-SWG014 Protection
- ESK 5NNS22, Bus 2NNS-SWG015 Supply ACB 15-3
- ESK 5NNS23, Bus 2NNS-SWG015 Supply ACB 15-8
- ESK 5NNS24, Bus 2NNS-SWG015 Protection
- ESK 5NNS25, Bus 2NNS-SWG014 & 015 UV Prot
- ESK 5NPS01, Bus 2NPS-SWG001 Norm Supply ACB 1-3
- ESK 5NPS02, Bus 2NPS-SWG001 Prot
- ESK 5NPS03, Bus 2NPS-SWG001 Reserve Supply ACB 1-16
- ESK 5NPS04, Bus 2NPS-SWG003 Normal Supply ACB 3-14
- ESK 5NPS05, Bus 2NPS-SWG003 Prot
- ESK 5NPS06, Bus 2NPS-SWG003 Reserve Supply ACB 3-1
- ESK 5NPS07, Bus 2NPS-SWG002 Supply ACB 2-5
- ESK 5NPS08, Bus 2NPS-SWG002 Prot
- ESK 5NPS09, Bus 2NPS-SWG002 Reserve Supply ACB 2-1
- ESK 5NPS10, Bus 2NPS-SWG001, 002 & 003 UV Prot

A.4.2 (Continued)

- ESK 5NPS11, 2NPS-SWG001 Reserve Supply ACB 1-1
- ESK 5NPS12, Bus 2NPS-SWG003 Reserve Sply ACB 3-16
- ESK 5NPS13, Auto Xfr Ckt Bus 2NPS-SWG001
- ESK 5NPS14, Auto Xfr Ckt Bus 2NPS-SWG003
- ESK 8NPS01, 13.8KV Bus Protection (TX 2RTX-XSR1A)
- ESK 8NPS02, 13.8KV Bus Protection (TX 2RTX-XSR1B)
- ESK 8NPS03, 13.8KV Bus 2NPS-SWG001 Rly & Mtr
- ESK 8NPS04, 13.8KV Bus 2NPS-SWG003 Rly & Mtr
- ESK 8NPS05, 13.8KV Bus 2NPS-SWG002 Rly & Mtr
- ESK 8SPX01, XFMR 2ATX-XS1 Rly
- ESK 8SPX02, XFMR 2ATX-XS3 Rly
- ESK 8SPX04, Xfmr 2ATX-XS1 Pri Prot
- ESK 8SPX08, XFMR 2ATX-XS3 Pri Prot
- ESK 8SPX09, Transformer 2ATX-XS1 & XS3 Fault Pressure Protection
- ESK 8SYS01, Synchronizing Station Service
- EE-001B, Reserve & Normal Station Service Transformers
- EE-001C, 4.16KV Aux XFMR Norm 4.16KV & 600V Sys
- EE-001D, Emergency 4.16KV & 600V System
- EE-001F, 2NPS-SWG001 Normal Bus
- EE-001G, Normal Bus 2NPS-SWG002
- EE-001H, 2NPS-SWG003 Normal Bus
- EE-001J, Emer Bus 2EPS*SWG001, 002, 003 & 004 Norm Bus 2NPS-SWG004 & 005
- EE-M01A, Normal Power Distribution
- EE-M01B, Emergency Power Distribution
- EE-M01C and M01D, Normal 600V & 120VAC
- EE-M01E, Emergency 600V & 120VAC
- EE-M01F, Emer & Norm 125V & 24/48 VDC
- EE-M01G, Normal 125 VDC
- EE-F01A, Facility Power Loop
- EE-F01B, Facility Power Loop
- EF-51X, 13.8 KV Facility Power Loop

A.4.3. Vendor Manuals

Electrical Protection Assembly - "Instruction Manual for Electrical Protection Assembly #914E175", Issued 17 Dec. 80

A.5. **Supplemental References**

A.5.1. P.O. Specifications:

- NMP2 - E011C - Normal Station Service Transformer
- NMP2 - E011G - Auxiliary Stepdown Transformer
- NMP2 - E015E - 13.8KV Metal Clad Switchgear
- NMP2 - E022D - 13.8KV & 4.16KV Non-segregated Phase Bus Duct
- MPL-C72 (G.E. Spec.) - Electrical Protection Assembly

A.5.2. Engineering Letter

- A. Julka to R. Dean, May 15, 1992 - File No. SM2-E91-0618
- A. Julka to R. Dean, Nov. 2, 1993 - File No. SM2-E93-0878
- A. Raju to A. Sassani, Sept. 20, 1995 - File No. ESB2-E95-1160. Subject: Main Gen Rating with Normal Station Service Xfmr Out of Service
- Safety Evaluation 98-049, Paralleling of 13.8KV Buses 2NPS*SWG001 and 003 during plant shutdown
- ECP-19-000363, ESR-16-000017 - Install Breakers in Alternate Feed Cubicles for 2NPS-SWG001 and 2NPS-SWG003

A.6. **Commitments**

Sequence Number	Commitment Number	Description
1	003793-00	NMP 64418, SER 25-89 Evaluation, Notify Electrical Maintenance to locate electrical fault, and coordinate actions with Meter & Test. Correct any problems before returning the system to normal.
2	DER 2-96-3449	Turbine Trip - POT Drawer open
3	DER 2005-2583	Notification to Power Control When Transferring House Loads

B. SYSTEM DESCRIPTION

The 24.9KV main generator leads are tapped to provide power to the Normal Station Service Transformer 2STX-XNS1, which has two 13.8KV windings.

During normal operation, 13.8KV bus 2NPS-SWG001 is fed from the 'X' winding of Normal Station Service Transformer 2STX-XNS1; and 13.8KV bus 2NPS-SWG003 is fed from the 'Y' winding of Normal Station Service Transformer 2STX-XNS1. 13.8KV bus 2NPS-SWG002 is fed from Auxiliary Boiler Service Transformer 2ABS-X1. During startup, shutdown and standby, Reserve Station Service Transformer 2RTX-XSR1A feeds 13.8KV bus 2NPS-SWG001; and Reserve Station Service Transformer 2RTX-XSR1B feeds 13.8KV bus 2NPS-SWG003. Reserve Station Service Transformers 2RTX-XSR1A and 2RTX-XSR1B and Auxiliary Boiler Service transformer 2ABS-X1 are fed at 115KV from two off-site sources.

Transformer 2ATX-XS1, fed from 13.8KV bus 2NPS-SWG001, supplies 4160V power to normal 4160V buses 2NNS-SWG011 and 2NNS-SWG014. Bus 2NNS-SWG011 has a tie breaker which feeds 4160V bus 2NNS-SWG012.

Transformer 2ATX-XS3, fed from 13.8KV bus 2NPS-SWG003, supplies 4160V power to normal 4160V buses 2NNS-SWG013 and 2NNS-SWG015. Bus 2NNS-SWG013 has a tie breaker which can feed 4160V bus 2NNS-SWG012.

There are eight 600V load centers, supplied by the 13.8KV buses 2NPS-SWG001 and 2NPS-SWG003. Seven 600V load centers, 2NJS-US1,2,3,4,8,9,10, are divided into 'A', 'B' and 'C' sections, The 'A' section is fed from 13.8KV bus 2NPS-SWG001 and a tie breaker supplies the 'C' sections. The 'B' sections are fed from 13.8KV bus 2NPS-SWG003 and a tie breaker can supply to 'C' sections. One 600V load center, 2NJS-US7, is divided into 'A' and 'B' sections. The 'A' section is supplied from 13.8KV bus 2NPS-SWG001; and the 'B' section is supplied from 13.8KV bus 2NPS-SWG003. There is a normally open tie breaker between the two sections.

All three divisions are normally energized from off-site power by Reserve Service Station Transformer 1A and Reserve Station Service Transformer 1B, with Auxiliary Boiler Transformer 2ABS-X1 as a backup for division I and II. In the event of a loss of off-site power, each division has a standby diesel generator which would start automatically in order to provide power to that division.

There are two 480 vac buses, Main 1 and Main 2, associated with the Security Switchgear which are fed from 13.8 kv switchgear. Main 1 is fed by 2NPS-SWG001-5 (through transformer 2JBK-X1) and Main 2 is fed by 2NPS-SWG003-3 (through transformer 2JBK-X2). There is an "Inter-tie" breaker that can be used to interconnect the two buses.

Modification ECP-11-000813 installs fuse status indication amber lamps in switchgear 2NPS-SWG001 and 2NPS-SWG003 voltage meter cabinets (cubical15). The fuses in which these lamps monitor facilitate slow transfer from normal power to reserve power for the switchgears via under-voltage relay 27R-2NPSX09. If the fuses were actuated, automatic slow transfer would be defeated because there would be no power to the auxiliary relay (27RX-2NPSX16) being driven by 27R-2NPSX09. A CR is required if either lamp is found extinguished during switchgear walkdowns.

B. SYSTEM DESCRIPTION (Continued)

Panel indicators associated with the Security Switchgear are as follows:

SOURCE 1 AVAILABLE - White lamp which indicates power to Main 1
SOURCE 2 AVAILABLE - White lamp which indicates power to Main 2

Blue lamp - Switchgear breaker control in MANUAL mode

For Main 1:	Red lamp	-	Breaker closed
	Green lamp	-	Breaker open
	Amber lamp	-	Breaker overcurrent trip (labeled O/C TRIP)
For Main 2:	Red lamp	-	Breaker closed
	Green lamp		Breaker open
	Amber lamp		Breaker overcurrent trip (labeled O/C TRIP)
Control switches:	For Main 1	-	OPEN/CLOSE
	For Main 2	-	OPEN/CLOSE
	Selector switch	-	MANUAL/AUTO
	Test switch	-	TEST1/NORM/TEST2
	Bus Tie	-	OPEN/CLOSE

Normal configuration for the Security Switchgear is with Main 1 and Main 2 closed and the Bus Tie breaker open. The Selector switch is in the AUTO position and the Test switch is in the NORM position.

With a loss of power to Main 1 or Main 2, the Bus Tie breaker will close and the breaker that lost power will trip open. When power is restored, the SOURCE 1(2) AVAIL. lamp will light. After a 15 minute time delay, the switchgear will re-align to normal. If the Selector switch is in MANUAL, no auto transfers will occur.

C. OPERATING REQUIREMENTS

The following systems must be in operation per their respective operating procedures to support the 13.8KV AC Power Distribution System:

- N2-OP-54A, Normal Switchgear Building Ventilation
- N2-OP-70, Station Electrical Feed And 115KV Switchyard
- N2-OP-73, Normal DC Distribution
- N2-OP-74, Emergency DC Distribution

D. PRECAUTIONS AND LIMITATIONS

D.1. 2RTX-SW001 Neutral Switch must be in the open position for normal operation of Reserve Station Service Transformer 1A. For operation with 2RTX-SW001 Neutral Ground Switch in the closed position, refer to Subsection 0.

D.2. At no time will both emergency bus 2ENS*SWG101 and 2ENS*SWG103 be paralleled to bus 2NNS-SWG018.

D.3. All breaker racking operations shall be performed per N2-ELU-01.

D.4. Load tap changers for 2STX-NS1, 2RTX-XSR1A and 2RTX-XSR1B will normally be operated in the AUTO mode.

D.5. Except as permitted in approved procedures,
2NPS-SWG001 cubicle 1-1 (normal feed from Reserve ‘A’) and
2NPS-SWG001 cubicle 1-16 (alternate feed from Reserve ‘B’)
may have breakers installed at the same time, provided
one is maintained open **AND** in pull-to-lock **AND** with the close circuit fuses removed.
These measures provide additional administrative controls (via Operations procedures) to prevent inadvertently paralleling two sources.

D.6. Except as permitted in approved procedures,
2NPS-SWG003 cubicle 3-1 (normal feed from Reserve ‘B’) and
2NPS-SWG003 cubicle 3-16 (alternate feed from Reserve ‘A’)
may have breakers installed at the same time, provided
one is maintained open **AND** in pull-to-lock **AND** with the close circuit fuses removed.
These measures provide additional administrative controls (via Operations procedures) to prevent inadvertently paralleling two sources.

D.7. Due to Main Transformer Limitations, Maximum Gross Generator Output shall be limited to 1343 MWe when the Normal Station Service Transformer is out of service.

D.8. When returning potential transformer drawers to their normal configuration, care must be taken to ensure that the drawer is fully engaged such that all circuit connections are properly connected. **[C2]**

D.9. As an operator good practice, the power supply breakers to 2NPS-SWG001 (1-1, 1-3 or 1-4) and 2NPS-SWG003 (3-1, 3-6 or 3-14) should be checked properly recharged following normal opening operation. Refer to N2-ELU-01 Subsection 3.4.

F. **NORMAL OPERATIONS (Continued)** Initials

F.3. **Transfer Station Service from Normal to Reserve**

NOTE

- The following step will ensure 2NPS-SWG001 and 2NPS-SWG003 Closing Permissives are satisfied.
- Main Generator output is limited to 1343 MWe when the Normal Station Service Transformer is taken out of service.
- Unless otherwise specified, the following steps are performed in the Control Room at 2CEC*PNL852.
- OP-NM-108-107-1002 contains requirements for making notifications of load transfers in Section 3.1 and Attachment 4. **[C3]**
- This subsection shall **not** be used to transfer 2NPS-SWG001 or 2NPS-SWG003 to 2RTX-XSR1A if 2RTX-XSR1A is supplying 2NPS-SWG002. Closing in 2NPS-SWG001 or 2NPS-SWG003 to 2RTX-XSR1A with 2RTX-SW001 Neutral Switch closed will compromise the switchgear's protection.

F.3.1. In Relay Room, verify the following Lockout Relays reset:

- At 2CEC-PNL809, RES STA SER XFMR 1B BU PROT LOCKOUT RELAY (86-2SPRZ08). ..... ☑
- At 2CEC-PNL808, RES STA SER XFMR 1B PRIM PROT LO RELAY (86-2SPRY01). ..... ☑
- At 2CEC-PNL806, RES STA SER XFMR 1A BU PROT LOCKOUT RELAY (86-2SPRZ01). ..... ☑
- At 2CEC-PNL805, RES STA SER XFMR 1A PRIM PROT LO RELAY (86-2SPRX01). ..... ☑ OPE

F.3.2. In Relay Room, perform the following:

a. Check ALL Relay Flags reset. ..... ☑

b. Inform the SM of any Relay Flags **not** reset. ..... ☑

c. Reset Relay Flags as directed by SM. ..... ☑ OPE

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.3 IF 2RTX-XSR1A is to supply 2NPS-SWG001,
THEN synchronize 2RTX-XSR1A Transformer to 13.8KV 2NPS-SWG001 Bus as follows:

CAUTION

Closing in 2NPS-SWG001 or 2NPS-SWG003 to 2RTX-XSR1A with 2RTX-SW001 Neutral Switch closed will compromise the switchgear's protection.

a. At 2RTX-SW001, confirm 2RTX-SW001, NEUTRAL GROUND SWITCH, locked open..... ☑

b. Place SYNCHRONIZE RESERVE A, Sync Switch, across Breaker 1-1 to NPS 13.8KV BUS 001 in ON..... ☐

c. Verify frequency is in sync as indicated by a stable SYNCHROSCOPE indicator. ..... ☐

d. Observe voltage indication on 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter. .. ☐

e. IF 2RTX-XSR1A Transformer Voltage needs to be adjusted, THEN perform the following:

1) Place RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT Switch in MAN. ..... ☐

2) Raise OR lower Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1A LOAD TAP CHANGER control switch. ..... ☐

f. Observe Amps indication on RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT Ammeters..... ☐

NOTE

Indication of load transferred to 2RTX-XSR1A by observation of a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT ammeters may **not** be evident depending on incoming and running voltages and on bus loading.

g. Close BREAKER 1-1..... ☐

h. Place SYNCHRONIZE RESERVE A, Synch Switch, across BREAKER 1-1 to NPS 13.8KV BUS 001 in OFF. ..... ☐

i. Open BREAKER 1-3, 2STX-XNS1 Normal Station Service Transformer Supply to NPS 13.8KV BUS 001..... ☐

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.3 (continued)

j. Observe load transferred to 2RTX-XSR1A transformer as indicated by a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT Ammeters. ..... ☐

k. Verify RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT Switch in AUTO. ..... ☐

l. Observe voltage indication at 13.8KV NORM BUS NPS 001 KILOVOLTS Voltmeter. ..... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.4. IF 2RTX-XSR1B is to supply 2NPS-SWG001, THEN synchronize 2RTX-XSR1B Transformer to 13.8KV 2NPS-SWG001 Bus as follows:

a. Place SYNCHRONIZE RESERVE B, Sync Switch, across Breaker 1-16 to NPS 13.8KV BUS 001 in ON..... ☐

b. Verify frequency is in sync as indicated by a stable SYNCHROSCOPE indicator. ..... ☐

c. Observe voltage indication on 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter. .. ☐

d. IF 2RTX-XSR1B Transformer Voltage needs to be adjusted, THEN perform the following:

1) Place RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in MAN. ..... ☐

2) Raise OR Lower Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1B LOAD TAP CHANGER control switch. ..... ☐

e. Observe amps indication on RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters..... ☐

NOTE

Indication of load transferred to 2RTX-XSR1B by observation of a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT ammeters may **not** be evident depending on incoming and running voltages and on bus loading.

f. Close BREAKER 1-16..... ☐

g. Place SYNCHRONIZE RESERVE B, Sync Switch, across BREAKER 1-16 to NPS 13.8KV BUS 001 in OFF. ..... ☐

h. Open BREAKER 1-3, 2STX-XNS1 Normal Station Service Transformer Supply to NPS 13.8KV BUS 001..... ☐

i. Observe load transferred to 2RTX-XSR1B transformer as indicated by a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters. ..... ☐

j. Verify RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in AUTO..... ☐

k. Observe voltage indication at 13.8KV NORM BUS NPS 001 KILOVOLTS Voltmeter..... ☐

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.5. IF 2RTX-XSR1B is to supply 2NPS-SWG003,
THEN synchronize 2RTX-XSR1B Transformer to 13.8KV 2NPS-SWG003 Bus as follows:

a. Place SYNCHRONIZE RES STA SVCE XFMR 1B, Sync Switch, across Breaker 3-1 to NPS 13.8KV BUS 003 in ON. ..... ☐

b. Verify frequency is in sync as indicated by a stable SYNCHROSCOPE indicator. ..... ☐

c. Observe voltage indication on 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter. .. ☐

d. IF 2RTX-XSR1B Transformer Voltage needs to be adjusted, THEN perform the following:

1) Place RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in MAN. ..... ☐

2) Raise OR lower Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1B LOAD TAP CHANGER Control Switch. ..... ☐

e. Observe amps indication on RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters..... ☐

NOTE

Indication of load transferred to 2RTX-XSR1B by observation of a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT ammeters may **not** be evident depending on incoming and running voltages and on bus loading.

f. Close BREAKER 3-1..... ☐

g. Place SYNCHRONIZE RES STA SVCE XFMR 1B, Sync Switch, across BREAKER 3-1 TO NPS 13.8KV BUS 003 in OFF..... ☐

h. Open BREAKER 3-14, 2STX-XNS1 Normal Station Service Transformer Supply to NPS 13.8KV BUS 003. ..... ☐

i. Observe load transferred to 2RTX-XSR1B transformer as indicated by a rise in RES STA SVCE XFMR 1B PHASE 1, 2, 3 PRIM CURRENT Ammeters. ..... ☐

j. Verify RES STA SVCE XFMR 1B LTC AUTO-MANUAL SELECT Switch in AUTO..... ☐

k. Observe voltage indication at 13.8KV NORM BUS NPS 003 KILOVOLTS Voltmeter. ..... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.6. IF 2RTX-XSR1A is to supply 2NPS-SWG003,
THEN synchronize 2RTX-XSR1A Transformer to 13.8KV 2NPS-SWG003 Bus as follows:

CAUTION

Closing in 2NPS-SWG001 or 2NPS-SWG003 to 2RTX-XSR1A with 2RTX-SW001 Neutral Switch closed will compromise the switchgear's protection.

a. At 2RTX-SWG001, confirm 2RTX-SWG001, NEUTRAL GROUND SWITCH Locked Open..... ☐

b. Place SYNCHRONIZE RES STA SVCE XFMR 1A, Sync Switch, across BREAKER 3-16 to NPS 13.8KV BUS 003 in ON. ..... ☐

c. Verify frequency is in sync as indicated by a stable SYNCHROSCOPE indicator. ..... ☐

d. Observe voltage indication on 13.8KV STX-XNS1/RTX-XSR1A/RTX-XSR1B RUNNING VOLTS meter is the same as 13.8KV BUS NPS 001/003 INCOMING VOLTS meter. .. ☐

e. IF 2RTX-XSR1A Transformer Voltage needs to be adjusted, THEN perform the following:

1) Place RES STA SVCE XFMR 1A LTC AUTO-MANUAL SELECT Switch in MAN. ..... ☐

2) Raise OR lower Transformer Voltage by changing Transformer Tap Changer setting with RES STA SVCE XFMR 1A LOAD TAP CHANGER Control Switch. ..... ☐

f. Observe amps indication on RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT Ammeters..... ☐

NOTE

Indication of load transferred to 2RTX-XSR1A by observation of a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT ammeters may **not** be evident depending on incoming and running voltages and on bus loading.

g. Close BREAKER 3-16..... ☐

h. Observe load transferred to 2RTX-XSR1A transformer as indicated by a rise in RES STA SVCE XFMR 1A PHASE 1, 2, 3 PRIM CURRENT Ammeters. ..... ☐

Constellation

Training ID: **Nov 2023 NRC JPM F** Revision: **0.0**

Title: **Temper Service Water Using Circulating Water (Alternate Path)**

Approvals:

	Signature	/ Printed Name	Date
Developed by:	Signature on File	/ Mike Alexander	10/27/23
Validated by:	N/A	Mike Carno	7/13/23
Facility Reviewer:	Signature on File	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-11, Rev. 01400, Service Water System
2. N2-ARP-601100, Rev. 00600, 2CEC*PNL601 Series 100 Alarm Response Procedures
3. NUREG 1123 K/A 510000 A4.02 (3.6)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the operator's ability to temper Service Water with Circulating Water.

   b. This JPM is considered alternate path because the operator will be required to respond to Annunciator 601137, SERVICE WATER INTAKE TUNNEL DIV 1/DIV 2 WATER TEMP LOW. The operator will determine that the Division 1 Bar Rack Heaters failed to energize as required and will manually energize the heaters.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-276000-01030, Respond To Annunciator 601137, Service Water Intake Tunnel DIV 1/DIV 2 Water Temp Low

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A	
Evaluation Method:	☒ Perform ☐ Simulate				
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom				
Time Critical Task:	☐ Yes ☒ No				
Alternate Path:	☒ Yes ☐ No				
Safety Function:	4		Heat Removal from Reactor Core		
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)			3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	510000 Service Water System A4.02 - Ability to manually operate and/or monitor in the control room: Valve operations				
K/A Importance Rating:	RO	3.6	SRO	-	

4. Recommended Start Location

   a. Simulator

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) None

2) Remotes

a) **MT03**, Lake Ontario Temperature, FINAL = 40 **Inserted**

3) Overrides

a) **DI-627**, OFF RACK HTR TL 1D, FINAL = ON **Inserted**

4) Annunciators

a) None

5) Event Triggers

Event #	Event Action	Command
1	hzacwswpfi511>0.075 (Tempering flow is >~1000 gpm)	irf mt03 (1 0) 37 1:00
2	zdcw2swpssr1a(2)==1 (Division 1 Bar Heater Control Switch placed in ON)	dor di-627

6) Equipment Out of Service

a) None

7) Support Documentation

a) Prepare a copy of N2-OP-11, Section F.9.0. Placekeep F.9.1 through F.9.4 as complete. Next step to be performed is F.9.5.

8) Miscellaneous

a) IC-207 (For ILT 22-2, not paired)
-OR-

b) IC setup:

(1) Reset to IC-021

6. Strategy Code

a. None

7. Tools and Equipment

a. None

8. Commitments

    a. None

9. Prerequisites

    a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Intake Bay Temperature is ~40°F and continuing to slowly lower. • CWS blowdown is in service. • The crew is preparing to temper Service Water with Circulating Water. • Initial in-plant actions for placing SWP tempering in service are complete. • An operator is available in the field to support any additional actions for SWP tempering. • SWP and CWS Radiation Monitors are in service with normal readings. • Chemistry has been informed that SWP tempering will be performed. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *place SWP tempering in service (~3500 gpm) using CWS with 2SWP-TIK512 in MANUAL; in accordance with N2-OP-11, Section F.9.0. The procedure is in progress up to step F.9.5.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	Manual Control of Tempering is the normal method of Tempering Flow control, however Automatic Control may also be used.		
2.	For Manual Control of Tempering, performs the following:	P (F.9.5)	SAT / UNSAT ***STD:*** *Reads/Reviews and placekeeps step.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.	Places 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW, in M (Manual).	P (F.9.5.1)	SAT / UNSAT ***STD:*** *Observes that 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW controller is already in M on 2CEC*PNL601 by observing "M" indicating light lit on the controller.*
Procedure Note:	This Controller works in reverse of most controllers in that the manual slider must be taken to the right to close the valve. 100% Controller Output is maximum Close Demand.		
4.	Using 2SWP-TIK512 manual slider, closes 2SWP-TV512	P (F.9.5.2)	SAT / UNSAT ***STD:*** *Observes that 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW controller is already closed on 2CEC*PNL601 by observing the HORIZONTAL OUTPUT indication of 100% on the controller.*
5.	Opens 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT. **Cue:** *As the field operator contacted to open V902, acknowledge the direction and inform the operator that 2SWP-V902 is open.*	P (F.9.5.3)	***PASS / FAIL** ***STD:*** *Directs field operator to open 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT. Acknowledges field operator report and placekeeps step.* ***Failure =*** *Direction to and acknowledgement from field operator to open 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT not performed.*
Procedure Note:	This Controller works in reverse of most controllers in that the manual slider must be taken to the left to open the valve. 0% Controller Output is maximum Open Demand.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
6.	Using 2SWP-TIK512 manual slider, opens 2SWP-TV512 to establish desired Tempering Flow less than or equal to 5000 gpm as indicated on 2SWP-FI511, INTAKE TEMPERING FLOW, OR NOT to exceed limit provided by Environmental Protection Department	P (F.9.5.4)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL601, OPENS 2SWP-TV512 by placing the MANUAL SLIDER on 2SWP-TIK512 to the LEFT until flow indicates ~3500 gpm on 2SWP-FI511. Credit can be taken to establish >1000 gpm if Annunciator 601137 alarms before establishing ~3500 gpm.* ***Failure =*** *Flow >1000 gpm as indicated on 2SWP-FI511 not achieved.*
Alternate Path:	Once Intake Tempering Flow has exceeded ~1000 gpm, intake bay temperature will begin to lower to 37°F over the course of 1 minute. When temperature drops below 39.6°F, Annunciator 601137 will alarm. The operator will reference the ARP and determine the Division 1 Bar Rack Heaters did not energize as expected. The operator will take manual action to energize the heaters.		
7.	Responds to Annunciator 601137 by referring to the ARP. **Cue:** If operator attempts to check computer points SWPTA49, 50, 51, or 52, inform them that they are 37°F.	P (N2-ARP-601100)	SAT / UNSAT ***STD:*** *Reads/Reviews the Automatic Response and required actions sections.*
8.	Performs the actions of N2-ARP-601137, Operator Action 1 (first bullet) and verifies Div I bar rack heaters tunnel 1 are on.	P (N2-ARP-601100 Operator Action 1 first bullet)	***PASS / FAIL** ***STD:*** *At 2CEC*PNL601, energizes DIV I BAR RACK HEATERS TUNNEL 1 by rotating the control switch clockwise to the ON position and verifies the RED light lit and the GREEN light not lit.* ***Failure =*** *DIV I BAR RACK HEATERS TUNNEL 1 RED light ON and GREEN light OFF not achieved.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Evaluator Note:	After the Operator has placed the DIV I BAR RACK HEATERS TUNNEL 1 control switch to on, provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*SWP tempering has been placed in service (>1000 gpm) using CWS with 2SWP-TIK512 in MANUAL; in accordance with N2-OP-11, Section F.9.0 with appropriate Bar Rack heaters on.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Intake Bay Temperature is ~40°F and continuing to slowly lower. • CWS blowdown is in service. • The crew is preparing to temper Service Water with Circulating Water. • Initial in-plant actions for placing SWP tempering in service are complete. • An operator is available in the field to support any additional actions for SWP tempering. • SWP and CWS Radiation Monitors are in service with normal readings. • Chemistry has been informed that SWP tempering will be performed.

INITIATING CUE	**(Operator Name)**, *place SWP tempering in service (~3500 gpm) using CWS with 2SWP-TIK512 in MANUAL; in accordance with N2-OP-11, Section F.9.0. The procedure is in progress up to step F.9.5.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Intake Bay Temperature is ~40°F and continuing to slowly lower. • CWS blowdown is in service. • The crew is preparing to temper Service Water with Circulating Water. • Initial in-plant actions for placing SWP tempering in service are complete. • An operator is available in the field to support any additional actions for SWP tempering. • SWP and CWS Radiation Monitors are in service with normal readings. • Chemistry has been informed that SWP tempering will be performed. **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *place SWP tempering in service (~3500 gpm) using CWS with 2SWP-TIK512 in MANUAL; in accordance with N2-OP-11, Section F.9.0. The procedure is in progress up to step F.9.5.*

NINE MILE POINT NUCLEAR STATION UNIT 2
OPERATING PROCEDURE

N2-OP-11

Revision 01500

SERVICE WATER SYSTEM

TECHNICAL SPECIFICATION REQUIRED

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
015	00	Minor Revision to incorporate:

PCR-18-05732 Delete two P+L that provide no value per review for AR-04172818-02/26.

PCR-19-03392 Revise method to secure SWP pump to ensure discharge check valve is properly seated.

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PCR-18-05732

- D.5.0 and D.29.0 where deleted and remaining P+L renumbered, NO rev bar used.

PCR-19-03392

- G.1.1 NOTE last bullet updated with flow which could cause alarm.
- G.1.4, change flow to 1,500 gpm.
- G.1.6 and G.1.7, revised to turn off SWP pump, and then verify discharge valve closes.

Writer Changes

- A.5.0 and A.6.0, DER/CR numbers updated to IR numbers – no rev bars used.
- F.7.0, G.1.0, H.9.0 and H.10.0, title changed from [OSP] to [SOP].

TABLE OF CONTENTS

SECTION			PAGE
A.		REFERENCES AND COMMITMENTS	5
B.		SYSTEM DESCRIPTION	9
C.		OPERATING REQUIREMENTS	15
D.		PRECAUTIONS AND LIMITATIONS	16
E.		STARTUP	23
	1.0	Starting First Pump from Full Shutdown Condition **[SOP]**	23
	2.0	Starting Additional Pumps When Other Pumps Are Running **[SOP]**	31
	3.0	Starting Jet Motive Pump	32
	4.0	Starting Service Water Monitoring Station Pump	33
F.		NORMAL OPERATIONS	34
	1.0	Routine System Checks	34
	2.0	Venting Idle Service Water Pump Discharge Strainers	34
	3.0	Starting Make-up Flow to Cooling Tower	35
	4.0	Changing Make-up Flow to Cooling Tower	36
	5.0	Stopping Make-up Flow to Cooling Tower	37
	6.0	Adjusting Discharge Header Total Flow Low Light Setpoints	37
	7.0	Fish Sampling **[SOP]**	38
	8.0	Operating Service Water During Station Shutdown	39
	9.0	Tempering Service Water with Circ Water	40
	10.0	Tempering Service Water with Service Water	42
	11.0	Stopping Tempering of Service Water	43
	12.0	Additional Restrictions on Service Water System Operations **[C5]**	46
	13.0	Flushing Service Water Side of RHS Heat Exchangers	47
	14.0	High Service Water Temperature Operation	48
G.		SHUTDOWN	51
	1.0	Stopping One Service Water Pump When Other Pumps Will Remain Running **[SOP]**	51
	2.0	Stopping Jet Motive Pump	52
	3.0	Stopping Service Water Monitoring Station Pump	52
H.		OFF-NORMAL PROCEDURES	53
	1.0	Nonessential Header Isolation Valve Closure	53
	2.0	Placing Strainers in Continuous Backwash	56
	3.0	Removing Strainers from Continuous Backwash	57
	4.0	Tempering Service Water with 2SWP-TV512 Inoperative	57
	5.0	Reversing Flow through Intake Shaft 1	60

SECTION PAGE

6.0 Restoring Normal Flow through Intake Shaft 1 60

7.0 Filling and Venting HVK Chiller Condensing Water Loop 61

8.0 Filling and Venting Turbine Building Nonessential Header 62

9.0 Isolating Division II Cooling Tower Make-up for Maintenance and Testing on 2SWP*MOV599, 2SWP*FV47B and 2SWP*FV54B **[SOP]** 67

10.0 Restoring Division II Cooling Tower Make-up After Maintenance and Testing On 2SWP*MOV599, 2SWP*FV47B and 2SWP*FV54B **[SOP]** 70

11.0 Operation of 2SWP*V17 and 2SWP*V32 During Unit Outages for Maintenance and Testing on Service Water System 71

12.0 Re-start of Tripped Service Water Pumps 72

13.0 Bypassing and Un-bypassing the Service Water Monitoring Station Pump 73

14.0 Methodology for Determining Service Water Temperature at the Discharge Header 73

Attachment 1, Controller Lineup 75

Attachment 2, Supplied Equipment List 76

Attachment 3, Turbine Building Vent Sequence 81

Attachment 4, Monitoring While Division ll Cooling Tower Make-Up Is Isolated 88

Attachment 5, Division 1 Service Water Draindown During RFO9 89

Attachment 6, Setup for Determining Temperature from Service Water Discharge Header 91

Attachment 7, Division 2 Service Water Draindown During RFO10 92

A. REFERENCES AND COMMITMENTS

1.0 Technical Specifications

- 3.3.3.2, Remote Shutdown System
- 3.7.1, Service Water (SW) System and Ultimate Heat Sink (UHS)

2.0 Licensee Documentation

USAR

- Chapter 2, SITE CHARACTERISTICS,
  Section 2.4, Hydrologic Engineering
  - Subsection 2.4.7, Ice Effects
  - Subsection 2.4.10, Flooding Protection Requirements
  - Subsection 2.4.11, Low Water Considerations
  - Subsection 2.4.11.5, Plant Requirements
  - Subsection 2.4.11, Low Water Considerations
  - Subsection 2.4.11.6, Heat Sink Dependability Requirements
- Chapter 8, ELECTRIC POWER,
  Section 8.3, Onsite Power System:
  - Subsection 8.3.1.1.2, Power Supply Buses, Standby Diesels Generators - Divisions I and II
- Chapter 9, AUXILIARY SYSTEMS,
  Section 9.2, Water Systems
  - Subsection 9.2.1, Service Water System
  - Subsection 9.2.5, Ultimate Heat Sink

Offsite Dose Calculation Manual (ODCM)

- D 3.3.1, Radioactive Liquid Effluent Monitoring Instrumentation

Technical Requirements Manual (TRM)

- 3.3.3.2, Remote Shutdown System
- 3.3.9, Service Water System Instrumentation
- 3.7.1, Plant Service Water System - Shutdown

3.0 Policies, Programs, And Procedures

- N2-OP-11A, Service Water Chemical Treatment System
- N2-OP-12, Traveling Water Screens and Wash Disposal System
- N2-OP-13, Reactor Building Closed Loop Cooling
- N2-OP-14, Turbine Building Closed Loop Cooling System
- N2-OP-19, Instrument and Service Air System
- N2-OP-31, Residual Heat Removal System

A. REFERENCES AND COMMITMENTS (Cont)

3.0 (Continued)

- N2-OP-38, Spent Fuel Pool Cooling and Cleanup System
- N2-OP-52, Reactor Building Ventilation
- N2-OP-53A, Control Building Ventilation System
- N2-OP-54B, Ventilation - Chilled Water (Lithium Bromide)
- N2-OP-55, Turbine Building Ventilation
- N2-OP-57, Diesel Generator Building Ventilation System
- N2-OP-58, Screenwell Building and Fire Pump Rooms Ventilation Systems
- N2-OP-59A, Control Building - Reactor Building Electric Tunnels Ventilation
- N2-OP-62, DBA Hydrogen Recombiner
- N2-OP-71C, 600V A.C. Power Distribution
- N2-OP-72, Standby and Emergency A.C. Distribution System
- N2-OP-74A, Emergency D.C. Distribution
- N2-OP-78, Remote Shutdown System
- N2-OP-79, Radiation Monitoring
- N2-OP-100A, Standby Diesel Generators
- N2-OP-100B, HPCS Diesel Generator
- N2-SOP-11, Loss of Service Water
- N2-OSP-SWP-Q004

4.0 Technical Information

4.1 Flow Diagrams

PID -11A thru 11Q, Service Water System Piping & Instrumentation Diagrams

4.2 Electrical Diagrams

- ESK-5ENS21, Bus 2ENS*SWG101 UV & Load Sequencing 4KV Swgr Cont DC Elementary Diagram
- ESK-5ENS22, Bus 2ENS*SWG103 UV & Load Sequencing 4KV Swgr Cont DC Elementary Diagram
- ESK-5SWP01 thru 5SWP12, Service Water System
- ESK-6SWP01 thru 6SWP42, Service Water System
- ESK-7SWP03 thru 7SWP36, Service Water System
- ESK-11SWP01 thru 11SWP03, Service Water System

A. REFERENCES AND COMMITMENTS (Cont)

4.3 Logic Diagrams

- LSK-9-10A thru 9-10AB, Service Water Logic Diagram
- LSK-24-9.4A thru 24-9.4D, Standby Diesel Generator Load Sequence Logic Diagram

4.4 Vendor Manuals

- A300-EL, Gould Pumps Installation, Operation, and Maintenance Instructions for Service Water Pumps, NMPCNO N2G20000PUMP002, File Code N20351
- 78 N 2307, Zurn Industries Strain-O-Matic Instruction Manual for Service Water Self-Cleaning Strainers, NMPCNO N2Z01000FILTER004, File Code N20065
- CF-SP-96703, Instruction Manual for Worthington Pumps, NMPCNO N2W31500PUMP003, File Code N20701

4.5 Vendor Drawings

- Gould Pumps Incorporated Drawing N239505 1, Service Water Pump Outline Drawing, File Code 0002.170-050-026E
- Gould Pumps Incorporated Drawing N780657 1, Condensing Water Pump Outline Drawing, File Code 0002.170-050-034L
- ITE/Rowan Drawing WD-56593-C-01, Schematic and Assembly Wiring Diagram FVNR Size 1 & 2, File Code 0001.340-211-006 (As updated by DDC 2E11118)

5.0 Supplemental References

- IOC, S. L. Heimovitz to J. Helker, Date: 5/11/93, File Code SM2-M93-0122, Subject: Engineering Guidance SWP Pump Running
- IR 02064797, Potential Misposition Of ASCO Solenoid Valves
- IR 02072408, SWP Strainer Failure When in Continuous Backwash Following A Loss Of Offsite Power
- NMP2 CALC. PH-087, Determination of minimum number of SWP Intake Structure Heaters required by Tech. Specs.
- NMP2 CALC. CS-SWP*35, Service Water Supply Header Temperature indication loop uncertainty
- SDBD-SWP, System Design Basis Document, Service Water System
- IR 02077899, SWP Design Bases
- Modification PN2Y94MX007, Service Water Chemical Treatment System, PCR PC2-9014-50
- IR 02085136, SWP Bar Rack Heaters
- NMPC Safety Evaluation 97-034, Closure Of 2SWP*MOV599 And Opening 2SWP-V8
- NMP2L-1821, SWP Tech. Specs. Change Request
- NMPC Safety Evaluation 98-054, Service Water Lineup During Unit Outages
- NER 2M-026, Service Water System Operational Guideline Modes 4 and 5

A. REFERENCES AND COMMITMENTS (Cont)

5.0 (Continued)

- File Code N9815.RJD, Service Water Operability Requirements in Operational Conditions 1, 2, and 3. R.J. Dean to D.P. Bosnic
- NMP2 Calc. A10.1-N-341, Service Water Flow to Control Room Chiller Condensers
- NER 2M-041 SWP Pump Operability Status Assessment for Inoperable SWP Valves (2SWP*MOV66A/B)
- NMPC Safety Evaluation 2000-046, Service Water System Low Flow Trip Avoidances
- Letter to File #97030, Past Reportability - Justification Why Not Reportable, part of response to DER NM-2002-4947, Single Missile Barrier for 2SWP*MOV30A and B (available as an attachment to Actual Action #2 of DER in eCAP).
- NER-25-005, Evaluation of the RHR Piping, from the Outlet of the Heat Exchanger to the Outermost Isolation Valve, for Lowest Service Temperature (LST)
- NMP2 Calc. HVK-011, Evaluation of Control Building Chillers With 84°F Service Water Temperature
- ECP-13-001035, Fukushima LL Project – Unit 2 FLEX RPV and SFP Make-up Water piping.

6.0 Commitments

Sequence Number	Commitment Number	Description
1	Deleted	Deleted
2	IR 02068849	Procedural Guidance for Fires Outside Control Room
3	IR 02073859	LER 95-04: Unmonitored Release Via DIV II SWP
4	IR 02077899	Possible Inadequate Design Of Service Water Chemical Treatment System
5	IR 02077189	SWP Design Bases
6	IR 02093476	Engineering Review Potentially Impacting SWP Valves
7	IR 02106367	Single Missile Barrier for 2SWP*MOV30A and B
8	IR 02026019	Turbine Building flooding analysis USAR 10.4.5.3
9	IR 02035577	INPO SOER 07-02, "INTAKE COOLING WATER BLOCKAGE"

B. SYSTEM DESCRIPTION

1.0 System Overview

The Service Water (SWP) System is an open-loop cooling system consisting of an Intake Complex, a Discharge Complex, six SWP pumps, and associated valves, piping, and cooling components. Water enters the SWP System from Lake Ontario through two Intake Structures, passes through Trash Rakes and Traveling Water Screens, and enters the Intake Bay. The six SWP Pumps take a suction from the Intake Bay and discharge through automatically operated Discharge Strainers to a common Discharge Header in the Screenwell Building. Two Motor Operated Valves (2SWP*MOV50A & B) are provided to split the System into two separate, redundant loops. These two loops are Safety-Related, and are referred to as Division I (containing Pumps A, C, E) and Division II (containing Pumps B, D, F). All Essential components are fed by the Safety-Related Loops and are powered by the Safety-Related Emergency Power Distribution System. The System also contains two Non Safety-Related Loops which are referred to as the Non-Essential Loops. During an accident, the Non-Essential Loops are isolated from the Safety-Related Loops.

Each of the Safety-Related Loops supply Service Water to the Essential loads in the Reactor, Control, and Diesel Generator Buildings. A takeoff from the Division I Safety-Related Loop common Screenwell Building Header supplies Service Water to the Turbine Building Non-Essential Loop during normal Plant operation. Takeoff from the Division I Safety-Related Loop Reactor Building Supply Header supplies Service Water to the Reactor Building Non-Essential Loop during normal Plant operation. The Service Water discharge from all loops is combined outside the buildings into two separate, redundant Discharge Headers. The Reactor Building Non-Essential Loop returns to the Division I Discharge Header and the Turbine Building Non-Essential Loop returns to the Division II Discharge Header.

Each of the Non-Essential Loops is provided with two Isolation MOVs in series, except the return from the Turbine Building Non-Essential Loop which has one MOV and a Check Valve. Two cross connection lines isolated by manual Isolation Valves (Turbine Building: 2SWP*V32; Reactor Building: 2SWP*V17) can supply the Non-Essential Loops from the Division II Safety-Related Loop. Another cross connection with a manual Isolation Valve (2SWP-V8) is provided to allow the Non-Essential Loops to return to either Safety-Related Discharge Header.

A 30" takeoff from each of the two Discharge Headers provides makeup to the Main Circulating System Cooling Tower, at a maximum of 25,000 gpm, through a pair of hydraulically operated Flow Control Valves in series in each CWS Makeup supply line.

The Discharge Headers return the heated Service Water to the Discharge Bay via Standpipes at elevation 279’ 9". These Standpipes maintain the Safety-Related Headers full to their highest elevations during a loss of all Service Water Pumps. Both Discharge Headers are monitored for Temperature, Pressure, Radiation, and Flow. Abnormal conditions on any of these monitored parameters will not cause any system automatic response. From the Discharge Bay the combined Plant discharge flows by gravity through the Discharge Tunnel, which is contained inside of Intake Tunnel number 1 until the Intake Tunnel ends at its Intake Structure, through the Discharge Tunnel which continues to a point about 1,500 feet from the shoreline, then rises to the two-port Discharge Diffuser.

B. SYSTEM DESCRIPTION (Continued)

The Fish Bypass System takes fish that enter the Plant and returns them to the Lake. The Jet Motive Pump (2SWP-P3) provides the motive force for the Peripheral Jet Pump (2SWP-P4) which takes a suction upstream of the Traveling Screens and discharges to one of two flow paths through a "Y" type, motor operated, Siamese Knife Gate Valve (2SWP-MOVX936/2SWP-MOVY936). The normal flow path is through the Fish Return Line, which runs through Intake Tunnel 2, to the Fish Discharge Structure in the Lake. The other flow path discharges to the Fish Sampling Tank which then overflows into the Intake Bay.

The Tempering System provides heated water to the Intake Bay to warm the water coming into the Plant when Lake Temperature is less than 38°F. Under normal operation this heated water comes from the CWS System, but can be supplied from the Discharge Bay when the Plant is shutdown. The heated water is controlled by the Tempering Valve (2SWP-TV512) and is discharged into the Intake bay via the Tempering Pipe (2-SWP-016-01L-4), a 16" perforated pipe with two rows of forty 1" diameter holes spaced approximately 42" apart.

The Tempering System also provides a water source for two 6 inch dry hydrants, located along the north side of the Screenwell Building and within a missile barrier, such that a nearly infinite source of make-up water to the RPV and SFP is available during Phase 2 of the FLEX strategy, when FLEX pumps are deployed and connected to the hydrants. Two valve lineups are required to ensure a suction path to the lake / intake tunnel is continuously maintained during normal and abnormal operating conditions. The normal lineup to the FLEX hydrants is through FLEX valve 2SWP-V1267, and the abnormal lineup is through FLEX valve 2SWP-V1266. During the normal lineup valve 2SWP-TV512 is aligned for tempering and valve 2SWP-V904 is open. During the abnormal lineup, tempering is aligned through bypass valve 2SWP-V905, and 2SWP-904 is closed.

A freeze protection system, consisting of a 3.2 KW infrared radiant heater and a controller, maintains the temperature of the dry hydrant pipes and missile barrier space above 32°F. The thermostat controller, set to 38°F, energizes the heater when the sensed temperature falls below the 38°F setpoint. Temperature indication is read from digital display 2SWP-TE1273, mounted in the interior of the Screenwell Building.

During normal operation, the Service Water System is controlled from Control Room panel 2CEC*PNL601. If the Control Room must be evacuated, controls are available on the Remote Shutdown Panel (2CES*PNL405) for all six Service Water Pumps, the Pump Discharge MOVs, the MOVs that supply Division I and II Diesel Generators, and the MOVs that supply both RHS Heat Exchangers. All other MOVs in the Service Water System rely on automatic signals only when the Control Room is inaccessible (refer to N2-OP-78 for more information). The Service Water System has a design Pressure and Temperature of 150 psig and 130°F. All components cooled by Service Water are designed for a Maximum Service Water Inlet Temperature of 84°F.

2.0 System Components

The two identical Intake Structures are located about 950 and 1,050 feet from the Lake shoreline on the lake bed at about elevation 223.5’. The structures are hexagonal in shape and are 22.5’ wide. Each of the six sides has a 7.5 x 3 foot intake opening which is set about three feet above the lake bed to minimize silt intake. The intake openings are equipped with vertical Bar Racks with 10" open spacing between each bar to prevent large debris from entering the Service Water System. The Bar Racks consist of nine vertical bars for each intake opening, seven of which are electrically heated to eliminate the potential for frazil ice adhesion. Three of the six sets of Bar Rack Heaters on each Intake Structure are powered by Division I, the other three being powered by Division II. The Bar Rack Heaters are set to energize at 38°F.

B. SYSTEM DESCRIPTION (Continued)

The six Service Water Pumps (2SWP*P1A-F) are motor-driven, horizontal, single-stage, centrifugal pumps. The Pumps are Gould Pumps, Incorporated model 3415M, size 14x16-22H. Each Pump is powered by a 600 hp, 4KV Westinghouse motor which operates at 1200 rpm. The Pumps are designed to deliver 10,000 gpm at 185 feet of TDH. Design Pressure is 150 psig and design Temperature is 100°F. The Pumps are located on elevation 224' in the Service Water Pump Bay area of the Screenwell Building. Division I Pumps are powered from 2ENS*SWG101, Division II Pumps are powered from 2ENS*SWG103. The motor operated Pump Discharge Valves (2SWP*MOV74A-F) open automatically when the respective Pump starts and close automatically when the Pump stops.

The Service Water Pumps auto-start only on Diesel Generator Load Sequencing and will trip on the following:

- Bus Undervoltage
- Motor Electrical Fault

The self-cleaning Discharge Strainers (2SWP*STR4A-F) remove debris which could foul or mechanically damage heat transfer surfaces. The Strainers are Model 593 Strain-O-Matic rotating strainers manufactured by the Water and Waste Treatment Division of Zurn Industries, Incorporated. The Strainers contain 44 cylindrical straining elements having 1/32" perforations which are rotated by an electric motor. This motor is a 600 Volt AC, 3 hp motor manufactured by Reliance Electric. The Strainer design parameters are 150 psig, 130°F, and 10,000 gpm. The Strainer Blowdown Valves (2SWP*MOV1A-F) will open on high ΔP, if in Auto, to backwash the straining elements to the Discharge Bay.

The Control Building Chiller Condensing Water Pumps (2SWP*P2A & B) are motor-driven, horizontal, single stage, centrifugal pumps and are located on elevation 261' in the Control Building Chiller Rooms. The Pumps are Gould Pumps, Incorporated model 3196MT, size 3x4-8. Each Pump is powered by a 10 hp, 600 Volt AC Westinghouse motor which operates at 1800 rpm. The Pumps are designed to deliver 340 gpm at 60 feet of TDH. Design Pressure is 150 psig and design Temperature is 130°F. The Condensing Water Pumps will auto-start on Chiller Inlet Temperature Low (less than or equal to 69.5°F) and the Chiller is running, and will trip if the Chiller stops or Chiller Inlet Temperature is not Low (greater than 69.5°F).

The Jet Motive Pump (2SWP-P3) is a motor-driven, horizontal, single stage, centrifugal pump and is located on elevation 228'-4" of the Screenwell Building. The Pump is a Worthington Pump Corporation Mixflo model 20MN-24 and is powered by a 150 hp, 575 Volt AC General Electric motor which operates at 600 rpm. The Pump is designed to deliver 10,000 gpm at 40 feet of TDH. Design Pressure is 150 psig and design Temperature is 130°F. The only Jet Motive Pump trip signal is Bus Undervoltage.

The Peripheral Jet Pump (2SWP-P4) is an eductor which contains no moving parts. This Pump uses the discharge of the Jet Motive Pump as its motive force and is located on elevation 228'-4" of the Screenwell Building.

The Service Water Monitoring Station Pump (2SWP-P10) is a motor-driven, self-priming, centrifugal pump located on elevation 261' in the Screenwell Building near the Trash Rake Assembly. The Pump is powered by a 3/4 hp, 575 Volt AC motor. Design Pressure is 85 psig and design Temperature is 130°F.

B. SYSTEM DESCRIPTION (Continued)

The four Service Water System Loops supply cooling water to the following Systems and Components:

1. Division I Safety-Related Loop:
   - RHS Heat Exchanger (2RHS*E1A)
   - RHS Pump Seal Cooler back-up supply (2RHS*P1A)
   - Recirc Pump Motor, Seal, & Bearing Oil Coolers (2RCS-P1A) emergency supply (isolated with locked closed valves)
   - Spent Fuel Pool Cooling Heat Exchanger (2SFC*E1A) back-up supply
   - Spent Fuel Pool Emergency Fill
   - Reactor Building Emergency Recirc Unit (2HVR*UC413A)
   - Division I Control Building Unit Coolers
   - Standby Diesel Generator Jacket Water Cooler (2EGS*EG1)
   - Diesel Generator Control Room Unit Cooler (2HVP*UC1A)
   - HPCS Diesel Generator Jacket Water Cooler (2EGS*EG2)
   - HPCS Diesel Generator Control Room Unit Cooler (2HVP*UC2)
   - HPCS Switchgear Room Unit Cooler (2HVC*UC102)
   - DBA Hydrogen Recombiner (2HCS*RBNR1A)
   - Control Room Chilled Water Chiller (2HVK*CHL1A)
   - Control Room Chilled Water emergency supply
   - Division I Reactor Building Unit Coolers
2. Division II Safety Related Loop:
   - RHS Heat Exchanger (2RHS*E1B)
   - RHS Pump Seal Coolers back-up supply (2RHS*P1B & C)
   - Recirc Pump Motor, Seal, & Bearing Oil Coolers (2RCS-P1B) emergency supply (isolated with locked closed valves)
   - Spent Fuel Pool Cooling Heat Exchanger (2SFC*E1B) back-up supply
   - Spent Fuel Pool Emergency Fill
   - Reactor Building Emergency Recirc Unit (2HVR*UC413B)
   - Division II Control Building Unit Coolers
   - Standby Diesel Generator Jacket Water Cooler (2EGS*EG3)
   - Diesel Generator Control Room Unit Cooler (2HVP*UC1B)
   - HPCS Diesel Generator Jacket Water Cooler (2EGS*EG2)
   - HPCS Diesel Generator Control Room Unit Cooler (2HVP*UC2)
   - HPCS Switchgear Room Unit Cooler (2HVC*UC102)
   - DBA Hydrogen Recombiner (2HCS*RBNR1B)

B. SYSTEM DESCRIPTION (Continued)

2. Division II Safety Related Loop (Continued):

- Control Room Chilled Water Chiller (2HVK*CHL1B)
- Control Room Chilled Water emergency supply
- Division II Reactor Building Unit Coolers
- Containment Spray back-up supply

3. Reactor Building Non-Essential Loop:

- CCP Heat Exchangers (2CCP-E1A, B, & C)
- Reactor Building Ventilation Supply Cooling Coil (2HVR-CLC2)
- Track Bay Cask Washdown Station

4. Turbine Building Non-Essential Loop:

- CCS Heat Exchangers (2CCS-E1A, B, & C)
- Condenser Air Removal Pump Seal Coolers (2ARC-E1A & B)
- Steam Jet Air Ejector Precoolers (2ARC-E2A & B)
- 62 Turbine Building Area Unit Coolers
- Water Treating System (Not Used)
- Hypochlorite System (Not Used - Retired In Place)
- Auxiliary Boiler Recirc Pump Seal Coolers (2ABM-P1A & B)
- Auxiliary Boiler Feed System Sample Heat Exchanger (2ABM-E2)
- Auxiliary Boiler Blowdown Sample Coolers (2ABM-SCL1A & B)
- Make-up to Fire Protection Water System Head Tank (2FPW-TK2)
- Li-Br Air Conditioning System (2HVN-CHL1A, B, & C)
- Radwaste Building Hose Stations
- Small Tool Decon and Storage Room Hose Stations
- Turbine Building Hose Stations

3.0 Service Water System Response on A Total Loss-Of-Offsite Power

Service Water Pump start logic will attempt to start any available Service Water Pumps until at least one Pump is running on each division. This procedure assumes that both 2SWP*P1A & B are first to start.

3.1 Immediately upon Loss-of-Offsite Power (LOP), 2SWP*FV47A/B and 2SWP*FV54A/B, Cooling Tower Makeup Valves, fail closed.

3.2 All running Service Water Pumps trip due to Load Shedding.

3.3 Within 10 seconds after the LOP, both Diesel Generators reenergize their respective Switchgear. For Service Water logic, this is T=0.

B. SYSTEM DESCRIPTION (Continued)

3.4 At T=0, the following valves will begin to close, if open:

- 2SWP*MOV74A, PMP 1A DISCH
- 2SWP*MOV74B, PMP 1B DISCH
- 2SWP*MOV74C, PMP 1C DISCH
- 2SWP*MOV74D, PMP 1D DISCH
- 2SWP*MOV74E, PMP 1E DISCH
- 2SWP*MOV74F, PMP 1F DISCH
- 2SWP*MOV50A, PMP 1A DISCH HEADER CROSS-TIE ISOL VLV
- 2SWP*MOV50B, PMP 1B DISCH HEADER CROSS-TIE ISOL VLV
- 2SWP*MOV3A, OUTLET TO TURBINE BLDG
- 2SWP*MOV3B, OUTLET TO TURBINE BLDG
- 2SWP*MOV19A, OUTLET TO REACTOR BLDG
- 2SWP*MOV19B, OUTLET TO REACTOR BLDG
- 2SWP*MOV93A, OUTLET FROM RX BLDG TO DISCH HEADER A
- 2SWP*MOV93B, OUTLET FROM RX BLDG TO DISCH HEADER A
- 2SWP*MOV599, OUTLET FROM TURB BLDG TO DISCH HEADER B

Due to the loss of cooling to CCS and CCP Systems, a Reactor Scram and Turbine Trip will occur with no Operator action.

3.5 At T=0, the following valves will begin to open:

- 2SWP*MOV66A, SERVICE WTR OUTLET
- 2SWP*MOV66B, SERVICE WTR OUTLET
- 2SWP*MOV94A, SERVICE WTR OUTLET
- 2SWP*MOV94B, SERVICE WTR OUTLET

3.6 At T=0, manual starting of Service Water Pumps is blocked (for 70 seconds).

3.7 At T=32, 2SWP*P1A and 2SWP*P1B receive a start signal if control switch is in Normal-After-Start (red flagged).

3.8 At T=37, 2SWP*P1C and 2SWP*P1D receive a start signal if control switch is in Normal-After-Start (red flagged).

3.9 At T=42, 2SWP*P1E and 2SWP*P1F receive a start signal if control switch is in Normal-After-Start (red flagged).

3.10 At T=48, 2SWP*P1A and 2SWP*P1B receive a start signal if control switch is in Normal-After-Stop (green flagged).

3.11 At T=54, 2SWP*P1C and 2SWP*P1D receive a start signal if control switch is in Normal-After-Stop (green flagged).

3.12 At T=61, 2SWP*P1E and 2SWP*P1F receive a start signal if control switch is in Normal-After-Stop (green flagged).

3.13 Immediately upon first Pump restart, 2SWP-MOV582 through 598, Service Water Vacuum Breakers, open.

3.14 2.5 seconds after first Pump restart, 2SWP-MOV582 through 598 close.

3.15 At T=70, block of Service Water Pump manual starting expires.

C. OPERATING REQUIREMENTS

The following systems or equipment must be in operation per their respective operating procedures to support the Service Water System:

- Traveling Water Screens and Wash Disposal System (N2-OP-12)
- Instrument and Service Air System (N2-OP-19)
- Screenwell Building and Fire Pump Rooms Ventilation Systems (N2-OP-58)
- 600V A.C. Power Distribution (N2-OP-71C)
- Standby and Emergency A.C. Distribution System (N2-OP-72)
- Emergency D.C. Distribution (N2-OP-74A)
- Radiation Monitoring (N2-OP-79)
- Instrumentation per the following requirements are in service as necessary to support operation of the Service Water System:

  - ODCM D 3.3.1

  - TRM 3.3.9

D. PRECAUTIONS AND LIMITATIONS

1.0 Service Water Pump Motor Start Limitations:

The following Pump Start Limitations are applicable to the Main Service Water Pumps (P1s), the HVK Chiller Condensing Water Pumps (P2s), and the Jet Motive Pump (P3):

- Two starts in succession from ambient temperature.
- One start from rated temperature.
- For subsequent starts allow 15 minutes running time or 45 minutes idle time between start attempts.

2.0 RHS Heat Exchanger Limitations

2.1 When placing an RHS Heat Exchanger in service, verify by Control Room indication that Service Water Flow exists BEFORE initiating RHS Flow through the Heat Exchanger.

2.2 When taking an RHS Heat Exchanger out of service, the Service Water side should be drained, flushed, and refilled with demineralized water, unless waived by the US/SM.

2.3 Rated Service Water Flow through an RHS Heat Exchanger is 7400 gpm. Service Water flow through an RHS Heat Exchanger should not exceed 7400 gpm.

2.4 When placing an RHS Heat Exchanger in service, 2SWP*RE23A/B, Effluent Radiation Monitor, should be verified in service BEFORE initiating Service Water Flow through the Heat Exchanger. If 2SWP*RE23A/B becomes unavailable or is inoperable, verify 2SWP*RE146A/B is on line and operating properly.

3.0 Closed Loop Cooling Heat Exchanger Limitations

3.1 When placing a Turbine Building Closed Loop Cooling Heat Exchanger or a Reactor Building Closed Loop Cooling Heat Exchanger in service, establish Service Water flow BEFORE initiating Closed Loop Cooling flow. See N2-OP-14 and N2-OP-13 respectively.

3.2 Do not exceed 8 psid across the Service Water side of the Turbine Building or Reactor Building Closed Loop Cooling Heat Exchangers during normal operation. If high temperatures are experienced, service water flow can be increased to a max ΔP of 10 psid provided SWP flow limits specified in N2-OP-13 and N2-SOP-14 are not exceeded. If the above psid limits are exceeded, contact the System Engineer for resolution.

4.0 To prevent damage to Reactor Building Ventilation Supply Air Cooling Coil (2HVR-CLC2) due to freezing, the Service Water side of the Coil should be drained and blown down with air during the winter months. See N2-OP-52 for more information.

5.0 Do not allow a Service Water Pump to operate at less than 2,500 gpm for long periods of time or increased maintenance will be required.

D. PRECAUTIONS AND LIMITATIONS (Continued)

6.0 The Service Water System is flow balanced by throttling rate set valves at the outlet of all supplied components. Whenever the position of a rate set valve is changed, its original (throttled) position should be noted so that the valve can be returned to its original position and the overall system flow balance will not be affected.

7.0 Service Water Pumps should never be started until the strainer and discharge piping have been verified to be completely filled and vented.

8.0 If service water flow is established through an RHS heat exchanger without establishing flow through the RHS side of the heat exchanger, then periodic temperature monitoring of RHS heat exchanger outlet temperature shall be performed. Engineering has evaluated this condition and no primary containment pressure boundary component concerns provided the water in the shell side (RHS side) is maintained above 43°F. With RHS heat exchanger shell side temperature above 43°F and the remainder of the RHS system piping above 70°F (normal reactor building ambient), the permissible lowest service metal temperature (PLSMT) of 55°F at the primary containment penetrations will not be violated during a system initiation. This evaluation is documented in NER-2S-005 dated 09/29/04 and DER 2004-761.

9.0 A fire in Zone 803NZ Screenwell General Area could prevent automatic opening of 2SWP*MOV77A or 2SWP*MOV77B. Manual operation of these valves may be required. **[C2]**

10.0 When the number of running pumps is reduced to one in each Division, then 2SWP*MOV50A or 2SWP*MOV50B must be closed to prevent pump run out in the event one of the remaining Pumps is lost.

11.0 To open or close 2SWP*MOV50A or 2SWP*MOV50B, two switches must be operated on 2CEC*PNL601. The switches do not have to be held (signal seals in) and they can be operated in any order, but must be operated to the same direction for valve motion to occur.

12.0 Allowable Service Water System Configurations

12.1 Operational Conditions 1, 2, and 3:

- In Operational Conditions 1, 2, and 3; the six Service Water Pumps should be operated in a 2-2, 3-2, or 2-3 configuration.
- The Service Water System (SWP) design requirements in Operational Conditions 1, 2, and 3 require four operable and running service water pumps.
- Service Water Pumps may be operated in a 1-3 or 3-1 configuration for a short period of time for brief evolutions (such as surveillance testing or PMTs).
- With three operable and running SWP pumps, the plant shall restore 4 operable pumps to service in an expedited fashion. In this condition the plant is in a degraded state, see Tech Spec 3.7.1 for required actions.
- With two or less operable and running SWP pumps and Non-Essential loads not isolated, the plant is outside its design bases for a LOCA without LOOP, see Tech Spec 3.7.1.

D. PRECAUTIONS AND LIMITATIONS (Continued)

12.2 Operational Conditions 4 and 5, during movement of recently irradiated fuel assemblies in the secondary containment:

- In Operational Conditions 4 and 5, during movement of recently irradiated fuel assemblies in the secondary containment the six Service Water Pumps may be operated in any of the configurations listed below provided that the safety function of the following Isolation Valves are not inhibited, unless specifically authorized by procedure:
  - 2SWP*MOV50A, PMP 1A DISCH HEADER CROSS-TIE ISOL VLV
  - 2SWP*MOV50B, PMP 1B DISCH HEADER CROSS-TIE ISOL VLV
  - 2SWP*MOV3A, OUTLET TO TURBINE BLDG
  - 2SWP*MOV3B, OUTLET TO TURBINE BLDG
  - 2SWP*MOV19A, OUTLET TO REACTOR BLDG
  - 2SWP*MOV19B, OUTLET TO REACTOR BLDG
  - 2SWP*MOV93A, OUTLET FROM RX BLDG TO DISCH HEADER A
  - 2SWP*MOV93B, OUTLET FROM RX BLDG TO DISCH HEADER A
  - 2SWP*MOV599, OUTLET FROM TURB BLDG TO DISCH HEADER B
  - 2SWP*FV47A, Header A Cooling Tower Make-up Valve
  - 2SWP*FV47B, Header B Cooling Tower Make-up Valve
  - 2SWP*FV54A, Header A Cooling Tower Make-up Valve
  - 2SWP*FV54B, Header B Cooling Tower Make-up Valve
- Pump configurations:
  - 0-1, 0-2, 0-3
  - 1-0, 1-1, 1-2, 1-3
  - 2-0, 2-1, 2-2, 2-3
  - 3-0, 3-1, 3-2, 3-3
- and with less than four OPERABLE Service Water pumps available and in operation, the ability to maintain Service Water Divisional Cross-tie header pressure as indicated by SWPPA07 (08) or 2SWP*PI2A (B) greater than or equal to 63.5 psig ensures all safety related components will receive the required Service Water flow and prevents pump runout following auto initiation of LOCA loads.

D. PRECAUTIONS AND LIMITATIONS (Continued)

12.2 (Continued)

If Service Water Divisional Cross-tie header pressure is less than 63.5 psig with less than 4 OPERABLE Service Water pumps in operation for greater than one hour, the associated safety related equipment will be declared inoperable and actions required by the Technical Requirements Manual (TRM) are applicable.

13.0 Major valves in the Service Water System may be lined up in any configuration necessary to support operations providing the following limitations are adhered to:

13.1 The Automatic Safety Functions of Isolation Valves may be inhibited ONLY by specific procedural guidance. See Precaution and Limitation D.21.0 and Section H.11.0 for further direction.

13.2 An Operator should be stationed at the Cross-Tie Valves (2SWP*V17 and 2SWP*V32) if open, to close the Valve on Loss of Off-Site Power. These Valves being open would inhibit the Safety Function of 2SWP*MOV3A, 2SWP*MOV3B, 2SWP*MOV19A, 2SWP*MOV19B, and 2SWP*MOV599.

13.3 During Modes 1, 2 OR 3, 2SWP-V8, DISCH HDR CROSS-CONN, may be opened as required to provide an alternate discharge path for some of the non-safety related portions of the SWP System to support normal maintenance activities.

13.4 During Modes 4 OR 5, if 2SWP*V32 is open, 2SWP*MOV599 shall be closed OR the Division 3 Diesel 2EGS*EG2 shall be declared inoperable. 2SWP-V8 may be open with 2SWP*V32 open providing 2SWP*MOV599 is closed and 2SWP*MOV93B is operable. **[C6]**

14.0 Service Water Throttle Valve Position Clarification

- The conventions used for valve position are as follows:

-Full open	0° or 0%
-Full closed	90° or 100%
-Intermediate positions	x° or x%

- If the Valve Position Indicator is marked with degree increments, degrees are used.
- If only the open and closed positions are indicated, then an approximate percent is used.
- 2SWP*V998A through 2SWP*V998F, STRAINER BACK FLUSH VALVEs, are marked backwards from the above conventions. For these valves only, 0° is full closed and 90° is full open.
- 2SWP*V231A, EGS*EG1 OUTLET ISOL VLV, is a reverse acting valve but follows the above conventions.

15.0 During times when the Service Water System has only one division in service SCT Chemical Injection shall be terminated. **[C4]**

D. PRECAUTIONS AND LIMITATIONS (Continued)

16.0 If the following valves, or combination of the following valves, are open, then 2SWP*RE146A must be operable and operating, or compensatory grab samples must be taken: **[C3]**

- 2SWP*V995, DIV I RETURN HDR ISOL
- 2SWP*V996A, DIESEL GEN DISCHARGE HDR ISOL
- 2SWP*MOV93A, OUTLET FROM RX BLDG TO DISCH HEADER A, and 2SWP*MOV93B, OUTLET FROM RX BLDG TO DISCH HEADER A

17.0 If the following valves, or combination of the following valves, are open, then 2SWP*RE146B must be operable and operating, or compensatory grab samples must be taken: **[C3]**

- 2SWP*V992, DIV II RETURN HDR ISOL
- 2SWP*V996B, D/G DISCHARGE HDR ISOL
- 2SWP*MOV599, OUTLET FROM TURB BLDG TO DISCH HEADER B

18.0 Service Water Make-up to Circ Water

18.1 Do not open 2SWP*FV47A/B and 2SWP*FV54A/B so far that all Loop Flows are directed away from the Discharge Bay. The Discharge Standpipes must be kept full and flowing at all times to keep System full and prevent damage due to water hammer following a loss of all Service Water Pumps. As long as there is greater than or equal to 3,500 gpm indicated on 2SWP*FI567 (Loop A) and 2SWP*FI523 (Loop B) the Discharge Standpipes are full.

18.2 Maximum Cooling Tower Blow Down Flow is as directed by Chemistry or Environmental Protection.

19.0 If the Emergency Switchgear is being powered by the Diesel Generator, Service Water Pumps should be started per N2-SOP-03, Loss of AC Power.

20.0 IST of 2SWP*FV47B and 2SWP*FV54B and 2SWP*V202B **[IST]**

20.1 If Section H.10.0 is to be implemented to support valve maintenance activities, the affected valves must be tested per N2-OSP-SWP-Q004 (if due) BEFORE any maintenance.

21.0 Service Water Cross Tie Operation During Unit Outages

21.1 Any Service Water lineup that utilizes open cross tie valves 2SWP*V17 or 2SWP*V32, shall ensure that the ability exists to automatically stop flow through the affected Non-Essential header via at least one operable motor operated discharge valve, and the associated EDG of the MOV relied upon is also operable.

21.2 In the event of an SWP Non-Essential header line break with the respective manual supply cross tie valve(s) open, the Division III EDG must be declared inoperable until the Non-Essential header manual supply cross tie valve(s) are closed.

21.3 An operator or other qualified individual shall be stationed at manual valves 2SWP*V17 or 2SWP*V32 when opened, so that they can be immediately closed in the event of a break or a loss of Offsite Power.

D. PRECAUTIONS AND LIMITATIONS (Continued)

21.4 During Modes 4 OR 5, if 2SWP*V32 is open, 2SWP*MOV599 shall be closed OR Division 3 Diesel 2EGS*EG2 shall be declared inoperable. 2SWP-V8 may be open with 2SWP*V32 open providing 2SWP*MOV599 is closed and 2SWP*MOV93B is operable. **[C6]**

22.0 The heat load associated with three operating Service Water pumps in one pump bay may be slightly greater that the capacity of a single HVY unit cooler. Therefore, it may be necessary to place the second SWP pump bay unit cooler in service when operating three pumps in one bay continuously, due to long term EQ concerns. For short term operation, i.e., SWP pump rotation, placing second HVY unit cooler in service is not required. Refer to N2-OP-58, Subsection H.5.0, AND Attachment 1 of N2-OP-58. **[C5]**

23.0 The non-safety related Service Water header flow paths refers to the two supply and two return headers. Each flow path contains two isolation valves (a total of eight valves). In Operational Conditions 1, 2, or 3, when one isolation valve is inoperable, the affected flow path must be isolated by closing one of the isolation valves in the associated flowpath within 72 hours (such as 2SWP*MOV3A or 3B, 2SWP*MOV19A or 19B, 2SWP*MOV93A or 93B, or 2SWP*MOV599). In the case when 2SWP*MOV599 is inoperable and cannot be closed, 2SWP*MOV3A or 3B and 2SWP-V8 shall be closed. **[C5]**

24.0 The following limitations shall be observed under normal operating conditions: **[C5]**

- Maximum flow from any single Service Water pump shall not exceed 10,000 gpm
- Service Water pump motor amps shall not exceed 76 amps

25.0 Following a LOCA, if four or less SWP pumps are available, to supply greater than 2000 GPM SWP flow to one or both RHR Heat Exchangers, it may be necessary to isolate Turbine Building loads by closing 2SWP*MOV3A OR *MOV3B OR *MOV599 AND 2SWP-V8. This is only required if more than four SWP pumps can not be placed in service. **[C5]**

26.0 As an operator good practice, the service water pump supply breaker should be checked properly recharged following normal opening operation. Refer to N2-ELU-01 Subsection 3.4.

27.0 The essential components cooled by the Service Water System are designed for a maximum supply header temperature of 84°F, therefore 84°F is the design basis analytical limit (AL), which contains no instrument uncertainty allowance. See table below for uncertainty values and maximum allowable readings. The reading can exceed 77.5°F on 2SWP*TI31A/B provided the associated computer point, SWPTA01/02, is less than or equal to 82.37°F. Additionally, the reading can exceed 77.5°F on 2SWP*TI31A/B and the associated computer point SWPTA01/02 may exceed 82.37°F provided the measured temperature at 2SWP*TT31A/B does NOT exceed 82.93°F. Specific instructions for obtaining temperature at 2SWP*TT31A/B is provided in the shift checks.

FUNCTION	°F Calculated Uncertainty	Max. Allowable Reading
Indicators 2SWP*TI31A/B	± 2.81°F	77.5°F
Comp. Pts. SWPTA01/02	± 1.63°F	82.37°F
2SWP*TT31A/B	± 1.07°F	82.93°F

D. PRECAUTIONS AND LIMITATIONS (Continued)

28.0 As a response to DER NM-2002-4947, any time the Missile Barrier over 2SWP*MOV30A/B is removed, both valves shall be declared inoperable and one of the two valves shall be closed (the Safety Function position). Per the discussion in DER NM-2002-4947, this does not cause either division of Service Water to be inoperable. Refer to DER NM-2002-4947 and Letter to File #97030 for further discussion on Operability of the Ultimate Heat Sink and Reportability concerns with the Missile Barrier removed. **[C7]**

29.0 In the event that flooding occurs within the Turbine Building, make an immediate assessment to ensure that the source of water is known. If it has been determined that a rupture of the condenser expansion joint or pipe break has occurred, Isolate Service Water make-up to the cooling tower (via 2SWP*FV47/54A&B) within 48 minutes to prevent water from exceeding the 257' elevation and affecting safe shutdown equipment. **[C8]**

30.0 When adverse weather conditions are encountered, the following website may provide detailed environmental data for Lake Ontario including wind speeds, lake temperatures, and water current speeds and direction: **[C9]**

"http//www.glerl.noaa.gov/res/glcfs/"

This website is maintained by the National Oceanic and Atmospheric Administration (NOAA)

31.0 Conditions affecting or projected to affect Lake Ontario that has led to or could lead to Service Water Intake Blockage should be reflected in the Probabilistic Risk Assessment. The Activities input function of the EOOS Program provides Lake Blockage selections to be used to model this risk.

32.0 PPE Hazard Assessment was performed for the task of Trash Rake Bin Cleaning (CR-2009-004752). The following PPE requirements exist when cleaning the Trash Rake Bins:

- Hard Hat
- Safety Glasses
- ½ Face Respirator with Organic Filters (workers may choose to wear full face respirator with organic vapor cartridges based on comfort).
- Long sleeved shirt
- Work gloves (specifically leather)

33.0 Service Water System and FLEX Dry Hydrant Connection

- With circulating water pump(s) and the tempering line in service (CWS pumps in service and 2SWP-V902 open), pressure will exist up to the dry hydrant caps. Do Not remove dry hydrant caps under these conditions.
- If failure of the freeze protection system is detected, consider deploying a temporary heat source until freeze protection system function can be restored.
- Following a loss of freeze protection system, dry hydrant dust caps should be inspected for damage that may have occurred during the loss of heating.
- When valve 2SWP-V904 is closed, FLEX dry hydrants become non-functional unless an alternate flow path is aligned. Refer to CC-NM-118-01 for required actions.

34.0 If shutting down a fifth running Service Water Pump verify two or Less CCP Heat Exchangers are in service and that two or Less CCS Heat Exchangers are in service.

F. NORMAL OPERATIONS (Continued)

9.0 Tempering Service Water with Circ Water

NOTE

- This Subsection is normally used for tempering Intake water when the Plant is operating and Circ Water tempering is available.
- Tempering is normally started when Intake Bay Temperature is less than or equal to 38°F and stopped when Intake Bay Temperature is consistently greater than 38°F. Tempering is allowed when Intake Bay Temperature is less than or equal to 45°F.

9.1 Verify 2SWP*V900, DISCHARGE ISOLATION TO TEMPERING, closed.

9.2 IF any of the following Radiation Monitors are inoperable, THEN verify Chemistry is performing grab samples every 12 hours per ODCM D 3.3.1 Condition D:

- 2CWS-RE157, Cooling Tower Blowdown Radiation Monitor
- 2SWP*RE146A, Service Water Effluent Line A Radiation Monitor
- 2SWP*RE146B, Service Water Effluent Line B Radiation Monitor

NOTE

Remaining steps in this section may be performed while monitoring for conditions listed in Step F.9.3.

CAUTION

Part of the Tempering Flow will return directly to the Lake without passing 2CWS-RE157, 2SWP*RE146A, or 2SWP*RE146B by going through the Fish Jet Pump. However, any activity present will be indicated by one of these Radiation Monitors or grab samples. Failure to stop Tempering if any of the Radiation Monitors indicates high activity, may result in a possible unmonitored release.

9.3 IF any of the following conditions exist, THEN stop Tempering per Subsection F.11.0:

- Any of the following indicate high activity:
  - 2CWS-RE157
  - 2SWP*RE146A
  - 2SWP*RE146B

F. NORMAL OPERATIONS (Continued)

9.3 (Continued)

- Grab Samples of Service Water OR Circulating Water Effluent
- Intake Bay Temperature is greater than 45°F
- Direction by Chemistry due to CWS chemistry problems OR slug chemical additions
- Cooling Tower Blowdown is stopped for any reason

9.4 Inform Chemistry that Service Water Tempering with Circ Water will be performed.

NOTE

Manual Control of Tempering is the normal method of Tempering Flow control, however Automatic Control may also be used.

9.5 For Manual Control of Tempering, perform the following:

9.5.1 Place 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW, in M (Manual).

NOTE

This Controller works in reverse of most controllers in that the manual slider must be taken to the right to close the valve. 100% Controller Output is maximum Close Demand.

9.5.2 Using 2SWP-TIK512 manual slider, close 2SWP-TV512.

9.5.3 Open 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT.

NOTE

This Controller works in reverse of most controllers in that the manual slider must be taken to the left to open the valve. 0% Controller Output is maximum Open Demand.

9.5.4 Using 2SWP-TIK512 manual slider, open 2SWP-TV512 to establish desired Tempering Flow less than or equal to 5000 gpm as indicated on 2SWP-FI511, INTAKE TEMPERING FLOW, OR NOT to exceed limit provided by Environmental Protection Department.

NOTE

Manual Control of Tempering is the normal method of Tempering Flow control, however Automatic Control may also be used.

9.6 For Automatic Control of Tempering, perform the following:

9.6.1 Verify 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW, in Automatic as indicated by light lit on the A pushbutton.

9.6.2 Verify 2SWP-TIK512 is set to desired temperature AND is set less than or equal to 38°F.

9.6.3 Open 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT.

10.0 Tempering Service Water with Service Water

NOTE

- This procedure is normally used for Tempering during cold weather operation when the Plant is shut down and Tempering with Circ Water is unavailable.
- Tempering is normally started when Intake Bay Temperature is less than or equal to 38°F and stopped when Intake Bay Temperature is consistently greater than 38°F. Tempering is allowed when Intake Bay Temperature is less than or equal to 45°F.

10.1 Inform Chemistry that Service Water Tempering with Service Water will be performed.

10.2 Verify 2SWP-V902, CIRC WATER TO TEMPERING CUT OUT, closed.

NOTE

The following Steps will place Tempering Temperature Control in automatic, however when Lake Temperature is low, Service Water Tempering will not raise temperature significantly.

10.3 Verify 2SWP-TIK512, SCREENWELL INTAKE TEMPERING FLOW, in A (Automatic).

10.4 Verify 2SWP-TIK512 is set to desired temperature AND is set less than or equal to 38°F.

10.5 Open 2SWP*V900, DISCHARGE ISOLATION TO CUT OUT.

Constellation

Training ID: **Nov 2023 NRC JPM G** Revision: **0.0**

Title: **Transfer Cooling Tower from Mode I to Mode II, Then Mode II to Mode III**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A		Dan Delaney	7/12/23
Facility Reviewer:	Signature on File		John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-10A, Circulating Water System
2. NUREG 1123 K/A 510001 A4.02 (3.1)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the operator's ability to transfer Cooling Tower modes.

   b. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-275000-01003, Operate (Monitor) Cooling Tower

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A	
Evaluation Method:	☒ Perform ☐ Simulate				
Evaluation Location:	☐ Plant ☒ Simulator ☐ Classroom				
Time Critical Task:	☐ Yes ☒ No				
Alternate Path:	☐ Yes ☒ No				
Safety Function:	8		Plant Service Systems		
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	510001 Circulating Water System A4.02 - Ability to manually operate and/or monitor in the control room: Circulating water valves				
K/A Importance Rating:	RO	3.1	SRO	-	

4. Recommended Start Location

   a. NLC Classroom

5. Simulator Setup

   a. IC Number

      1) IC-021

   b. Presets / With Triggers

      1) Malfunctions

         a) None

      2) Remotes

         a) **MT01**, Ambient Outside Air Temp, Final = 15 **Inserted**

         b) **MT03**, Lake Ontario Temperature, Final = 40 **Inserted**

      3) Overrides

         a) None

      4) Annunciators

         a) None

      5) Event Triggers

Event #	Event Action	Command
None		

      6) Equipment Out of Service

         a) None

      7) Support Documentation

         a) Prepare a copy of N2-OP-10A, Section F.2 with no steps placekept.

      8) Miscellaneous

         a) IC-204 (For ILT 22-2, paired with JPM A)
         -OR-

         b) IC setup:

(1) Reset to IC-021

6. Strategy Code

   a. None

7. Tools and Equipment

   a. None

8. Commitments

   a. None

9. Prerequisites

   a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Ambient air temperature is 15°F and lowering. • The Cooling Tower is operating in Mode I. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *transfer the Cooling Tower from Mode I to Mode II in accordance with N2-OP-10A section F.2.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	Transferring from Mode I to Mode II, Ice Prevention Operation, should be performed when: • Within the operation band of Mode II, or • Significant ice accumulation, or • Maintain Hotwell temperature greater than 100 °F (see Step D.22) • Support online Cooling Tower maintenance per section H.10		
2.	**VERIFY** Cooling Tower is in Mode I, Normal Operation	P (F.2.2.a)	SAT / UNSAT ***STD:*** *Determines Cooling Tower is in Mode I from initial conditions and/or from observing panel indications*
3.	**OPEN** the following valves: **Note:** The following bulleted JPM steps can be performed in any order or concurrently.	P (F.2.2.b)	

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.a	• 2CWS-MOV53A, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53A by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53A green light off, red light on NOT achieved*
3.b	• 2CWS-MOV53B, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53B by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53B green light off, red light on NOT achieved*
3.c	• 2CWS-MOV53C, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53C by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53C green light off, red light on NOT achieved*
3.d	• 2CWS-MOV53D, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53D by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53D green light off, red light on NOT achieved*
3.e	• 2CWS-MOV53E, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53E by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53E green light off, red light on NOT achieved*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.f	• 2CWS-MOV53F, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53F by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53F green light off, red light on NOT achieved*
3.g	• 2CWS-MOV53G, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53G by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53G green light off, red light on NOT achieved*
3.h	• 2CWS-MOV53H, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53H by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53H green light off, red light on NOT achieved*
3.i	• 2CWS-MOV53J, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53J by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53J green light off, red light on NOT achieved*
3.j	• 2CWS-MOV53K, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53K by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53K green light off, red light on NOT achieved*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.k	• 2CWS-MOV53L, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53L by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53L green light off, red light on NOT achieved*
3.l	• 2CWS-MOV53M, COOLING TOWER DE-ICING VLV	P (F.2.2.b)	***PASS / FAIL** ***STD:*** *Opens 2CWS-MOV53M by rotating control switch clockwise to OPEN* ***Failure =*** *2CWS-MOV53M green light off, red light on NOT achieved*
Evaluator Note:	Once the Operator has placed the Cooling Tower in Mode I, new conditions and a new cue will be given (see below) to place the Cooling Tower in Mode II.		
Evaluator Cue:	*Ambient temperature is now -3^{0}F and lowering. Transfer the Cooling Tower from Mode II to Mode III in accordance with N2-OP-10A section F.2.*		
Procedure Note:	Transferring from Mode II to Mode III, Zoned Ice Prevention Operation, should be performed when: • Within the operation band of Mode III, or • Significant ice accumulation, or • Maintain Hotwell temperature greater than 100 °F (see Step D.22)		
Procedure Caution:	2CWS-MOG54E must be left full open to prevent flooding at 2CWS-MOG54E Gate House.		
4.	**VERIFY** Cooling Tower is in Mode II, Ice Prevention Operation	P (F.2.3.a)	SAT / UNSAT ***STD:*** *Determines Cooling Tower is in Mode II from previous JPM actions and/or from observing panel indications*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
5.	**POSITION** the following gates to Mid-Position (Blue light lit) by momentarily placing their respective control switches to CLOSE: **Note:** The following bulleted JPM steps can be performed in any order or concurrently.	P (F.2.3.b)	
5.a	• 2CWS-MOG54B, COOLING TOWER INLET DISTRIBUTION GATE VLV	P (F.2.3.b)	***PASS / FAIL** ***STD:*** *Partially closes 2CWS-MOG54B by momentarily rotating control switch counterclockwise to CLOSE and then releasing the control switch back to the NEUTRAL position* ***Failure =*** *2CWS-MOG54B blue light on NOT achieved and sustained*
5.b	• 2CWS-MOG54C, COOLING TOWER INLET DISTRIBUTION GATE VLV	P (F.2.3.b)	***PASS / FAIL** ***STD:*** *Partially closes 2CWS-MOG54C by momentarily rotating control switch counterclockwise to CLOSE and then releasing the control switch back to the NEUTRAL position* ***Failure =*** *2CWS-MOG54C blue light on NOT achieved and sustained*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
5.c	• 2CWS-MOG54D, COOLING TOWER INLET DISTRIBUTION GATE VLV	P (F.2.3.b)	***PASS / FAIL** ***STD:*** *Partially closes 2CWS-MOG54D by momentarily rotating control switch counterclockwise to CLOSE and then releasing the control switch back to the NEUTRAL position* ***Failure =*** *2CWS-MOG54D blue light on NOT achieved and sustained*
6.	**VERIFY** the blue Mid-Position indicating lights are illuminated for their respective gates	P (F.2.3.c)	SAT / UNSAT ***STD:*** *Observes blue light lit for 2CWS-MOG54B, 2CWS-MOG54C, and 2CWS-MOG54D*
Evaluator Note:	*After the Operator has verified the blue Mid-position indicating lights are illuminated for the three required gates, provide the following cue:* ***Cue:*** *Your task is complete, another operator will complete any remaining actions.*		

TASK STANDARD	*Cooling Tower has been transferred from Mode I to Mode II, and then from Mode II to Mode III.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Ambient air temperature is 15°F and lowering. • The Cooling Tower is operating in Mode I.

INITIATING CUE	**(Operator Name)**, *transfer the Cooling Tower from Mode I to Mode II in accordance with N2-OP-10A section F.2.*

Simulator JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant is at 100% power. • Ambient air temperature is 15°F and lowering. • The Cooling Tower is operating in Mode I. **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *transfer the Cooling Tower from Mode I to Mode II in accordance with N2-OP-10A section F.2.*

NINE MILE POINT NUCLEAR STATION UNIT 2
OPERATING PROCEDURE

N2-OP-10A

CIRCULATING WATER SYSTEM

Revision 02300

Print in COLOR

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director Operations

SUMMARY OF ALTERATIONS

PCR-21-00788:

- Description: Stop Service Water (SWP) Tempering during and for at least 1 hour after wood flour additions.
- Reason: SWP Tempering with CWS returns CWS to the Service Water Intake, causes SWP strainer clogging.
- D.28 added SWP Tempering with CWS.
- H.16.1, H.17.1, H.18.1, H.19.1, H.20.1, and H.21.1 preceding caution added SWP Tempering with CWS.
- H.16.8, H.17.8, H.18.8, H.19.8, H.20.8, and H.21.8:
    - Moved CWS manual valve preceding note and step to isolate CWS Blowdown to where it is sequentially needed, after all preparatory steps are complete
    - Deleted unnecessary Chemistry notification already directed in referenced Section F.8.
- H.16.9, H.17.9, H.18.9, H.19.9, H.20.9, and H.21.9 added conditional step to isolate SWP Tempering with CWS, if in service.
- H.16.13, H.17.13, H.18.13, H.19.13, H.20.13, and H.21.13 added recording Wood Flour Injection Completion Time.
- H.16.14, H.17.14, H.18.14, H.19.14, H.20.14, and H.21.14 added control room notification of Wood Flour Injection Completion Time.
- H.16.19, H.17.19, H.18.19, H.19.19, H.20.19, and H.21.19 3rd bullet revised to record total active wood flour injection start and stop times.
- H.16.21, H.17.21, H.18.21, H.19.21, H.20.21, and H.21.21:
    - Moved step to be last in the Section
    - Added 1-hour wait before restoring CWS Blowdown and/or SWP Tempering with CWS.
    - Added restoration of SWP Tempering with CWS,
    - Deleted unnecessary Chemistry notification already directed in referenced Section F.8.

Procedure Writer Review:

- F.1.1 3rd bullet removed “automatic **OR**”, as CWS blowdown is always controlled in manual to comply with SPDES Permit flows.
- F.1.3 removed previous revision unnecessary “as directed by SM” wording.
- F.8.2.c reworded previous revision “**VERIFY** lack of flow indication” to “**VERIFY** 0 gpm”.
- F.8.3.c reworded previous revision “**VERIFY** lack of flow indication” to “**VERIFY** 0 gpm”.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 5
B. SYSTEM DESCRIPTION 8
C. OPERATING REQUIREMENTS 10
D. PRECAUTIONS AND LIMITATIONS 11
E. STARTUP 15
E.1. Filling Cooling Tower Basin 15
E.2. Use of Temporary Vacuum Pump Priming for CWS Pump Start 22
E.3. CWS Pump Start with One or More Pumps Operating Using Blowdown Valve and Water Box Priming Pump 2CWS-P3 [EOP] 36
F. NORMAL OPERATIONS 46
F.1. Normal Operating Indications 46
F.2. Cooling Tower Operations 47
F.3. Establishing Manual Blowdown for Basin Level Control 54
F.4. Securing Manual Blowdown 57
F.5. Establishing Automatic Blowdown for Basin Level Control 60
F.6. Securing Automatic Blowdown 62
F.7. Screenwash Pump Operation 64
F.8. Isolating/Restoring CWS Blowdown and/or SWP Tempering for Chemical Addition 66
G. SHUTDOWN 69
G.1. CWS Pump Shutdown with More Than One in Service and Turbine Offline 69
G.2. CWS Pump Shutdown with Only One in Service 72
H. OFF-NORMAL OPERATIONS 75
H.1. Draining to Lake Level 75
H.2. Draining Below Lake Level 78
H.3. CWS Pump Shutdown with Turbine On Line 82
H.4. Foaming for Condenser Tube Leak Identification 86
H.5. Removing Seal Water Pressure Control Valve From Service 88
H.6. Returning Seal Water Pressure Control Valve to Service 90
H.7. Overriding the Basin Water Low Temperature Interlock 91
H.8. Maintaining Hotwell Temperature with the Main Turbine Off-Line 93
H.9. Raising Basin Water Temperature by Opening a Single Bypass Gate Valve 94
H.10. Isolating a Cooling Tower Zone to Perform Maintenance On-Line 97
H.11. Isolating a Single Cooling Tower Zone to Perform Maintenance On-Line 107
H.12. Actions in Event of Cooling Tower Basin Overflow 110

TABLE OF CONTENTS (Continued)

SECTION **PAGE**

H.13. CWS Pump Start with No Pumps Running Using Jet Motive Pump Method [EOP] 111
H.14. CWS Pump Start with One or More Pumps Operating Using Blowdown Valve [EOP]. 118
H.15. CWS Pump Start with One or More Pumps Operating with a Degraded (leaking) Waterbox 125
H.16. Injecting Wood Flour into the A Waterbox 133
H.17. Injecting Wood Flour into the B Waterbox 138
H.18. Injecting Wood Flour into the C Waterbox 143
H.19. Injecting Wood Flour into the D Waterbox 148
H.20. Injecting Wood Flour into the E Waterbox 153
H.21. Injecting Wood Flour into the F Waterbox 158
Attachment 1, Foam Generator Diagram..... 163
Attachment 2, CWS Stop Log Configuration 164
Attachment 3, Data Sheet for Securing CWS Pump with Turbine Online 165
Attachment 4, Cooling Tower Operation Curve 166
Attachment 5, CWS-P3 Generic Flowpath 167
Attachment 6, Temporary Vacuum Pump Generic Flowpath 168
Attachment 7, Cooling Tower Zoning 169
Attachment 8, Materials for Wood Flour Injection 170

A. REFERENCES AND COMMITMENTS

A.1. Technical Specification

None

A.2. Licensee Documentation

- Updated Safety Analysis Report (USAR), Volume 24, Section 10.4.5, Circulating Water System
- Offsite Dose Calculation Manual (ODCM), D3.3.1, Radioactive Liquid Effluent Monitoring Instrumentation.

A.3. Policies, Programs, and Procedures

- N2-OP-10B, Acid Treatment/Hypochlorite
- N2-OP-11, Service Water System
- N2-OP-14, Turbine Closed Loop Cooling System
- N2-OP-15, Makeup Water Treatment
- N2-OP-19, Instrument and Service Air System
- N2-OP-21, Main Turbine System
- N2-OP-58, Screenwell Building and Fire Pump Rooms Ventilation Systems
- N2-OP-64, Turbine Building Drains
- N2-OP-71A, 13.8KV AC Power Distribution
- N2-OP-71B, 4.16KV AC Power Distribution
- N2-OP-71C, 600V AC Power Distribution
- N2-OP-73A, Normal DC Distribution
- N2-OP-79, Radiation Monitoring
- N2-OP-91A, Process Computer
- N2-OP-101D, Power Changes

A. REFERENCES AND COMMITMENTS (Continued)

A.4. Technical Information

A.4.1. Flow Diagrams

P&ID 10A through 10H, Circulating Water, Acid & Hypochlorite

A.4.2. Electrical Diagrams

- ESK-5CWS01-06, Circ Water Pump 2CWS-P1A-F
- ESK-6CWS01-06, 09-16, 24-27, Circulating Water System
- ESK 7CWS01, Circulating Water SystemLogic Diagrams
- LSK 2-1.A, Circulating Water
- LSK 2-1.B, Circulating Water
- LSK 2-1.D, Circulating Water
- LSK 2-1.E, Circulating Water
- LSK 2-1.F, Circulating Water
- LSK 2-1.K, Circulating Water
- LSK 2-1.L, Circulating Water
- LSK 2-1.M, Circulating Water

A.4.3. Vendor Manuals

GEK 49966, Vol 1, Turbine Section

A. REFERENCES AND COMMITMENTS (Continued)

A.5. Supplemental References

- File Code SM2-M89-0134, PR09063, Circulating Water System 5 Pump Operation
- File Code SM2-M89-0150, Internal Correspondence, CWS/5 Pump Operation, September 19, 1989
- SE 91-093, Defeat Low Level Interlock for Pump Start
- SE 92-066, Defeat Low Suction Pressure Interlock for CWS Pumps
- SE 96-028, Cooling Tower Bypass Gate Logic Changes
- File Code NMP93442, Letter from C. Merrit to R.G. Smith, Unit 2 CWS Blowdown/Tempering Recommendations, January 2, 1997
- DER 2-95-1787, Potential Overflow of Cooling Tower Basin due to Possible Inappropriate Level Band
- DCP N2-05-111, Implement 84 °F UHS Temperature Without 82 °F Averaging (Tech Spec Amendment 119)
- ECP-11-000729, NMP2 CWS Chemical Treatment
- ECP-10-000816, TMA System EPU Instrumentation Changes

A.6. Commitments

Sequence Number	Commitment Number	Description
1	DER 2-95-1633 (IR 02073859) LER 95-04	Unmonitored Release Via Div II SWP
2	CR 2006-3010 (IR 02026019)	Turbine Building flooding analysis USAR 10.4.5.3

B. SYSTEM DESCRIPTION

The purpose of the Circulating Water (CWS) System is to provide the Main Condenser with a continuous supply of cooling water. The circulating water is used to remove heat rejected from the turbine exhaust and turbine bypass steam, as well as from other equipment, over the full range of operating loads.

The CWS System is designed as a closed loop system using a natural draft Cooling Tower to reject waste heat to the atmosphere. This heat is acquired from the Main Condenser via six circulating water pumps. At rated turbine-generator load, the CWS design temperature differential through the Main Condenser is approximately 27 °F.

Six circulating water pumps normally operate continuously to meet the design heat transfer rate of the Main Condenser (7.9×10^9 Btu/hr). The CWS pumps have the capacity to supply additional flow over the system design flow of 580,000 gpm to the Main Condenser. Based on operating test data, the CWS pumps are capable of pumping 680,000 gpm at a head of 66 feet. The extra pumping capacity is designed to be used to balance flows through the six main condenser paths.

The CWS System is designed to assist in maintaining condenser vacuum during reduced loads with only four or five pumps operating and the matching four or five main condenser paths open. By design, acceptable main condenser vacuum can be maintained at approximately half load with only three pumps operating and half the main condenser tubes in operation. Three pumps are capable of pumping 345,000 gpm at a head of 63 feet.

Blowdown flow rate from the CWS System is controlled to maintain the closed loop chemical concentrations ≤ four times the lake water concentration. The blowdown rate is dependent on the circulating water level in the cooling tower discharge flume and water chemistry parameter limits of the State Pollutant Discharge Elimination System (SPDES) Permit. Water chemistry is maintained by addition of a dispersant and scale inhibitor. Hypochlorite and Control Brom (stabilized bromine) will be batch applied for bio-fouling control.

In order to eliminate the buildup of ice on the cooling tower fill material, the Cooling Tower can be sectionalized and/or an ice prevention ring can be utilized. Both sectionalizing and the use of the ice prevention ring tend to lower the efficiency of the Cooling Tower, thus the temperature to which CWS water is cooled remains higher than normal. Each of the supply flumes is supplied with CWS water from the supply channel through a motor-operated slide gate (2CWS-MOG54A-F). The slide gates can be systematically closed to stop CWS water flow to the fill sections in the center section of the tower.

Twelve hot water distribution pipes are provided with 18-inch motor-operated butterfly ice prevention valves (2CWS-MOV53A-H, J-M), which are attached to the flume. The ice prevention valves are connected to a peripheral ice prevention ring. The peripheral ice prevention ring is a perforated 14-inch pipe which encircles the entire inside of the Cooling Tower just above the air inlet. As ice prevention valves are opened, hot CWS water is sprayed down over the air inlet, creating a veil. This tends to heat the incoming air and lowers the amount of air entering the tower. The CWS water falling down through the fill is cooled less and the cooling tower basin water temperature rises.

B. SYSTEM DESCRIPTION (Continued)

If utilization of the ice prevention ring and sectionalizing prove inadequate in maintaining the basin temperature high enough, three motor-operated bypass gates (2CWS-MOG52A, B, C) open to bypass hot CWS water flow directly to the Cooling Tower basin. 2CWS-MOG52A, B, C are located at the base of the three respective risers carrying flow from the hot water inlet to the concrete supply flumes. 2CWS-MOG52A, B, C can be opened together when a master control switch is held in the open position.

The Cooling Tower has four modes of operation. Mode I is normal operation (full cooling) and consists of having the bypass gates closed, de-icing valves closed, and the distribution gates open. Mode II is for ice prevention and consists of having the bypass gates closed, de-icing valves open, and the distribution gates open. Mode III is for zoning and consists of having the bypass gates closed, de-icing valves open, and distribution gates 2CWS-MOG54B-D in the mid-position. Mode IV is for full bypass with the bypass gates being full open.

The Cooling Tower must be operated in the appropriate mode to prevent ice damage while maximizing thermal performance. Attachment 4, Cooling Tower Operation Curve, is to be used in determining the correct mode of operation for existing air temperature and heat load (Reactor power). As the air temperature changes, a mode change is required before reaching the bottom or top of each mode band. For instance, at full heat load with **no** ice accumulation, entry into Mode II (Ice Prevention Operation) may occur as low as -2 °F. A mode change is also immediately required upon ice accumulation that impacts thermal performance or poses a threat of damage to the tower.

Two 30-inch service water lines penetrate the pipe provided with 2CWS-FE/FT47A. These lines provide Service Water makeup to the CWS System to compensate for evaporative and blowdown losses up to a rate of 25,000 gpm.

A screenwash pump (2CWS-P2) is installed in the Screenhouse Building for cleaning of the screens in the cooling tower basin discharge flume. The pump discharge has three 3/4" hose connections. Level transmitters (2CWS-LIT66A and B) measure water level downstream of the screens and compare it to level upstream of the screens. If a high differential level is indicated, an alarm is activated which indicates the screens are full of debris and should be cleaned.

C. OPERATING REQUIREMENTS

C.1. The following systems must be in operation to support the CWS System:

- N2-OP-10B, Acid Treatment/Hypochlorite
- N2-OP-11, Service Water System
- N2-OP-14, Turbine Closed Loop Cooling System
- N2-OP-15, Makeup Water Treatment
- N2-OP-19, Instrument and Service Air System
- N2-OP-58, Screenwell Building and Fire Pump Rooms Ventilation Systems
- N2-OP-64, Turbine Building Drains
- N2-OP-71A, 13.8KV AC Power Distribution
- N2-OP-71B, 4.16KV AC Power Distribution
- N2-OP-71C, 600V AC Power Distribution
- N2-OP-73A, Normal DC Distribution
- N2-OP-79, Radiation Monitoring
- N2-OP-91A, Process Computer

D. PRECAUTIONS AND LIMITATIONS

D.1. Chemistry shall be notified of any of following:

- Prior to draining the Cooling Tower Basin, sample requirements must be met and be within specifications.
- Prior to starting CWS System blowdown, sample requirements must be met and be within specifications.
- Prior to starting or stopping CWS Pumps, appropriate chemical feed rate adjustments must be made.
- When blowdown rate changes more than 1000 GPM, appropriate chemical feed rate adjustments must be made.
- When blowdown is secured, appropriate chemical feed rate adjustments must be made.
- When 2CWS-RE157 is inoperable and grab samples are required.

D.2. Due to copper concerns, Environmental Protection shall be notified prior to commencing CWS System draining or initial blowdown.

D.3. Radiation Protection shall be notified of any of following:

- Prior to starting CWS System blowdown, 2CWS-RE157 is required to be in service.
- When blowdown is secured and radiation monitoring is **no** longer required.

D.4. CWS Pump start limitations are as follows:

- Two starts in succession from ambient temperature or one start from rated temperature.
- For subsequent starts, allow 30 minutes running time or 60 minutes idle time.

D.5. A 3-minute interval is required between starting or stopping additional pumps.

D.6. When starting the first CWS Pump with basin water temperature less than or equal to 40 °F, it may be necessary to defeat the basin water low temperature interlock to permit placing 2CWS-MOG52A (B,C), COOLING TOWER BYPASS GATE VLV, in mid-position.

D.7. Operate 2CWS-MOG52A (B,C), COOLING TOWER BYPASS GATE VLV, in mid-position within the following limitation the first CWS Pump start when **no** other pumps are running.

D.8. When filling waterboxes using the priming pump, suction strainer clogging may be indicated by having good vacuum at pump suction and poor vacuum at drain valve upstream of the suction strainer.

D.9. If 2CWS-AOV21A (B-F), Water Box A (B-F) Priming Line Isol Vlv, leaks by while filling condenser waterbox, then, the associated waterbox may be prevented from being filled.

D.10. When the ice prevention ring is in service, 2CWS-MOV53A (B-H, J-M), COOLING TOWER DE-ICING VLV, must be full open to prevent ice-up.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.11. When the ambient air temperature is less than 35 °F, the Cooling Tower basin water temperature should be monitored. Basin water temperature must be maintained above 45 °F.

D.12. If CWS blowdown flow control is in Manual, then, Cooling Tower basin level AND Service Water makeup flow to the CWS System must be closely monitored to prevent basin overflow or loss of CWS Pump suction.

D.13. If CWS blowdown is secured, it will be necessary to adjust Service Water makeup flow to the CWS System to prevent Cooling Tower basin overflow.

D.14. CWS System requirements for Main Turbine operation:

- At least one flow path in each condenser.
- Vacuum differential between any two condensers is less than 2" Hg.
- Condenser vacuum greater than or equal to 24.6" when turbine load is less than or equal to 410 MWE (30%).

D.15. After steam has been secured to the Main Condenser, a two-hour wait should be performed before securing the CWS System to permit temperatures to equalize across the condenser.

D.16. A CWS Pump shall be secured prior to its associated 2CWS-M0V5A (B-F), CONDENSER A(B,C) OUTLET VLV, traveling more than 85% closed.

D.17. When draining the CWS System using installed sump pumps, drainage flow should be within capacity of operating sump pumps to prevent flooding the CWS Pump pit.

D.18. During prolonged CWS Pump outages, cooling water to pump coolers should be isolated and the associated pump control switch placed in PULL-TO-LOCK. A Tagout must be attached to isolation valves and pump control switch.

D.19. Maximum full load current on a running CWS Pump is 110 amps.

D.20. When the Stop Logs are to be left in a suspended condition, ensure they are properly dogged and wedged. To be properly dogged, ensure that the associated chock is in place on the dog to prevent canting of the dog. Wooden wedges are also provided to be placed between the Stop Log and the floor opening edge to prevent lateral movement of the Stop Log.

D.21. Radiation monitor 2CWS-RE157 must be operable and in service or compensatory actions must be taken if one or both of the following valves are open: **[C1]**

- 2CWS-V10, 2CWS-LV140 Outlet Isol Vlv
- 2CWS-V11, 2CWS-LV140 Bypass Isol Vlv

D.22. At power, main condenser hotwell temperatures less than about 100 °F can have an adverse impact on condensate demineralizers (release of iron into feedwater). Therefore, at the SM direction, it is permissible to place the cooling tower in a mode that provides less cooling prior to actually entering the range specified on Attachment 4, Cooling Tower Operation Curve.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.23. During periods of high temperature, to facilitate lower hotwell temperature, or when otherwise requested, cooling tower blowdown may be raised above 6,000 gpm provided the following are met:

- Chemistry management concurrence is obtained.
- Blowdown flow does **not** exceed 15,000 gpm.
- Raised blowdown periods do **not** exceed 2 consecutive weeks when blowdown is greater than 8,000 gpm or as otherwise directed by Chemistry Supervision.
- SPDES discharge temperature limit of 110 °F is **not** exceeded.

D.24. In the event that flooding occurs within the Turbine Building, make an immediate assessment to ensure that the source of water is known. If it has been determined that a rupture of the condenser expansion joint or pipe break has occurred, Isolate Service Water make-up to the cooling tower (via 2SWP*FV47/54A & B) within 48 minutes to prevent water from exceeding the 257' elevation and affecting safe shutdown equipment. **[C2]**

D.25. If condensate demineralizer resin processing is in progress following a cooling tower mode change, ensure resin backwash and URC flowrates are set properly during the resultant hotwell temperature changes.

D.26. When Shutdown and some CWS pumps are secured, waterbox D/P will increase. This is due to reduction in overall systems resistance. The reduction in system resistance will cause flow rate and consequently D/P to increase through the waterbox. During this condition it is allowable for waterbox D/P to exceed 8 psid.

D.27. Starting or securing of a CWS pump will cause changes in actual water level in waterboxes with the associated CWS pump is **not** in service. This could lead to personnel injury inside the waterbox. Prior to starting or securing a CWS pump, verify that **no** personnel are physically within any waterbox that has been opened for access.

D.28. Clogging of CWS instrument lines may occur during and for 1 hour after application of wood flour (sawdust) injection if surveillance testing, or clogging of SWP strainers when SWP Tempering with CWS is being performed simultaneously. The potential clogging is applicable to components taking suction from the fore bay, circulating water system, condenser, condensate, and the service water system when circulating water is used for tempering. Ensuring **no** surveillances occurring at this time and SWP Tempering with CWS is secured eliminates the potential for drawing wood flour into intrument lines. The wood flour will flow rapidly through the circulating water system. After the wood flour injection is complete and flour has settled out, there is **no** impact on equipment.

D.29. Clogging of hose lines may occur if the soak time of wood flour is **not** minimized. Avoid soaking wood flour for excessive periods of time.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.30. NYS DEC Water Treatment Chemical (WTC) Requirements are as follows:

- The average daily limit for wood flour injected to CWS is 400 pounds per day.
- The maximum total daily limit of wood flour injected to CWS is limited to 1200 pounds per day.
- Wood flour injection limited to 360 minutes per day.

D.31. CNM and FWS System minimum flow lines to the Main Condenser return to the following locations:

- Condensate Pumps Min flow line to the 2CWS-P1A Waterbox
- Booster Pump Min flow lines:
  - 2CNM-P2A to the 2CWS-P1B Waterbox
  - 2CNM-P2B to the 2CWS-P1C & D Waterbox
  - 2CNM-P2C to the 2CWS-P1F Waterbox
- Low energy clean-up returns below the CWS tubes.
- Feedwater Pump Min flow lines:
  - 2FWS-P1A to the 2CWS-P1A Waterbox
  - 2FWS–P1B to the 2CWS-P1C Waterbox
  - 2FWS-P1C to the 2CWS-P1F Waterbox

F.2. **Cooling Tower Operations**

NOTE

- Attachment 4, Cooling Tower Operation Curve, is used in determining the correct mode of operation for existing air temperature and heat load (Reactor power).
- As the air temperature raises or lowers, a mode change may be delayed if temperature is **not** expected to stay in a mode's temperature band. This prevents unnecessary cycling of cooling tower valves. For example, MODE II should **not** be entered if overnight temperature is expected to lower to the MODE II temperature band and return to the MODE I temperature band during the day. Mode change is required before reaching the bottom of the mode temperature band (ie: MODE II is required before reaching the bottom of the MODE II temperature band).
- A mode change is also immediately required upon ice accumulation that impacts thermal performance or when excessive ice accumulation on the cooling tower is observed.
- Actions in this Subsection are performed at 2CEC*PNL851 unless otherwise specified.
- When the ambient air temperature is less than 35 °F, the Cooling Tower basin water temperature should be monitored.

CAUTION

Basin water temperature must be maintained above 45 °F.

F.2.1. Place Cooling Tower in Mode I (Normal Operation) as follows:

NOTE

Mode I is the principle mode of operation when ambient air temperatures are likely to remain above the Mode II curve shown on Attachment 4.

CAUTION

Basin level will lower significantly when the bypass gates are closed.

a. **CLOSE** the following valves:

- 2CWS-MOG52A, COOLING TOWER BYPASS GATE VLV ... ☑
- 2CWS-MOG52B, COOLING TOWER BYPASS GATE VLV ... ☑
- 2CWS-MOG52C, COOLING TOWER BYPASS GATE VLV ... ☑

OPE

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.2. Transfer Cooling Tower from Mode I to Mode II

NOTE

Transferring from Mode I to Mode II, Ice Prevention Operation, should be performed when:

- Within the operation band of Mode II, or
- Significant ice accumulation, or
- Maintain Hotwell temperature greater than 100 °F (see Step D.22)
- Support online Cooling Tower maintenance per section H.10

a. **VERIFY** Cooling Tower is in Mode I, Normal Operation. ..... ☐

CAUTION

The de-icing valves listed in Step F.2.2.b are fully open to prevent freezing the ice prevention header.

b. **OPEN** the following valves:

- 2CWS-MOV53A, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53B, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53C, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53D, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53E, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53F, COOLING TOWER DE-ICING VLV..... ☐
- 2CWS-MOV53G, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53H, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53J, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53K, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53L, COOLING TOWER DE-ICING VLV..... ☐
- 2CWS-MOV53M, COOLING TOWER DE-ICING VLV..... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.3. Transfer Cooling Tower from Mode II to Mode III as follows:

NOTE

Transferring from Mode II to Mode III, Zoned Ice Prevention Operation, should be performed when:

- Within the operation band of Mode III, or
- Significant ice accumulation, or
- Maintain Hotwell temperature greater than 100 °F (see Step D.22)

CAUTION

2CWS-MOG54E must be left full open to prevent flooding at 2CWS-MOG54E Gate House.

a. **VERIFY** Cooling Tower is in Mode II, Ice Prevention Operation. ..... ☐

b. **POSITION** the following gates to Mid-Position (Blue light lit) by momentarily placing their respective control switches to CLOSE:

- 2CWS-MOG54B, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐
- 2CWS-MOG54C, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐
- 2CWS-MOG54D, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐

c. **VERIFY** the blue Mid-Position indicating lights are illuminated for their respective gates. ..... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.4. Transfer Cooling Tower from Mode III to Mode IV

NOTE

When placing the Cooling Tower in Mode IV there is a potential of losing approximately 20 MW electric.

a. **WHEN** Basin Water temperature **cannot** be maintained above 45 °F **OR** ice accumulation impacts Cooling Tower thermal performance, **THEN ENTER** Mode IV, Bypass Operation. ..... ☐

b. **VERIFY** Cooling Tower is in Mode III, Zoned Ice Prevention Operation. ..... ☐

CAUTION

Basin level will raise significantly when the bypass gates are open. Overflow of the basin could occur if initial basin level is high.

c. **FULLY OPEN** one **OR** more of the following valves to maintain basin temperature greater than 45 °F:

- 2CWS-MOG52A, COOLING TOWER BYPASS GATE VLV. .. ☐
- 2CWS-MOG52B, COOLING TOWER BYPASS GATE VLV. .. ☐
- 2CWS-MOG52C, COOLING TOWER BYPASS GATE VLV. .. ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.5. Transfer Cooling Tower from Mode IV to Mode III

NOTE

Mode III, Zoned Ice Prevention Operation, is to be used when:

- Ambient air temperature falls within the Mode III band, or
- Mode IV operation is **no** longer required to reduce ice accumulation on the Cooling Tower which affected the Cooling Tower thermal performance.

a. **VERIFY** Cooling Tower is in Mode IV, Bypass Operation..... ☐

CAUTION

Basin level will lower significantly when the bypass gates are closed

b. **CLOSE** the following valves:

- 2CWS-MOG52A, COOLING TOWER BYPASS GATE VLV ... ☐
- 2CWS-MOG52B, COOLING TOWER BYPASS GATE VLV ... ☐
- 2CWS-MOG52C, COOLING TOWER BYPASS GATE VLV ... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.6. Transfer Cooling Tower from Mode III to Mode II

NOTE

Mode II, Ice Prevention Operation, should be entered when:

- Ambient air temperature falls within the Mode II band, or
- Mode III operation is **no** longer required to maintain hotwell temperature greater than 100 °F (See Step D.22).

a. **VERIFY** Cooling Tower is in Mode III, Zoned Ice Prevention Operation. ..... ☐

b. **FULLY OPEN** the following gates by momentarily placing their respective control switches to OPEN:

- 2CWS-MOG54B, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐
- 2CWS-MOG54C, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐
- 2CWS-MOG54D, COOLING TOWER INLET DISTRIBUTION GATE VLV ..... ☐ _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.7. Transfer Cooling Tower from Mode II to Mode I

NOTE

Mode I, Normal Operation, should be entered when:

- Ambient air temperature falls within the Mode I band, or
- Mode II operation is **no** longer required to maintain hotwell temperature greater than 100 °F. (See Step D.22).

a. **WHEN** ambient air temperature falls within Mode I band
OR Mode II operation is **no** longer required to maintain hotwell temperature greater than 100 °F,
THEN ENTER Mode I, Normal Operation (See Step D.22). ..... ☐

b. **VERIFY** Cooling Tower is in Mode II, Ice Prevention Operation. ..... ☐

c. **CLOSE** the following valves:

- 2CWS-MOV53A, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53B, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53C, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53D, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53E, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53F, COOLING TOWER DE-ICING VLV..... ☐
- 2CWS-MOV53G, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53H, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53J, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53K, COOLING TOWER DE-ICING VLV ..... ☐
- 2CWS-MOV53L, COOLING TOWER DE-ICING VLV..... ☐
- 2CWS-MOV53M, COOLING TOWER DE-ICING VLV..... ☐ _____

F.3. **Establishing Manual Blowdown for Basin Level Control**

NOTE

- Cooling Tower Basin level can be controlled either manually per this Section, or automatically per Section F.5.
- Actions in this Subsection are performed at 2CEC*PNL851 unless otherwise specified.

CAUTION

Manual control will require monitoring Cooling Tower Basin level closely to prevent overflow or loss of CWS Pump suction. Changes in Service Water System, Turbine load or atmospheric conditions will affect level.

F.3.1. **VERIFY** the following:

- 2CWS-LIK140, CLG TOWER BASIN WTR LEVEL CONTROL, is in MANUAL..... ☐
- 2CWS-LV140, COOLING TOWER BASIN LEVEL CONTROL VLV, is closed..... ☐ _____

F.3.2. **NOTIFY** Chemistry that a CWS system blowdown will be commencing **AND VERIFY** the following:

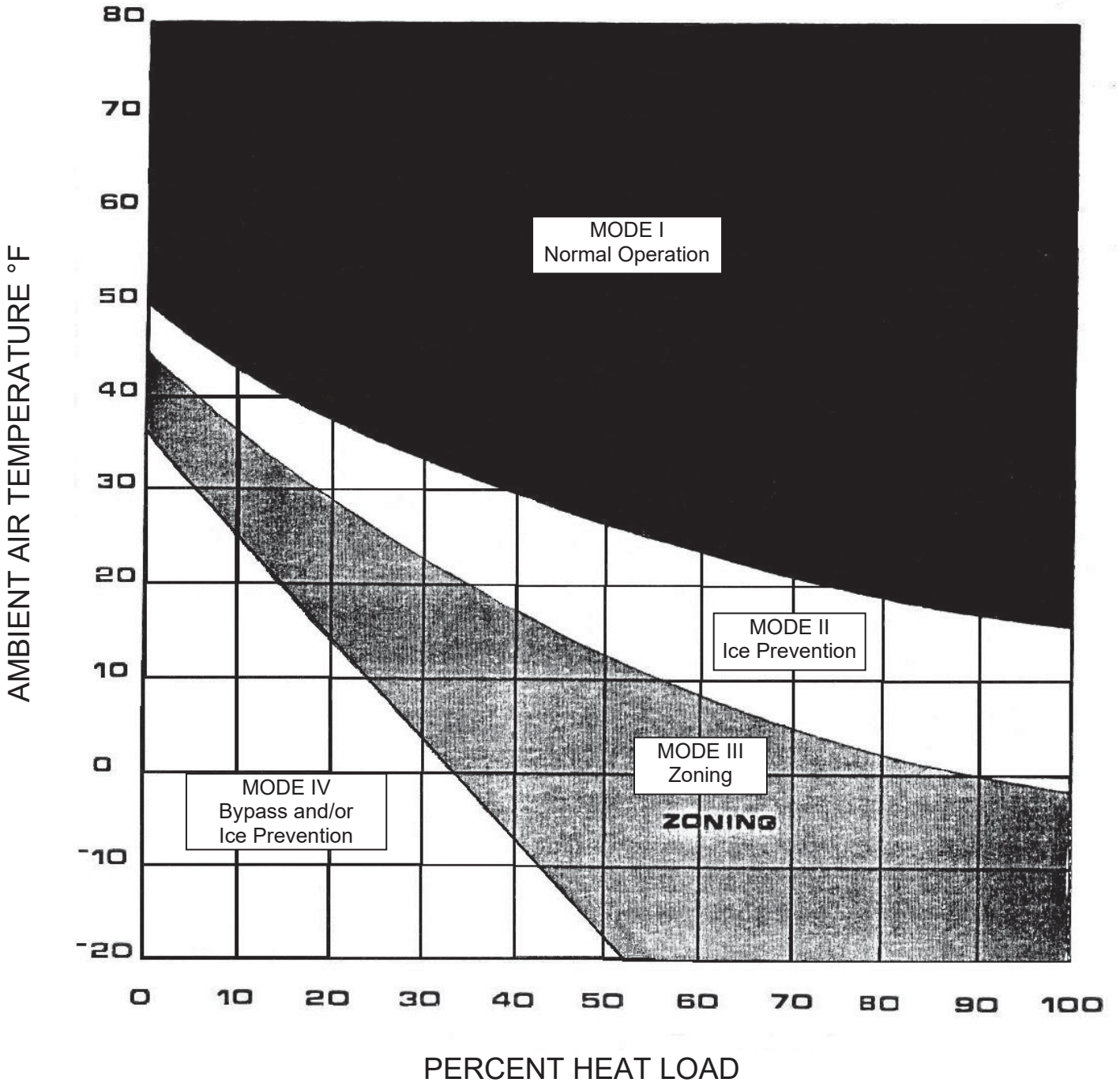
- pH between 6.0 and 9.0..... ☐
- Sampling requirements have been met **AND** are within specification. ..... ☐ _____

F.3.3. **NOTIFY** Radiation Protection to start radiation monitor 2CWS-RE157 **AND VERIFY** the monitor is online **AND** working properly. _____

F.3.4. **CONTACT** Environmental Protection for allowable blowdown flow rate. _____

F.3.5. **VERIFY** Service Water makeup to CWS System is in operation per N2-OP-11, Section F.3.0. _____

Attachment 4, Cooling Tower Operation Curve



Training ID: **Nov 2023 NRC JPM I** Revision: **0.0**

Title: **Diesel Fire Pump Local Start (Alternate Path)**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A	/	Mike Carno	7/14/23
Facility Reviewer:	Signature on File	/	John Toothaker	10/27/23

Approximate Duration: 15 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-43, Fire Protection Water
2. NUREG 1123 K/A 286000 A4.06 (3.4)

Instructor Information

A. JPM Information

1. Description

   a. This JPM evaluates the ability of the Operator to start the Diesel Fire Pump.

   b. This JPM is considered alternate path because once the Operator has started the diesel fire pump, a fuel leak will occur with sufficient enough fuel spray to render the diesel fire pump room inaccessible. The Operator will be forced to perform an emergency shutdown of the diesel fire pump using N2-OP-43 section H.5.0.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-286000-04003, Start 2FPW-P1, Diesel Fire Pump Locally

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☐ Perform ☒ Simulate			
Evaluation Location:	☒ Plant ☐ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☒ Yes ☐ No			
Safety Function:	8		Plant Service Systems	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	286000 Fire Protection System A4.06 – Diesel-driven fire pump start			
K/A Importance Rating:	RO	3.4	SRO	-

4. Recommended Start Location

   a. Unit #2 R.P. Access Building

5. Simulator Setup

   a. IC Number

      a. N/A

b. Presets / With Triggers

a. Malfunctions

a) None

b. Remotes

a) None

c. Overrides

a) None

d. Annunciators

a) None

e. Event Triggers

Event #	Event Action	Command
None	N/A	N/A

f. Equipment Out of Service

a) None

g. Support Documentation

a) Prepare a copy of N2-OP-43 section E.3.0 with no steps placekept as completed.

b) Also have N2-OP-43 section H.5.0 available to give to the Operator during performance of the alternate path.

h. Miscellaneous

a) None

6. Strategy Code

a. None

7. Tools and Equipment

    a. None

8. Commitments

    a. None

9. Prerequisites

    a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For Plant JPM's:

    a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary.

    b. With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

2. For Simulator JPM's:

    a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will provide cues as necessary.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • Preventive maintenance has just been completed on the Diesel Fire Pump • Post maintenance Testing requires the Diesel Fire Pump to be run for 30 minutes to verify proper operation **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *perform a local start of 2FPW-P1, Diesel Fire Pump in accordance with N2-OP-43 section E.3.0.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provide repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary.*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	• 2FPW-P1 can be started either locally at 2FPW-PNL234 or remotely from 2CEC-PNL849. • The preferred method is local starting of 2FPW-P1. • Annunciator 849205, DIESEL FIRE PUMP RUNNING, will alarm when the pump is started.		
Procedure Caution:	• When 2FPW-P1 is operated in Auto, the only protective trip in effect is the engine overspeed trip. • When 2FPW-P1 is operated in MAN A or MAN B, there is no protective trip in effect. Frequent checks of the engine are required while it is running to prevent unnecessary damage. • 2FPW-P1 should be run at least 20 minutes before being secured to reach stable operating engine temperatures as recommended by the manufacturer.		
2.	BEFORE starting 2FPW-P1, notifies the Fire Brigade Leader ***Cue:*** *As the fire brigade leader, acknowledge notification.*	S (E.3.1)	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	The following step is performed inside 2FPW-PNL234 in the Diesel Fire Pump Room.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
3.	Locally starts 2FPW-P1, Diesel Engine Driven Fire Pump, as follows:	S (E.3.2)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step.*
Procedure Note:	Annunciator 849221, DIESEL FIRE PUMP NOT IN AUTO START, will alarm in the next step.		
4.	Places SELECTOR switch in MAN A OR MAN B **Cue:** *The component you have identified is in the position you described.*	S (E.3.2.a)	***PASS / FAIL** ***STD:*** *At 2FPW-PNL234 rotates SELECTOR switch clockwise/counter clockwise to the MAN A/MAN B position.* ***Failure =*** *SELECTOR Switch in MAN A or MAN B not achieved as determined by Operator verbalization.*
5.	Starts 2FPW-P1 by depressing START pushbutton. **Cue:** *The component you have identified is in the position you described.* **Cue:** *The Diesel Fire pump starts and comes up to rated speed.* **Cue:** *If the Operator contacts the control room to verify receipt of annunciator 849205, DIESEL FIRE PUMP RUNNING, inform the Operator that the annunciator is in alarm.*	S (E.3.2.b)	***PASS / FAIL** ***STD:*** *At 2FPW-PNL234 depresses START pushbutton.* ***Failure =*** *START pushbutton depression not achieved as determined by Operator verbalization.*
6.	IF 2FPW-P1 fails to crank, THEN performs the following: • Briskly turns SELECTOR switch to opposite Manual position • Depresses 2FPW-P1 START pushbutton	S (E.3.2.c.1 & 2)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step as N/A.*
Procedure Note:	The following step is performed at 2CEC-PNL849 in the Control Room.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
7.	**IF** remotely starting Diesel Engine-Driven Fire Pump 2FPW-P1 from the U2 Main Control Room, **THEN** at 2CEC-PNL849, **DEPRESS** DIESEL ENG DRIVEN FIRE PUMP START pushbutton, **OTHERWISE MARK** this step N/A.	S (E.3.3)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step as N/A.*
Procedure Note:	The following parameters should be monitored at 5 minute intervals during the first 15 minutes of pump operation and then once per hour.		
8.	Confirm proper pump operation as indicated by the following: • NO unusual noise OR vibrations **Cue:** *If asked, report that there are no unusual noises or vibrations.*	S (E.3.4 first bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Proper communications used.*
9.	• NO red lights lit on 2FPW-PNL234 **Cue:** *If asked, report that there are no red lights lit on 2FPW-PNL234.*	S (E.3.4 second bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates red light indications on 2FPW-PNL234. Proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
10.	System pressure is approximately 150 psig as indicated on one of the following: • 2FPW-PI108, 2FPW-P1 discharge pressure gauge, located at the pump • 2FPW-PI214, FPW System pressure gauge, located on the wall by 2FPW-PNL234 • 2FPW-PI215, FIRE HEADER WTR PRESS meter, located on 2CEC-PNL849 **Cue:** *If asked, report that all pressure gauges read approximately 150 psig.* **Cue:** *If the Operator contacts the control room to verify 2FPW-PI215, FIRE HEADER WTR PRESS meter on 2CEC*PNL849, inform the Operator that the pressure meter reads 150 psig.*	S (E.3.4 third bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates/evaluates 2FPW-PI108 (at pump) and 2FPW-PI214 (2FPW-PNL234) pressure indications. Proper communications used.*
11.	Engine speed is approximately 1950 RPM as indicated on the diesel **Cue:** *If asked, Engine speed indicates approximately 1950 rpm.* **Note:** The tachometer for DG fire pump is currently broken, but it is desired to still give this reading to the Operator.	S (E.3.4 fourth bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates/evaluates diesel rpm indications. Proper communications used.*
12.	Lube oil pressure is 59 to 80 psi as indicated on the diesel **Cue:** *If asked, engine lube oil pressure indicates approximately 70 psig.*	S (E.3.4 fifth bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates/evaluates diesel lube oil pressure indications. Proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
13.	Jacket water temperature is 165 to 195°F as indicated on 2FPW-TI1012, DSL ENG COOLANT TEMP gauge, on the diesel **Cue:** *If asked, jacket water temperature is 175°F as indicated on 2FPW-TI1012, DSL ENG COOLANT TEMP gauge, on the diesel.*	S (E.3.4 sixth bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates/evaluates 2FPW-TI1012, DSL ENG COOLANT TEMP gauge. Proper communications used.*
Alternate Path:	During performance of the next step, when the Operator checks the discharge strainer D/P, a fuel leak will develop at the fuel filter causing diesel fuel to start spraying into the area. This will require an emergency shutdown of the diesel fire pump. The instructor should provide the associated cue and inform the Operator that the area around the diesel fire pump engine is now inaccessible. If asked for further guidance, the Operator should be given the cue to make a recommendation and carry out the recommended action. Ensure that the cue given does not infer that the entire room is inaccessible. The intent is that only the area around the "NOD STOPP" handle and the area immediately near the diesel fire pump become inaccessible. The evaluator can provide leak cues as necessary, if questioned by the Operator to ensure the JPM objective can be achieved with regards to room accessibility.		
14.	Observes 2FPW-PDI211, DIESEL FIRE PUMP DISCHARGE STRAINER DP. **Cue:** *If asked, 2FPW-PDI211 indicates 4 psid.* **Cue:** *When the Operator is finished checking operating parameters, inform the Operator that a severe diesel fuel leak has started and that diesel fuel is spraying out the side of the engine near the fuel filter making the engine area inaccessible.*	S (E.3.5)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Locates/evaluates 2FPW-PDI211, DIESEL FIRE PUMP DISCHARGE STRAINER DP gauge. Proper communications used.*
15.	Reports to the control room that the diesel fire pump has developed a fuel leak **Cue:** *As the control room, acknowledge the report. If additional direction is requested, inform the Operator to make a recommendation and carry out the recommended action.*	S	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step. Proper communications used.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
16.	Operator references and performs N2-OP-43, section H.5.0, Emergency Shutdown of Diesel Driven Fire Pump 2FPW-P1	S (H.5.0)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step.*
Procedure Note:	The "NOD STOPP" handle is located behind the lube oil dipstick on the south side of the Diesel Driven Fire Pump engine.		
17.	Stops fuel oil flow to Diesel Driven Fire Pump engine by performing one OR both of the following:	S (H.5.1)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step.*
18.	**SHUTS OFF** fuel oil by **PULLING DOWN AND HOLDING** "NOD STOPP" handle **UNTIL** engine stops	S (H.5.1 first bullet)	SAT / UNSAT ***STD:*** *Reads/reviews and placekeeps step as N/A.*
19.	**SHUTS OFF** fuel oil by **UNLOCKING AND CLOSING** 2FOF-V10, FUEL OIL TANK OUTLET LINE ISOL **Cue:** *The component you have identified is in the position you described. The diesel has stopped running.*	S (H.5.1 second bullet)	***PASS / FAIL** ***STD:*** *Locates 2FOF-V10, FUEL OIL TANK OUTLET LINE ISOL, unlocks valve handwheel and rotates the valve handwheel clockwise to the full closed position. Verifies valve stem lowering.* ***Failure =*** *Closure of 2FOF-V10, FUEL OIL TANK OUTLET LINE ISOL not achieved as determined by Operator verbalization.*
20.	Reports to the control room that an emergency stop of the diesel fire pump has been performed and that the diesel fire pump has been successfully shutdown.	S	SAT / UNSAT ***STD:*** *Proper communications used.*
Instructor Note:	After the Operator has closed 2FOF-V10, FUEL OIL TANK OUTLET LINE ISOL and informed the control room of actions taken, provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*Diesel Fire pump secured using the emergency shutdown section (H.5.0) of N2-OP-43.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • Preventive maintenance has just been completed on the Diesel Fire Pump • Post maintenance Testing requires the Diesel Fire Pump to be run for 30 minutes to verify proper operation

INITIATING CUE	**(Operator Name)**, *perform a local start of 2FPW-P1, Diesel Fire Pump in accordance with N2-OP-43 section E.3.0.*

Plant JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • Preventive maintenance has just been completed on the Diesel Fire Pump • Post maintenance Testing requires the Diesel Fire Pump to be run for 30 minutes to verify proper operation **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *perform a local start of 2FPW-P1, Diesel Fire Pump in accordance with N2-OP-43 section E.3.0.*

Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-43

FIRE PROTECTION WATER

Revision 02300

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-21-00421 [Permanent Temp Change]:

Description: NOTE changed to ensure correct solution is used.
Reason: Changes made in support of IR 04373608-06.

- Step A.4 added CY-NM-210-618 as procedure reference.
- Step F.4.3 NOTE updated to ensure that coolant solution is verified within the required band.

PCR-21-00800:

Description: Change steps H.5.1.a and H.5.1.b to bullets. Added another note prior to step H.5.1.
Reason: Steps are ambiguous when performing both steps. Impossible to perform both steps if Diesel Driven Fire Pump Engine is accessible.

- Step H.5.1 updated to align with approved PCR content.

PCR-21-01437:

Description: Delete N2-OP-43 Attachment 10.
Reason: Attachment 10 was added in 2002 under NMP2-PCE 62933 to allow installation of temporary change packages when 2FPW-TK1 requires maintenance. This attachment does not meet current procedural guidance for PCTCC requirements of CC-AA-112.

- Attachment 10 was deleted.

Procedure Writer Changes:

- Step A.4 updated procedure number for Fire Fighting.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 5
B. SYSTEM DESCRIPTION 7
C. OPERATING REQUIREMENTS 9
D. PRECAUTIONS AND LIMITATIONS 10
E. STARTUP 13
E.1. System Fill and Vent 13
E.2. Starting the Electric Fire Pump [SOP] 18
E.3. Starting the Diesel Fire Pump [SOP] 20
F. NORMAL OPERATIONS 22
F.1. Standby Condition Status Checks 22
F.2. Shifting Press Maint Pump Lead-Lag Operation 24
F.3. Fuel Oil Addition to Fuel Oil Storage Tank 25
F.4. Adding Coolant to The Diesel Fire Pump Radiator 30
F.5. Cross-connecting Unit 1 And Unit 2 Fire Systems 33
F.6. Restoring Fire Protection Water Systems 34
F.7. Diesel Fuel Addition Direct from Vendor Truck 47
G. SHUTDOWN 49
G.1. System Shutdown 49
G.2. Securing the Electric Fire Pump 50
G.3. Securing the Diesel Fire Pump 51
H. OFF-NORMAL OPERATIONS 52
H.1. Supplying Water for Condenser Tube Cleaning 52
H.2. Filling and Venting the Drywell Hose Reel Stations 54
H.3. Isolating and Draining The Drywell Hose Reel Stations 56
H.4. Swapping Fire Pump Discharge Strainers [C1] 59
H.5. Emergency Shutdown of Diesel Driven Fire Pump 2FPW-P1 62
H.6. Fire Sprinkler Head Flow Verification – Preaction Zones Fed from 2FPW-V987 63
H.7. Fire Sprinkler Head Flow Verification – Preaction Zones Fed from 2FPW-V988 76
H.8. Recovering from Ground in Diesel Fire Pump Remote Start Circuit 88
H.9. Adjustment and Water Priming of Pre-action System Air Pressure 90
H.10. Loss of Automatic Makeup to 2FPW-TK2 Pressure Maintenance Pump Supply Tank 96
Attachment 1, System Fuse Lineup 98
Attachment 2, Automatic Wet Pipe Sprinkler Systems 102

TABLE OF CONTENTS (Continued)

SECTION **PAGE**
Attachment 3, Automatic Dry Pipe Sprinkler Systems 104
Attachment 4, Automatic Water Spray and Deluge Sprinkler Systems 105
Attachment 5, Remote-Manual Water Spray Systems..... 107
Attachment 6, Preaction Sprinkler Systems 108
Attachment 7, 2FOF-TK1, Diesel Fire Pump Fuel Oil Storage Tank, Curve 112
Attachment 8, Fire Hydrants and Locations..... 114
Attachment 9, Fire Hose Reel and Hose Rack Locations 116
Attachment 10, Fire Pump Discharge Strainer 127
Attachment 11, NMP2 Pre-Action System Generic Drawing..... 128

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

None

A.2. Licensee Documentation

Updated Safety Analysis Report (USAR)

- Volume 21, Section 9.5.1, Fire Protection System
- Volume 23, Appendix 9A, Section 9A.3.6.2, Fire Protection Water Supply System
- Volume 23, Appendix 9A, Section 9A.3.6.3, Water Sprinkler and Hose Standpipe Systems

A.3. Standards, Regulations and Codes

- 10CFR50, Appendix A, Criterion 3, Fire Protection
- BTP APCSB 9.5-1, Appendix A, Guidelines for Nuclear Power Plants Docketed Prior to July 1, 1976, dated August 23, 1976
- NRC Guideline, Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance, dated June 14, 1977
- NRC Generic Letter 86-10, Implementation of Fire Protection Requirements, dated April 24, 1986
- Nuclear Regulatory Guide 1.33, Rev. 2
- 40 CFR 112, Oil Pollution Prevention

A.4. Policies, Programs, and Procedures

- OP-NM-201-005, Fire Fighting
- CC-AA-209, Fire Protection Program Configuration Change Review
- CC-AA-209-1001, Guidelines for Performing Fire Protection Program Configuration Change Review
- CC-AA-211, Fire Protection Program
- CC-AA-211-1001, Fire Protection Engineering Evaluations
- CY-NM-210-618, Glycol Chemical Surveillance
- N2-OP-11, Service Water System
- N2-OP-19, Instrument and Service Air System
- N2-OP-47, Fire Detection
- N2-OP-50, Domestic Water
- N2-OP-71B, 4.16KV A.C. Power Distribution
- N2-OP-71C, 600V A.C. Power Distribution
- N2-MPM-FPW-A854, Diesel Driven Fire Pump Engine Inspection

A.5. **Technical Information**

A.6. **Flow Diagrams**

P&IDs 43A - 43H, Fire Protection - Water

A.7. **Electrical Diagrams**

- ESK-5FPW01 - 02, Motor Driven Fire Pump 2FPW-P2
- ESK-6FPW01 (02), Motor Driven Fire Maint Pump P3A(B)
- ESK-7FPW01, Motor Driven Fire Maintenance Pump Auxiliary Circuit
- ESK-7FPW02, Diesel Engine Driven Fire Pump 2FPW-P1
- ESK-7FPW03, Fire Protection - Water
- ESKs-7FPW08 - 38, Water Spray - Zones

A.8. **Logic Diagrams**

- LSK-15-1.1A, Motor Driven Fire Pumps
- LSK-15-1.1B, Motor Driven Fire Press Maint Pumps 3A/B
- LSK-15-1.1C, Motor Driven Fire Maintenance Pumps
- LSK-15-1.2A - 2D, Engine Driven Fire Pump
- LSK-15-1.3, Fire Protection - Water Sprinklers
- LSK-15-1.4A - 4H, Fire Protection - Water Spray

A.9. **Vendor Manuals**

- File Number 20799, Vertical Turbine Fire Pump with Right Angle Gear Drive Diesel Engine and Controller
- OE-2009-003462, Potential Failure of Fire Water Supply Pumps to Automatically Start Due to a Fire

A.10. **Supplemental References**

- CR-2009-004036, Gaps exist in procedure scope for Unit 2 Diesel Fire Pump

A.11. **Commitments**

Sequence Number	Commitment Number	Description
1	NM 2002-4131 Passport IR 02105564	2FPW-STR2 Reliability Degraded Prior to Preventive Maintenance

B. SYSTEM DESCRIPTION

The function of the Fire Protection Water (FPW) System is to provide a reliable, readily available source of water for controlling and extinguishing fires. The FPW System is composed of hose stations, hydrants, deluge and water spray systems, fire pumps, sprinkler systems and pressure maintenance pumps. The FPW System utilizes the loop header concept to ensure a highly reliable fire fighting system is available for use when required.

The primary source of water for the main FPW loop is from the Intake Bay. The water is supplied by two parallel fire pumps (2FPW-P1 and P2) located in the Fire Pump Room in the Screenwell Building. 2FPW-P1, Diesel Engine Driven Fire Pump, is a diesel powered vertical turbine type pump. 2FPW-P2, Motor Driven Fire Pump, is an electric motor driven vertical turbine type pump. Each pump is rated for 2500 gpm at 125 psig. An alternate supply of water for the main FPW loop is from the Unit 1 main FPW loop by cross-connecting through manual valve 2FPW-V149 or motor operated valve 100-54 which is controlled from the Unit 1 Control Room.

When the fire pumps are not running, pressure in the FPW System is maintained by the pressure maintenance system. This system consists of a Hydro Pneumatic Tank (2FPW-TK1), a Pressure Maintenance Pump Water Supply Tank (2FPW-TK2), and two Motor Driven Pressure Maintenance Pumps (2FPW-P3A and P3B). 2FPW-TK1 has a 150 gallon capacity and is used to absorb system pressure surges. 2FPW-TK2 maintains NPSH for the pumps and is automatically filled from the Domestic Water (DWS) System or the Service Water (SWP) System. On low tank level (16.5" above tank bottom), 2FPW-AOV132 will auto open to fill the tank with DWS water. On low/low tank level (12.25" above tank bottom), 2FPW-AOV222 will auto open to fill the tank with SWP water. Both valves will auto close when level is 16" below the top of the tank.

2FPW-P3A and P3B are rated for 30 gpm each and maintain pressure at 120 - 135 psig. The pumps take suction from 2FPW-TK2 to supply makeup water to the FPW loop supply lines. A selector switch is provided at 2FPW-PNL233 to select one pressure maintenance pump as the lead pump and one as the lag pump. The lead pump will auto start on a low fire header pressure of 120 psig and the lag pump auto starts on a low/low pressure of 110 psig. The pumps will auto stop when pressure is 135 psig. The pumps will trip on motor overload.

If the FPW header pressure drops below 90 psig, the Motor Driven Fire Pump (2FPW-P2) will auto start to provide system requirements. If pressure drops to 85 psig, the Diesel Engine Driven Fire Pump (2FPW-P1) will auto start. The pumps will continue to operate until they are manually secured. 2FPW-P2 will auto trip on motor electrical fault. 2FPW-P1 will auto trip on Engine Overspeed (greater than 120%), low engine oil pressure (less than 8 psig), or high engine cooling water temperature (greater than 205 °F). When 2FPW-P1 is running in the AUTO or MANUAL modes, the low oil pressure and high cooling water temperature trips are bypassed.

The FPW System piping consists mainly of two closed loops. The outer loop connects various points within the plant and the inner loop supplies FPW to all required locations in the Turbine Building through various branch lines. The branch lines and associated post indicator valves are arranged to enable double valve isolation of any branch line.

B. SYSTEM DESCRIPTION (Continued)

Diesel Fire Pump Protective Features

When the Diesel Fire Pump Control switch at 2FPW-PNL234 is in the TEST position, protective trips for Oil Pressure, Engine Temperature and Overspeed are in effect. Local audible and indicator lights at 2FPW-PNL234 (and Control Room annunciator 849213) will also activate if any of the setpoints are reached.

When the Diesel Fire Pump Control switch at 2FPW-PNL234 is in the MAN A or B position, none of the Protective Trips for Oil Pressure, Engine Temperature and Overspeed are in effect. However, local audible and indicator lights at 2FPW-PNL234 (and Control Room annunciator 849213) will activate if any of the setpoints are reached.

When the Diesel Fire Pump Control switch at 2FPW-PNL234 is in the AUTO position, only the protective trip for overspeed is in effect. Local audible and indicator lights at 2FPW-PNL234 (and Control Room annunciator 849213) will activate if the Overspeed, Low Oil Pressure or High Engine Temperature setpoints are reached.

The Diesel Fire Pump engine will attempt six sequential 15 second duration starting cranks if required to start the engine. If the engine does not start after the six crank attempts, an audible and indicator light will alarm at 2FPW-PNL234 after a 165 second delay indicating a failure to start.

A ground in the remote start circuit between 2FPW-PNL234 and 2CEC-PNL849, caused by a fire or other event, could disable the 24VDC control circuit by opening the DC breaker for the battery bank which is aligned. Refer to Subsection H.8, Recovering from Ground in Diesel Fire Pump Remote Start Circuit, for information regarding recovery from this event.

C. OPERATING REQUIREMENTS

C.1. The following systems must be in operation to support the FPW System:

- N2-OP-11, Service Water System
- N2-OP-19, Instrument and Service Air System
- N2-OP-47, Fire Detection
- N2-OP-50, Domestic Water
- N2-OP-71B, 4.16KV A.C. Power Distribution
- N2-OP-71C, 600V A.C. Power Distribution

C.2. The level in 2FOF-TK1, Diesel Fire Pump Fuel Oil Storage Tank, must be greater than 60% (350 gal).

C.3. 2FPW-TK2, Pressure Maintenance Pump Supply Tank, must be filled.

D. PRECAUTIONS AND LIMITATIONS

D.1. The FPW System should never be completely shutdown. A section of the system can be isolated for maintenance while the remainder of the system remains in operation.

D.2. Caution shall be used when isolating a section of the FPW header as sections can be fed from two directions.

D.3. Prior to plant startup, the Drywell hose stations shall be isolated and drained per Section H.3, Isolating and Draining The Drywell Hose Reel Stations.

D.4. Prior to initiating the FPW System to suppress an electrical fire or to suppress a Charcoal Filter Train fire, an actual hazard shall be verified to exist. Inadvertent or mistaken application of water can cause extensive damage to electrical and filter components.

D.5. Always drain FPW hoses prior to restoring them to the hose station.

D.6. Prior to changing any FPW valve or switch position, Operations shall be notified.

D.7. Maximum cooling water shell side temperature rise in the diesel engine cooler is 22 °F.

D.8. Maximum diesel engine coolant temperature should **not** exceed 200 °F.

D.9. It may be necessary to vent and prime 2FPW-P1 fuel line and filters following maintenance to prevent engine stalls or long starts.

D.10. If the Diesel Engine Driven Fire Pump exhaust is smokey, corrective action should be taken for the following indications:

- Black smoke indicates improper combustion due to restricted air cleaner, over-loading, over fueling, etc.
- Blue smoke suggests lube oil in the combustion chamber.

D.11. When the Diesel Engine Driven Fire Pump is operated in Auto, the only protective trip in effect is the engine overspeed trip. When the Diesel Engine Driven Fire Pump is operated in MAN A or MAN B, there are **no** protective trips in effect. Frequent checks of the engine are required while it is running to prevent unnecessary damage. The Diesel Engine Driven Fire Pump should be run at least 20 minutes before being secured to reach stable operating engine temperatures as recommended by the manufacturer.

D.12. The following motor start limitations shall be observed for 2FPW-P2, Motor Driven Fire Pump:

- Two starts in succession from ambient temperature after which a 60 minute wait is required prior to subsequent start attempts.
- One start from rated temperature (established after 30 minutes run time), after which a 60 minute wait is required prior to subsequent start attempts.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.13. If either the Motor or Diesel Engine Driven Fire Pump will be out of service for GREATER THAN 7 Days, an alternate source of fire water must be provided, as required by USAR Section 9A.3.6.2.6.

D.14. FPW System requirements and compensatory actions are listed in the USAR Volume 23, Appendix 9A, Sections 3.6.2 and 3.6.3.

D.15. To completely disable the diesel fire pump from starting, the battery leads need to be lifted.

D.16. As an operator good practice, the Motor driven Electric Fire pump supply breaker should be checked properly recharged following normal opening operation.
Refer to N2-ELU-01 Subsection 3.4.

D.17. Fire Water Pump Discharge Duplex Strainer, 2FPW-STR1(2) should be swapped per Section H.4, if differential pressure exceeds 10 psid. Differential pressure should **not** exceed 20 psid at any time. **[C1]**

D.18. If AC power is lost to any of the following buildings, the associated dry pipe fire suppression system(s) may discharge due to loss of power to the respective air compressor(s) and the resultant loss of air header pressure.

- West Annex
- Cold Storage
- Bottled Gas Storage

D.19. When a fire pump must be started to maintain fire header pressure in support of system flushing or long term maintenance activities, the Unit 2 Construction Fire Pump is preferred over the Motor Driven Fire Pump or Diesel Driven Fire Pump. Use of the Motor Driven Fire Pump or Diesel Driven Fire Pump under these circumstances unnecessarily challenges the pumps, drivers, discharge check valves, and discharge PSVs.

D.20. Pre-action and dry pipe systems must be drained utilizing additional installed drains plugs and caps at the end of branch lines when those branches have been filled by initiation for testing or maintenance.

D.21. 2FPW-V1010, RX BLDG SUPPLY HDR DRAIN, is identified as a normally closed manual Secondary Containment boundary component per TRM Table T3.6.4.2-1. If 2FPW-V1010 is opened, a direct breach of Secondary Containment will occur through the Reactor Building Floor Drain system. Refer to TS 3.6.4.2 for required actions when 2FPW-V1010 is open.

To maintain Secondary Containment operability 2DFR-V128 must be isolated or verify total Secondary Containment Leakage is **not** exceeded using N2-ODP-OPS-0113.

D.22. PMP-100-556, NMP-2 Fire Pump at Unit 1, may Auto start on a Low Pressure Condition if it is in Standby.

D.23. The Diesel Fire Pump Emergency Cooling Filter Bypass Line valve 2FPW-V4, PCV117 BYPASS is **not** to be used during normal pump operation.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.24. Diesel Engine Coolant Precautions

D.24.1. Ensure proper protective clothing including goggles, impermeable gloves and aprons are worn at all times when working with diesel fuel **OR** chemicals such as ethylene **OR** diethylene glycol.

D.24.2. Ensure spill kits are at the job site **AND** are installed around floor drains prior to draining fluids from the diesel fire pump engine. If a chemical spill should occur, notify the Control Room.

D.24.3. Ensure suitable containers are used to collect **AND** transport hazardous material fluids to proper disposal areas.

D.24.4. Ethylene **OR** diethylene glycol can cause irritation to the skin and eyes. It is absorbed through the skin. If ingested, glycol can be harmful **OR** fatal.

D.24.5. If glycol contacts the body internally **OR** externally while performing maintenance, the following emergency first aid steps are recommended:

- Notify Control Room if an emergency condition exists.
- Skin contact: flush with water 15 minutes.
- Eye contact: flush with water 15 minutes, call control room to notify 911.
- Ingestion: induce vomiting, give water, call control room to notify 911.
- Inhalation: remove to fresh air.
- If person is unconscious or having difficulty breathing: **DO <u>not</u>** induce vomiting **OR** give water, **CALL** control room to notify 911 immediately.

E. STARTUP (Continued) **Initials**

E.3. **Starting the Diesel Fire Pump [SOP]**

NOTE

- 2FPW-P1 can be started either locally at 2FPW-PNL234 or remotely from 2CEC-PNL849.
- The preferred method is local starting of 2FPW-P1.
- Annunciator 849205, DIESEL FIRE PUMP RUNNING alarms when pump is started.

CAUTION

- When 2FPW-P1 is operated in Auto, the only protective trip in effect is the engine overspeed trip.
- When 2FPW-P1 is operated in MAN A or MAN B, there is **no** protective trip in effect. Frequent checks of the engine are required while it is running to prevent unnecessary damage.
- 2FPW-P1 should be run at least 20 minutes before being secured to reach stable operating engine temperatures as recommended by the manufacturer.

E.3.1. **BEFORE** starting 2FPW-P1, **NOTIFY** the Fire Brigade Leader. _____

NOTE

- The next step is performed inside 2FPW-PNL234 in the Diesel Fire Pump Room.
- Annunciator 849221, DIESEL FIRE PUMP not IN AUTO START, will alarm in the next step.

E.3.2. **IF** Locally starting 2FPW-P1, Diesel Engine Driven Fire Pump,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **PLACE** SELECTOR switch in MAN A **OR** MAN B. ..... ☐

b. **START** 2FPW-P1 by **DEPRESSING** START pushbutton. ..... ☐

c. **IF** 2FPW-P1 fails to crank,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **BRISKLY TURN** SELECTOR switch to opposite Manual position. ..... ☐

2) **DEPRESS** 2FPW-P1 START pushbutton. ..... ☐ _____

E. STARTUP (Continued) **Initials**

NOTE

The following step is performed at 2CEC-PNL849 in the Control Room.

E.3.3. **IF** remotely starting Diesel Engine-Driven Fire Pump 2FPW-P1 from the U2 Main Control Room,
THEN at 2CEC-PNL849,
DEPRESS DIESEL ENG DRIVEN FIRE PUMP START pushbutton,
OTHERWISE MARK this step N/A. _____

NOTE

The following parameters should be monitored at 5 minute intervals during the first 15 minutes of pump operation and then once per hour.

E.3.4. **CONFIRM** proper pump operation as indicated by the following:

- **No** unusual noise **OR** vibrations..... ☐
- **No** red lights lit on 2FPW-PNL234 ..... ☐
- System pressure is approximately 150 psig as indicated on **one** of the following:
  - 2FPW-PI108, 2FPW-P1 discharge pressure gauge, at pump .... ☐
  - 2FPW-PI214, FPW System pressure gauge, on the wall by 2FPW-PNL234 ..... ☐
  - 2FPW-PI215, FIRE HEADER WTR PRESS meter, on 2CEC-PNL849..... ☐
- Engine speed is approximately 1950 RPM as indicated on the diesel... ☐
- Lube oil pressure is 59 to 80 psi as indicated on the diesel..... ☐
- Jacket water temperature is 165 to 195 °F as indicated on 2FPW-TI1012, DSL ENG COOLANT TEMP gauge, on the diesel ..... ☐ _____

E.3.5. **IF** 2FPW-PDI211, DIESEL FIRE PUMP DISCHARGE STRAINER DP reaches 10 psid,
THEN SWAP strainers per Section H.4,
OTHERWISE MARK this step N/A. **[C1]** _____

E.3.6. **IF** operating the Diesel Fire Pump in an emergency,
AND normal cooling is inadequate,
THEN THROTTLE 2FPW-V4 PCV117 BYPASS as needed
TO CONTINUOUSLY MAINTAIN pressure on 2FPW-PI111 less than or equal to 28 psig,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS

F.1. Standby Condition Status Checks

F.1.1. At 2CEC-PNL849, **CONFIRM** the following:

- 2FPW-AOV132, PRESS MAINT TK2 DOMESTIC WTR FILL, control switch is in AUTO..... ☐
- 2FPW-AOV222, PRESS MAINT TK2 SERVICE WTR FILL, control switch is in AUTO..... ☐
- 2FPW-P2 POWER AVAILABLE blue light is lit..... ☐
- FPW header pressure is greater than 125 psig as indicated on 2FPW-PI215, FIRE HEADER WTR PRESS meter..... ☐ _____

NOTE

2FPW-PNL233 is located in the Electric Fire Pump Room.

F.1.2. At 2FPW-PNL233, **CONFIRM** the following:

- 2FPW-P2 POWER AVAILABLE white light is lit. ..... ☐
- SUSTAINED BUS UNDERVOLTAGE white light is **not** lit..... ☐
- 2FPW-P2, MOTOR DRIVEN FIRE PUMP P2, control switch is in Normal-After-STOP..... ☐
- 2FPW-P3A, MOTOR DRIVEN PRESS MAINTENANCE PMP P3A, control switch is in Normal-After-STOP. ..... ☐
- 2FPW-P3B, MOTOR DRIVEN PRESS MAINTENANCE PMP P3B, control switch is in Normal-After-STOP. ..... ☐ _____

H. OFF-NORMAL OPERATIONS (Continued) **Initials**

H.5. **Emergency Shutdown of Diesel Driven Fire Pump 2FPW-P1**

NOTE

- The "NOD STOPP" handle is located behind the lube oil dipstick on the south side of the Diesel Driven Fire Pump engine.
- When the Diesel Driven Fire Pump engine is accessible, the “NOD STOPP” handle is preferred.

H.5.1. **STOP** fuel oil flow to Diesel Driven Fire Pump engine by **PERFORMING ONE OR BOTH** of the following:

- **SHUT OFF** fuel oil by **PULLING DOWN AND HOLDING** "NOD STOPP" handle **UNTIL** engine stops. ..... ☐

AND / OR

- **SHUT OFF** fuel oil by **UNLOCKING AND CLOSING** 2FOF-V10, FUEL OIL TANK OUTLET LINE ISOL..... ☐ _____

Training ID: **Nov 2023 NRC JPM J** Revision: **0.0**

Title: **Vent Control Rod Overpiston**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A		Mike Carno	7/14/23
Facility Reviewer:	Signature on File		John Toothaker	10/27/23

Approximate Duration: 20 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

_____

Evaluators Signature: _____ Date: _____

References

1. N2-EOP-6.14, Rev. 00200, Alternate Control Rod Insertions
2. NUREG 1123 K/A 295037 EA1.05 (3.9)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to manipulate plant controls associated with Control Rod Hydraulic Control Units. The operator will perform actions to insert control rod 30-43 to notch 00 by locally venting its overpiston area in accordance with N2-EOP-6.14.

   b. This JPM is not considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-201001-04023, Vent Control Rod Over-Piston Volume During Anticipated Transient Without Scram Conditions

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO	☒ RO	☐ EO	☐ N/A
Evaluation Method:	☐ Perform ☒ Simulate			
Evaluation Location:	☒ Plant ☐ Simulator ☐ Classroom			
Time Critical Task:	☐ Yes ☒ No			
Alternate Path:	☐ Yes ☒ No			
Safety Function:	1		Reactivity Control	
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3		
K/A Statement: (Add justification statement below for K/A's < 3.0)	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.05 – Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: CRD hydraulics systems			
K/A Importance Rating:	RO	3.9	SRO	-

4. Recommended Start Location

   a. Unit #2 R.P. Access Building

5. Simulator Setup

   a. IC Number

      a. N/A

   b. Presets / With Triggers

      a. Malfunctions

         a) None

b. Remotes

a) None

c. Overrides

a) None

d. Annunciators

a) None

e. Event Triggers

Event #	Event Action	Command
None	N/A	N/A

f. Equipment Out of Service

a) None

g. Support Documentation

a) Prepare a complete copy of N2-EOP-6.14 with no steps placekept as completed.

h. Miscellaneous

a) None

6. Strategy Code

a. None

7. Tools and Equipment

a. None

8. Commitments

a. None

9. Prerequisites

a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

a. None

B. Read Before Every JPM Performance

1. For Plant JPM's:

   a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary.

   b. With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

2. For Simulator JPM's:

   a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will provide cues as necessary.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • A scram has occurred. • Several rods have not fully inserted. • NO CRD pumps are available. • Communications are established with Control Room. • An OD-7, Print out of Rod Positions is **NOT** available. • You have been given a F2-57 key. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *insert control rod 30-43 to notch 00 by locally venting its overpiston area in accordance with N2-EOP-6.14, Flowchart* C

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provides repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Caution:	• This activity will result in draining of Rx coolant and poses the risk of creating a high level of radioactivity airborne as well as creating high radiation conditions and contamination in local and general areas of Rx Bldg El. 261. • ALL appropriate protective measures must be taken for this activity as deemed necessary by Radiation Protection.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Evaluator Note:	• Where possible when providing cues for component configuration the use of the words "As indicated" is desired if the actual plant configuration matches the JPM condition. In cases where the configuration of actual plant components does not match the intended JPM condition, provide the scripted JPM cue. • In JPM steps 2 through 5, the cue provides the component position because the cues are describing the initial condition of each component prior to manipulation by the Operator. • It is not desired to have the Operators actually get out hoses/adapters/tool/etc. Once the Operator indicates where they would get the item, you can cue them that they now have the item.		
2.	IF possible, obtains an OD-7 printout of rod positions from the R*Time Computer	S (1.1.1)	SAT / UNSAT ***STD:*** *Recalls from the initial conditions that an OD-7, Print out of Rod Positions is not available and placekeeps step.*
3.	Establishes communications between the Control Room AND the HCU Areas. (RB 261')	S (1.1.2)	SAT / UNSAT ***STD:*** *Recalls from the initial conditions that a communications are established with the Control Room and placekeeps step.*
4.	Selects a control rod at OR near the center of the core.	S (1.1.3)	SAT / UNSAT ***STD:*** *Determines from initiating cue that control rod 30-43 is to be inserted and placekeeps step.*
5.	Locates the HCU for the selected control rod	S (1.1.4)	SAT / UNSAT ***STD:*** *Determines that Control Rod 30-43 is to be inserted and locates Control Rod HCU 30-43 on the north Side of Reactor Building El 261'. Uses Figure 1, "RDS HCU Locations" as a guide, if necessary.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Procedure Note:	• The Withdraw and Insert lines and are located on the mezzanine above the HCUs. • The Withdraw line connects to the Over-piston Volume and is the smaller of the two lines. • Figure 13 shows the general arrangement and equipment connections listed in this sub-section.		
6.	On the mezzanine above selected HCU, removes the drain plug from 2RDS*V1, Withdraw Line Vent Valve. **Cue:** *The component you have identified is in the position you described.*	S (1.1.5)	***PASS / FAIL** ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the wrench discussed in section 4.1, slowly rotates the drain plug on 2RDS*V1, Withdraw Line Vent Valve in the counter clockwise direction until the drain plug is removed.* ***Failure =*** *Drain plug on 2RDS*V1 removal not achieved as determined by Operator verbalization.*
7.	Installs the quick disconnect adapter into 2RDS*V1. **Cue:** *The component you have identified is in the condition you have described.*	S (1.1.6)	SAT / UNSAT ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the quick disconnect adapter discussed in section 4.1, slowly threads the quick disconnect adapter into 2RDS*V1 by rotating the adapter in the clockwise direction until tight.*
8.	Connects the high-pressure hose to 2RDS*V1. **Cue:** *The component you have identified is in the condition you have described.*	S (1.1.7)	SAT / UNSAT ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the high pressure hose discussed in section 4.1, pushes and twists hose to the quick disconnect adapter on 2RDS*V1 until latched.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Evaluator Note:	In the JPM step below if the Operator starts to go down the ladder to secure the hose at the drain, tell them another operator has secured the bottom of the hose.		
9.	Routes the hose to the nearest Rx Bldg Drain. **Cue:** *The component you have identified is in the condition you have described.*	S (1.1.8)	SAT / UNSAT ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, routes the high pressure hose connected to 2RDS*V1 to a Rx Building floor drain.*
Procedure Warning:	The high pressure hose end should be secured at the drain to prevent flow induced hose whip.		
10.	Secures the hose at the drain to prevent flow induced hose whip. **Cue:** *The component you have identified is in the condition you have described.*	S (1.1.9)	SAT / UNSAT ***STD:*** *At RB Elev 261 north side, secures the high pressure hose connected to 2RDS*V1 to the Rx Building floor drain.*
11.	Removes the cap from 2RDS*V1 operator. **Cue:** *The component you have identified is in the position you described.*	S (1.1.10)	***PASS / FAIL** ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the wrench discussed in section 4.1, slowly rotates the cap on 2RDS*V1, Withdraw Line Vent Valve in the counter clockwise direction until the cap is removed.* ***Failure =*** *Cap on 2RDS*V1 removal not achieved as determined by Operator verbalization.*
12.	Monitors the motion of the selected control rod by selecting that control rod on the RMCS Rod Select Matrix (2CES*PNL603). **Cue:** *As the Control Room acknowledge the request and confirm that you will monitor control rod 30-43.*	S (1.1.11)	SAT / UNSAT ***STD:*** *At Control Rod HCU 30-43, repeats back cue using proper communications and placekeeps step.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
Procedure Caution:	Venting the over-piston area will result in Rx coolant being drained through the high-pressure hose connected to 2RDS*V1. This may create a high level of radioactive airborne as well as create high radiation conditions and contamination in local and general areas of RB 261'.		
13.	Using T handled HCU Vent Tool, slowly opens 2RDS*V1 **Cue:** *The component you have identified is in the position you described.* **Cue:** *If contacted as the control room, inform the operator that control rod 30-43 has fully inserted.*	S (1.1.12)	***PASS / FAIL** ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the T-handled HCU Vent Tool discussed in section 4.1, slowly rotates the T Handled Vent Tool attached to 2RDS*V1 in the counter clockwise direction until the valve is full open by local position indication and/or stem position.* ***Failure =*** *2RDS*V1 opening not achieved as determined by Operator verbalization.*
14.	WHEN control rod motion stops OR NO control rod movement is observed, THEN using T handled HCU Vent Tool, closes 2RDS*V1. **Cue:** *The component you have identified is in the position you described.*	S (1.1.13)	SAT / UNSAT ***STD:*** *At Control Rod HCU 30-43, on RB Elev 261 north side, using the T-handled HCU Vent Tool discussed in section 4.1, slowly rotates the T Handled Vent Tool attached to 2RDS*V1 in the clockwise direction until the valve is full closed by local position indication and/or stem position.*
Evaluator Note:	When 2RDS*V1 has been closed, provide the following cue: **Cue:** Your task is complete, another operator will complete any remaining actions.		

TASK STANDARD	*Control Rod 30-43 at notch 00 and 2RDS*V1 shut.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • A scram has occurred. • Several rods have not fully inserted. • NO CRD pumps are available. • Communications are established with Control Room. • An OD-7, Print out of Rod Positions is **NOT** available. • You have been given a F2-57 key.

INITIATING CUE	**(Operator Name)**, *insert control rod 30-43 to notch 00 by locally venting its overpiston area in accordance with N2-EOP-6.14, Flowchart* C

Plant JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • A scram has occurred. • Several rods have not fully inserted. • NO CRD pumps are available. • Communications are established with Control Room. • An OD-7, Print out of Rod Positions is **NOT** available. • You have been given a F2-57 key.

INITIATING CUE	**(Operator Name)**, *insert control rod 30-43 to notch 00 by locally venting its overpiston area in accordance with N2-EOP-6.14, Flowchart* C

Nine Mile Point Nuclear Station UNIT 2
EMERGENCY OPERATING PROCEDURE

N2-EOP-6.14

ALTERNATE CONTROL ROD INSERTIONS

Revision 00300

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required.

Approval Authority: Director - Operations

This procedure contains pages with COLOR

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
003	00	Minor Revision to incorporate:

PCR-20-04046

Description: Editorial changes to update references. Replaced NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document with N2-EOP-B-HOT, NMP2 Emergency Operating Procedures (EOP) Basis Document, N2-EOP-B-CSD/RF, NMP2 Emergency Operating Procedures for Cold Shutdown and Refueling (EOP-SCD/RF) Basis Document, N2-SAP-B-MODES 1-4, NMP2 Severe Accident Procedures (SAP) Basis Document, and N2-SAP-B-RF (MODE 5), NMP2 Severe Accident Procedures (SAP) Basis Document.

Reason: To implement EOP/SAG Rev. 4 changes.

Procedure Writer Review

- Applied Exelon Technical Procedure Template, updated Section and Step numbering, moved procedure information from page footer to header, **no** revision bars used.
- Updated N/A statements to comply with AD-NM-101-1005 5.4.C.13 and in preparation for the transition to Exelon Smart Procedures, by removing N/A check-boxes, made into **IF/THEN** statements and "**OTHERWISE MARK** this step (or these steps) N/A".
- Restructured multiple procedure steps to **ACTION VERB** first, **no** revision bars used.
- Updated fonts for **ACTION VERBS** and **LOGIC WORDS**, bold and lowercase underline for **not**, **cannot** and **no** per AD-NM-101-1005 and AD-AA-101-1002, **no** revision bars used.
- Moved locations from the end of action statements to the beginning.
- Steps 7.5.3 through 7.5.7, moved location to a proceeding note.
- Steps 7.7.1 and 7.7.2, converted SM actions into actual steps.
- Note after step 7.3.2, added to show steps continue on the next page.
- Note after step 7.4.2, added to show steps continue on the next page.
- Moved all figure numbers to the top of their respective pages. No revision bars used.
- **FIGURE 13**, added a title to the figure.

TABLE OF CONTENTS

SECTION		PAGE
1.0	PURPOSE	4
2.0	APPLICABILITY/SCOPE	4
3.0	REFERENCES AND DEFINITIONS	4
4.0	PREREQUISITES	5
5.0	PRECAUTIONS AND LIMITATIONS	7
6.0	PROCEDURE	7
7.0	RESTORATION	13
8.0	BASES	24
9.0	RECORDS	24

1.0 PURPOSE

To provide instructions for alternate methods of inserting control rods during Anticipated Transient Without a Scram (ATWS) conditions.

2.0 APPLICABILITY/SCOPE

2.1. When used to support N2-EOP-C5, Failure To Scram, or N2-SAP-2, RPV, Containment and Radioactivity Release Control, this procedure is performed, using one or more of the following methods, to insert control rods to shutdown the Reactor following an ATWS:

- Scram Solenoid De-energization, Hard Card Page 1
- Scram Air Header Venting, Hard Card Page 1
- Additional Manual Scram Initiations, Hard Card Page 2
- Manual Control Rod Insertion, Hard Card Page 3
- Control Rod Over-piston Volume Venting, Hard Card Pages 4, 5 and 6
- Individual Control Rod Scrams, Hard Card Pages 5 and 6

3.0 REFERENCES AND DEFINITIONS

3.1. Developmental References

- NUREG 1358, Lessons learned from the Special Inspection Program for EOPs
- OP-NM-114-101, EOP/SAP Revisions
- SER 90-145 Attachment 9, Engineering Analysis of the NMP2 Off Normal Operating Procedures
- NMP2 Plant Specific Technical Guidelines
- NMP2 Plant Specific Severe Accident Guidelines
- N2-EOP-B-HOT, NMP2 Emergency Operating Procedures (EOP) Basis Document
- N2-EOP-B-CSD/RF, NMP2 Emergency Operating Procedures for Cold Shutdown and Refueling (EOP-SCD/RF) Basis Document
- N2-SAP-B-MODES 1-4, NMP2 Severe Accident Procedures (SAP) Basis Document
- N2-SAP-B-RF (MODE 5), NMP2 Severe Accident Procedures (SAP) Basis Document
- NRC Correspondence - Emergency Operating Procedures Inspection and Initial Examination - Report No.50-410/91-80
- N2-SAP-2, RPV, Containment, and Radioactivity Release Control

3.2. Definitions

3.2.1. CHECK To observe an expected condition or characteristic; to determine; to ascertain.

3.2.2. ENSURE To confirm a condition. (No subsequent action is implied to establish that condition if not already there.)

3.2.3. GO TO 1. To proceed to, to transport oneself to a given location.

2. To discontinue use of present procedure or section and execute another procedure or section.

3.2.4. VERIFY To confirm a condition and take action to establish that condition if required.

4.0 PREREQUISITES

4.1. Special Tools and Equipment Recommended

NOTE

Keys PA235, PA1235, and PA2235 are interchangeable.

TOOL/MATERIAL	QTY	LOCATION
For Scram Solenoid De-energization per Hard Card Page 9 and Section 7.2		
L660 Key	1	Control Room EOP Toolbox
PA235 Keys	2	Control Room CRO Desk
For Additional Manual Scram Initiations per Hard Card Page 10 and Sections 7.1, 7.3, & 7.4		
L660 Key	1	Control Room EOP Toolbox
PA235 Keys	4	Control Room CRO Desk
Fuse Pullers	1	Control Room EOP Toolbox
Flashlight	1	Control Room EOP Toolbox
For Manual Control Rod Insertion per Hard Card Page 11 and Section 7.5		
SHH 5366 Key (For RWM)	1	SM Key Locker, Hook #151
L660 Key	1	Control Room EOP Toolbox
EOP Jumper #5	1	2CEC*PNL613
EOP Jumper #6	1	2CEC*PNL613
Continued on next page.		

4.1 (Continued)

TOOL/MATERIAL	QTY	LOCATION
For Control Rod Over-Piston Volume Venting per Hard Card Pages 12, 13, 14, and Section 7.6		
Leather Gloves	2	Control Room EOP Toolbox
Communications devices capable of communicating between the Control Room and the HCU areas	2	None, not reserved for EOP use only
Operator Aid for HCU Layout	1	RQ EOP Toolbox (*Rx Bldg, EL 261, near south HCUs*)
Operator Aid for Core Map	1	RQ EOP Toolbox
Tubes Pipe Sealant - Loctite Nuclear Grade PST	1	RQ EOP Toolbox
Wrench	1	RQ EOP Toolbox
T-handled HCU Vent Tool	1	RQ EOP Toolbox
High pressure Hose with Quick Disconnect Fitting	1	RQ EOP Toolbox
Quick Disconnect Adapter for HCU Vent Lines	1*	RQ EOP Toolbox
* 10 Quick Disconnects are provided in the toolbox to speed the setup process if there are sufficient operators available.		

5.0 PRECAUTIONS AND LIMITATIONS

5.1. The Restoration section shall be performed only when specifically directed by the SM/EOP Director. This permission shall not be granted until the system/equipment is in a condition to support restoration.

5.2. All tools, materials, keys, etc. that are required to perform this procedure are listed in Section 4.1.

5.3. A (T) notation in the left margin adjacent to a step number or note indicates that a tool or material is required for performance.

5.4. Common tools (screwdrivers, tape etc.) are not specified in procedure steps. Only special tools or situations where confusion may result have a particular tool specified in a step or note.

5.5. Independent verification is required in the Restoration section when restoring temporary alterations or returning permanent plant equipment to normal status. This verification may be delayed if emergency conditions still exist, and it is imperative that restoration be completed immediately. The EOP Director/SM permission is required to delay independent verification.

5.6. During plant conditions which require implementation of this procedure, environmental conditions may be potentially extreme (temperature, radiation, water levels).

In many cases this will require coordination and support from the OSC. Where access may be needed in areas of elevated temperatures, prudency dictates protective equipment be used and precautions taken. Stay times and activity levels should be minimized.

Consultation with the Safety Department is recommended when possible. Above 135°F personnel access may be significantly hampered.

When it is anticipated or known that radiation levels are elevated, radiation protection assistance should be sought. Some evolutions may require utilization of emergency exposure guidelines or emergency dosimetry.

5.7. Where applicable, this procedure provides locations of panels when installing jumpers and lifting leads. However, inside many panels there are Operator Aids providing amplifying instructions as to the exact locations of affected equipment.

5.8. Any fuse pulled per this procedure should be labeled and secured. Fuses should be marked with their Equipment Identification Numbers AND this procedure number AND stored in a safe place.

6.0 PROCEDURE

See Attached 11 x 17 Hard Cards

N2-EOP-6.14 - ALTERNATE CONTROL ROD INSERTIONS

Start

8 White Scram Solenoid Power Lights off?

Yes

No

Perform the following:

1. At 2CEC*PNL610, position POWER SOURCE SELECTOR switch to ALT A.
2. At 2CEC*PNL610, position POWER SOURCE SELECTOR switch to ALT B.
3. At 2CEC*PNL610, position POWER SOURCE SELECTOR switch to NORM.

8 White Scram Solenoid Power Lights off?

No

Yes

(Perform Concurrently)

Continue at D to De-energize Scram Solenoids with HCU SRI switches while continuing.

Scram Discharge Volume Vent & Drain Valves are OPEN?

No

Yes

Perform the following:

- (T) In 2CEC*PNL609, Bay A, pull FUSE C72A-F17A (Figure 5).
- (T) In 2CEC*PNL611, Bay A, pull FUSE C72A-F17B (Figure 6).
- At 2CEC*PNL603, observe RDS*AOV124/132 SCRAM DISCH VOLUME VENT VLVS closed.
- At 2CEC*PNL603, observe RDS*AOV123/130 SCRAM DISCH VOLUME DRAIN VLVS closed.

More than one rod withdrawn beyond 00?

Yes

No

Inform EOP Director

L660 Key

Blue SCRAM Lights Lit on Full Core Display

Yes

No

Concurrently perform the following:

- Additional manual scrams Go to A

AND

- Insert Control Rods Go to B

Vent Scram Air Header Using Division 1 & 2 ARI:

1. (T) In Relay Rm in C22-P001, put Div I ARI TEST SWITCH CH-A A320-S2 in TEST.
2. (T) In Relay Rm in C22-P001, put Div I ARI TEST SWITCH CH-B A420-S2 in TEST.
3. (T) In Relay Rm in C22-P002, put Div 2 ARI TEST SWITCH CH-A A320-S2 in TEST.
4. (T) In Relay Rm in C22-P002, put Div 2 ARI TEST SWITCH CH-B A420-S2 in TEST.
5. Observe P603 Div I ARI INIT light lit.
6. Observe P603 Div 2 ARI INIT light lit.

More than one rod withdrawn beyond 00?

No

Yes

Scram Air Header Pressurized?

No

Yes

Manually Vent the Scram Air Header:

1. At RB 261' ARI Valve Rack, CLOSE 2RDS-V595 Scram Pilot Air Hdr Isol.
2. At RB 261' East of MCC012, OPEN 2RDS-V43 Scram Air Header Drain.

More than one rod withdrawn beyond 00?

No

Yes

A

Additional Manual Scrams

Perform Concurrently with Driving Rods at B

(Perform Concurrently)

Detail 1: DE-energize/Fail Closed ARI Solenoids

1. (T) In Relay Room, C22-P001, Bay B, using L660 key, north side, lower door (see Figure 7), pull 20 amp fuses:
   - C22A-F5A
   - C22A-F6A
2. (T) In Relay Room, C22-P002, Bay B, using L660 key, north side, lower door (see Figure 8), pull 20 amp fuses:
   - C22A-F5B
   - C22A-F6B

IF Scram Discharge Volume Vent & Drain valves were de-energized by pulling fuses:

- (T) In 2CEC*PNL609, Bay A, install fuse C72A-F17A (Figure 5).
- (T) In 2CEC*PNL611, Bay A, install fuse C72A-F17B (Figure 6).

IF Scram Air Header Was Locally Isolated and Vented:

1. Close 2RDS-V43, Scram Air Header Drain (RB 261' by 2NHS-MCC012).
2. Open 2RDS-V595, Scram Pilot Air Header Isol (RB 261' by RDS FCVs).

Detail 2: Defeat RPS Interlocks:

- (T) At 2CEC*PNL609, using PA235 key, place keylock switch C72A-S10A, RPS A1 SCRAM LOGIC BYPASS in BYPASS
- (T) At 2CEC*PNL609, using PA235 key, place keylock switch C72A-S10C, RPS A2 SCRAM LOGIC BYPASS in BYPASS
- (T) At 2CEC*PNL611, using PA235 key, place keylock switch C72A-S10B, RPS B1 SCRAM LOGIC BYPASS in BYPASS
- (T) At 2CEC*PNL611, using PA235 key, place keylock switch C72A-S10D, RPS B2 SCRAM LOGIC BYPASS in BYPASS

IF RPM EPAs are tripped, reset in the following order:

1. 2RPM*ACB1A (CB 237' West Cable Area)
2. 2RPM*ACB2A (CB 237' West Cable Area)
3. 2RPM*ACB1B (CB 237' East Cable Area)
4. 2RPM*ACB2B (CB 237' East Cable Area)

IF 2RDS-V28, Charging Water Header Isolation was closed:

Open 2RDS-V28.

Reset RPS:

1. Reset REACTOR SCRAM RESET LOGIC A, B, C and D.
2. Verify the following:
   - Eight white PILOT SCRAM VALVE SOLENOIDS lights are lit.
   - 603306, CRD Scram Valve Pilot Air Header Press High/Low is clear.
   - 2RDS*AOV124/132, Scram Discharge Volume Vent Valves open.
   - 2RDS*AOV123/130, Scram Discharge Volume Drain Valves open.
   - Scram Discharge Volume is drained (603130, 603109, or 603409 clear).

Initiate a Manual Scram

OR

Perform **Detail 3** at D to De-energize Scram Solenoids with HCU Single Rod Insertion (SRI) Switches while continuing.

NOTE:

If full scrams result in little control rod motion individual rod scrams may be more effective as more drive force is applied to the rod, than during a full scram.

Did Control Rods move?

Yes

No

More than one rod withdrawn beyond 00?

Yes

No

Inform EOP Director

- Drive Control Rods per B concurrently with re-performing additional scrams.
- Vent Control Rod Overpiston Volume per C

B

Manual Control Rod Insertion

Perform Concurrently with Manual Scrams at A

1. Verify 2RDS-P1A, CRD Pump 1A, running.
2. Verify 2RDS-P1B, CRD Pump 1B, running.
3. Place 2RDS-FC107, CRD Flow Control, in MANUAL (M).
4. WHILE maintaining 2RDS-P1A AND 2RDS-P1B current < 40 amps, OPEN 2RDS-FC107.
5. Close 2RDS-PV101, Drive Wtr Press Control MOV, to maximize Drive Water ΔP.
6. (T) At 2CEC*PNL603, place the RWM in BYPASS with an SHH 5366 key (SM Key Locker Tag #151).
7. IF NOT performing Additional Manual Scrams, and IF needed to raise system flow OR Drive Water ΔP, close 2RDS-V28 Charging Water Header Isol.

Using Figure 2 Rod Insertion First Sequence, starting with a control rod at OR near the center:

1. Select a NOT FULL IN control rod.
2. Depress the RMCS CONTINUOUS INSERT pushbutton UNTIL control rod motion stops.
3. Repeat Steps 1 AND 2, working outward in a spiral pattern rapidly inserting control rods.

Using Figure 3 Rod Insertion Second Sequence, starting with a control rod at OR near the center:

1. Select a NOT FULL IN control rod.
2. Depress the RMCS CONTINUOUS INSERT pushbutton UNTIL control rod motion stops.
3. Repeat Steps 1 AND 2, working outward in a spiral pattern rapidly inserting control rods.

More than one rod withdrawn beyond 00?

No → Inform EOP Director

Yes →

- Re-perform Rod Insertion per this page, and/or
- Return to **Start** on page 1, and/or
- Perform Control Rod Over-Piston Volume Venting at C

NOTES:

- Control Rods should be inserted using Figure 2 and Figure 3 as a guide.
- If no rods have inserted, CRAM Rods can be inserted, followed by inserting Control Rods using the Shutdown Rod Sequence.
- Inserting Deep Control Rods (notch position 00 to 16) will have the greatest impact on reducing power.
- Based on Control Rod Positions or LPRM readings, if there is a region of the core that is not or is likely not shutdown, then the operator may insert Control Rods in that area as a priority.

Figure 2 **Rod Insertion First Sequence**

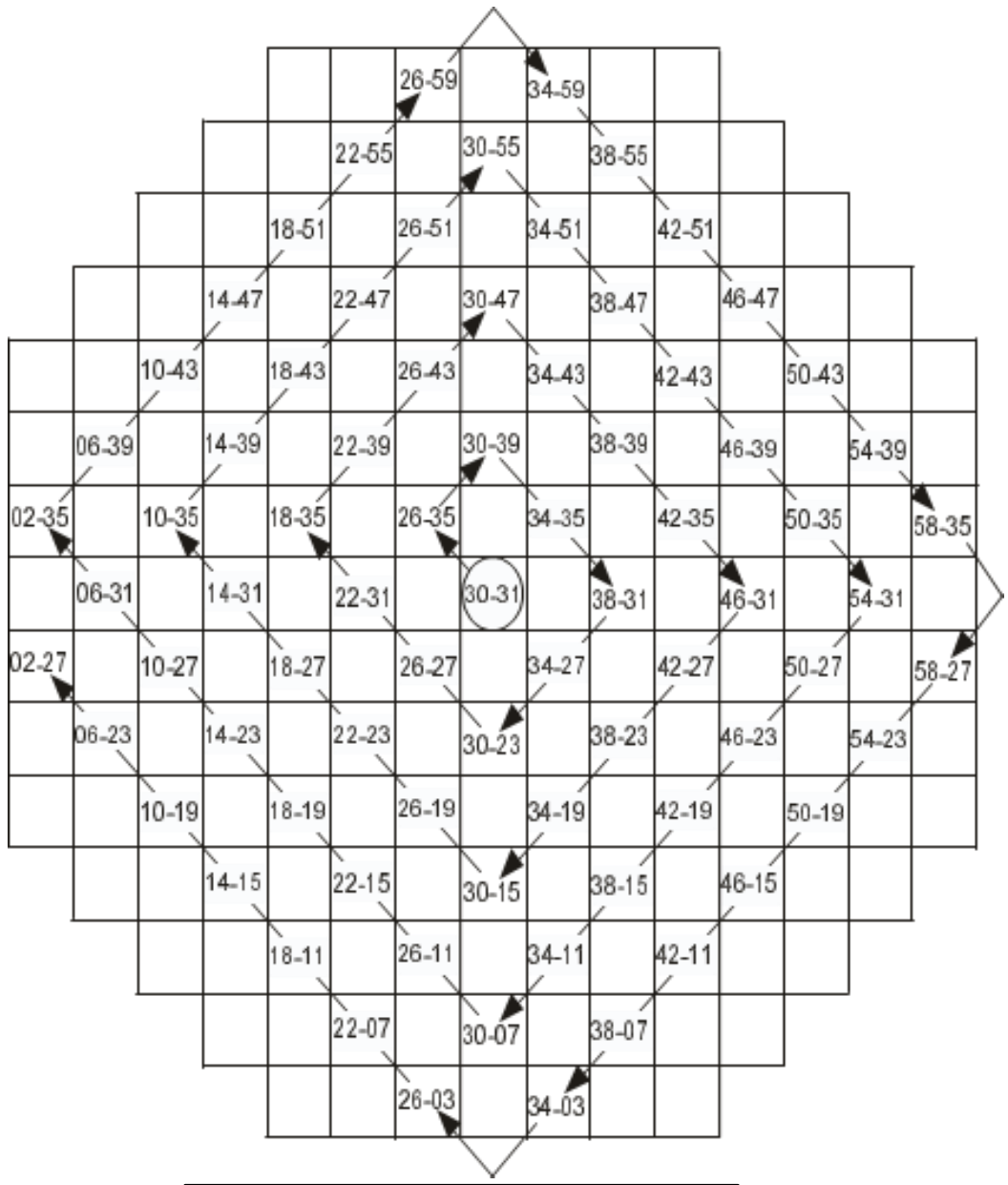
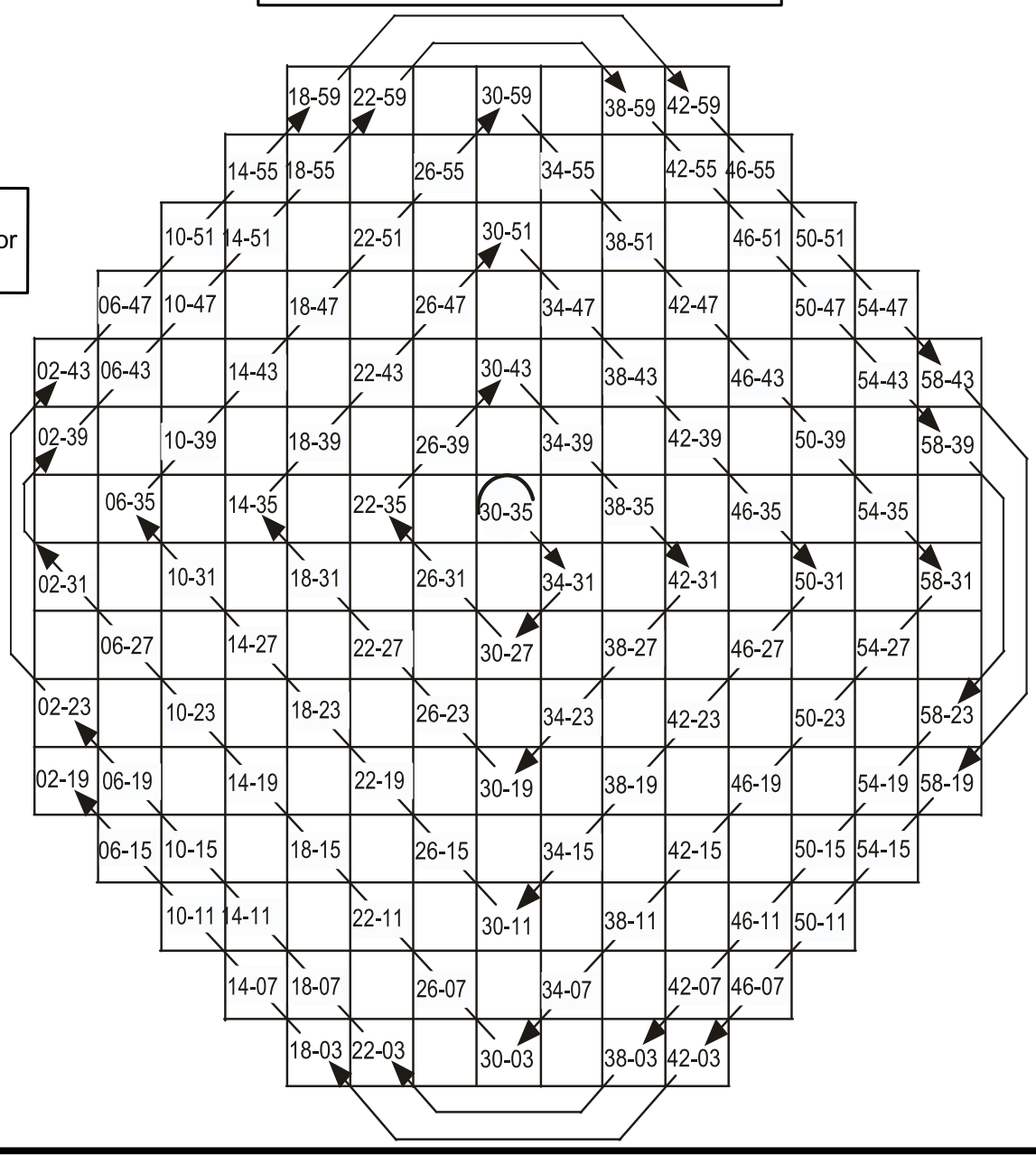


Figure 3 **Rod Insertion Second Sequence**



C

Control Rod Over-Piston Volume Venting

CAUTION

- This activity will result in draining of Rx coolant and poses the risk of creating a high level of radioactivity airborne as well as creating high radiation conditions and contamination in local and general areas of Rx Bldg El. 261.
- ALL appropriate protective measures must be taken for this activity as deemed necessary by Radiation Protection.

1.1.1 **IF** possible, obtain an OD-7 printout of rod positions from the R*Time Computer.

(T) 1.1.2 Establish communications between the Control Room **AND** the HCU Areas. (*RB 261'*)

1.1.3 Select a control rod at **OR** near the center of the core.

1.1.4 Locate the HCU for the selected control rod.

NOTE

- The Withdraw and Insert lines and are located on the mezzanine above the HCUs.
- The Withdraw line connects to the Over-piston Volume and is the smaller of the two lines.
- Figure 13 shows the general arrangement and equipment connections listed in this sub-section.

1.1.5 On the mezzanine above selected HCU, remove the drain plug from 2RDS*V1, Withdraw Line Vent Valve.

1.1.6 Install the quick disconnect adapter into 2RDS*V1.

1.1.7 Connect the high-pressure hose to 2RDS*V1.

1.1.8 Route the hose to the nearest Rx Bldg Drain.

WARNING

The high pressure hose end should be secured at the drain to prevent flow induced hose whip.

1.1.9 Secure the hose at the drain to prevent flow induced hose whip.

1.1.10 Remove the cap from 2RDS*V1 operator.

1.1.11 Monitor the motion of the selected control rod by selecting that control rod on the RMCS Rod Select Matrix (2CES*PNL603).

CAUTION

Venting the over-piston area will result in Rx coolant being drained through the high-pressure hose connected to 2RDS*V1. This may create a high level of radioactive airborne as well as create high radiation conditions and contamination in local and general areas of RB 261'.

(T) 1.1.12 Using T handled HCU Vent Tool, slowly open 2RDS*V1.

1.1.13 **WHEN** control rod motion stops **OR NO** control rod movement is observed,
THEN using T handled HCU Vent Tool, close 2RDS*V1.

1.1.14 Replace the cap on the operator for 2RDS*V1.

WARNING

Extreme caution should be exercised when removing the high pressure hose and connections. The draining of Rx coolant will result in the hose and connections being contaminated and may have caused them to be at a high temperature.

1.1.15 Disconnect the high-pressure hose from 2RDS*V1.

1.1.16 Remove the quick disconnect adapter from 2RDS*V1.

1.1.17 Remove the quick disconnect adapter from 2RDS*V1 **AND** move rig to the HCU for the next control rod to be inserted.

1.1.18 Select the next control rod to insert by working in an outward spiral pattern.

1.1.19 Repeat steps for each control rod to be inserted.

1.1.20 **WHEN** ROD INSERTION, has been completed,
THEN check position of ALL control rods. (2CEC*PNL603). **IF ALL** rods except one are full in, notify EOP Director.

D

Detail 3

De-energize Scram Solenoids with HCU Single Rod Insertion (SRI) Switches:

At RB 261' HCUs, de-energize individual HCU Scram Solenoids:

- Place upper SRI TEST switch to TEST position.
- Place lower SRI TEST switch to TEST position.

Table 1 Control Rod SRI Test Switch SCRAM Sequence and Locations

Sequence	Control Rod	North or South Side	SRI Switches (✓)	Sequence	Control Rod	North or South Side	SRI Switches (✓)	Sequence	Control Rod	North or South Side	SRI Switches (✓)	Sequence	Control Rod	North or South Side	SRI Switches (✓)	Sequence	Control Rod	North or South Side	SRI Switches (✓)
1	30-31	North		43	30-07	South		85	34-39	North		127	58-31	North		148	38-59	North	
2	26-35	South		44	26-11	South		86	38-35	North		128	54-27	North		149	42-55	North	
3	30-39	North		45	22-15	South		87	42-31	North		129	50-23	North		150	46-51	North	
4	34-35	North		46	18-19	South		88	38-27	North		130	46-19	North		151	50-47	North	
5	38-31	North		47	14-23	South		89	34-23	North		131	42-15	North		152	54-43	North	
6	34-27	North		48	10-27	South		90	30-19	South		132	38-11	North		153	58-39	North	
7	30-23	South		49	06-31	South		91	26-23	South		133	34-07	North		154	58-23	North	
8	26-27	South		50	02-35	South		92	22-27	South		134	30-03	South		155	54-19	North	
9	22-31	South		51	06-39	South		93	18-31	South		135	26-07	South		156	50-15	North	
10	18-35	South		52	10-43	South		94	14-35	South		136	22-11	South		157	46-11	North	
11	22-39	South		53	14-47	South		95	18-39	South		137	18-15	South		158	42-07	North	
12	26-43	South		54	18-51	South		96	22-43	South		138	14-19	South		159	38-03	North	
13	30-47	North		55	22-55	South		97	26-47	South		139	10-23	South		160	22-03	South	
14	34-43	North		56	26-59	South		98	30-51	North		140	06-27	South		161	18-07	South	
15	38-39	North		57	34-59	North		99	34-47	North		141	02-31	South		162	14-11	South	
16	42-35	North		58	38-55	North		100	38-43	North		142	02-39	South		163	10-15	South	
17	46-31	North		59	42-51	North		101	42-39	North		143	06-43	South		164	06-19	South	
18	42-27	North		60	46-47	North		102	46-35	North		144	10-47	South		165	02-23	South	
19	38-23	North		61	50-43	North		103	50-31	North		145	14-51	South		166	02-43	South	
20	34-19	North		62	54-39	North		104	46-27	North		146	18-55	South		167	06-47	South	
21	30-15	South		63	58-35	North		105	42-23	North		147	22-59	South		168	10-51	South	
22	26-19	South		64	58-27	North		106	38-19	North		148	38-59	North		169	14-55	South	
23	22-23	South		65	54-23	North		107	34-15	North		149	42-55	North		170	18-59	South	
24	18-27	South		66	50-19	North		108	30-11	South		150	46-51	North		171	42-59	North	
25	14-31	South		67	46-15	North		109	26-15	South		151	50-47	North		172	46-55	North	
26	10-35	South		68	42-11	North		110	22-19	South		152	54-43	North		173	50-51	North	
27	14-39	South		69	38-07	North		111	18-23	South		153	58-39	North		174	54-47	North	
28	18-43	South		70	34-03	North		112	14-27	South		154	58-23	North		175	58-43	North	
29	22-47	South		71	26-03	South		113	10-31	South		155	54-19	North		176	58-19	North	
30	26-51	South		72	22-07	South		114	06-35	South		156	50-15	North		177	54-15	North	
31	30-55	North		73	18-11	South		115	10-39	South		157	46-11	North		178	50-11	North	
32	34-51	North		74	14-15	South		116	14-43	South		158	42-07	North		179	46-07	North	
33	38-47	North		75	10-19	South		117	18-47	South		159	38-03	North		180	42-03	North	
34	42-43	North		76	06-23	South		118	22-51	South		160	22-03	South		181	18-03	South	
35	46-39	North		77	02-27	South		119	26-55	South		161	18-07	South		182	14-07	South	
36	50-35	North		78	30-35	North		120	30-59	North		162	14-11	South		183	10-11	South	
37	54-31	North		79	34-31	North		121	34-55	North		163	10-15	South		184	06-15	South	
38	50-27	North		80	30-27	South		122	38-51	North		164	06-19	South		185	02-19	South	
39	46-23	North		81	26-31	South		123	42-47	North		165	02-23	South					
40	42-19	North		82	22-35	South		124	46-43	North		166	02-43	South					
41	38-15	North		83	26-39	South		125	50-39	North		167	06-47	South					
42	34-11	North		84	30-43	North		126	54-35	North		168	10-51	South					

7.0 RESTORATION

NOTE

- This section is not performed until specifically directed by the SM/EOP Director. This permission shall not be granted until the system/equipment is in a condition to support restoration.
- Independent verification may be delayed until emergency conditions no longer exist per SM/EOP Director.

7.1. Scram Solenoid Power Restoration

7.1.1. **IF** the Reactor Protection System (RPS) Electrical Protection Assemblies (EPAs) are tripped,
THEN on Control Building El 237'
RESET the RPS EPAs in the following order,
OTHERWISE MARK these steps N/A:

1. At West Cable Tray Aisle, 2RPM*ACB1A..... ☐
2. At West Cable Tray Aisle, 2RPM*ACB2A..... ☐
3. At East Cable Tray Aisle, 2RPM*ACB1B..... ☐
4. At East Cable Tray Aisle, 2RPM*ACB2B..... ☐ _____/_____

_____/_____
IV

7.1.2. **IF** the power fuses for the Scram Discharge Volume (SDV) vent and drain valves were removed,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

NOTE

(T) A L660 key may be needed to gain entry to 2CEC*PNL609 and 2CEC*PNL611.

(T) • In 2CEC*PNL609, Bay A**,**
VERIFY installed Fuse C72A-F17A. (Figure 5).....☐

(T) • In 2CEC*PNL611, Bay A**,**
VERIFY installed Fuse C72A-F17B. (Figure 6).....☐ _____/_____

_____/_____
IV

7.1.3. **IF** control rod HCU SRI TEST switches were used to insert control rods,
THEN VERIFY the following at ALL Control rod HCUs,
OTHERWISE MARK these steps N/A:

- Upper SRI TEST switch is in the NORM position .....☐
- Lower SRI TEST switch is in the NORM position .....☐ _____/_____

_____/_____
IV

7.2. **Scram Air Header Restoration**

7.2.1. **IF** the Scram Air Header was vented using the RRCS ARI Test switches,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

NOTE

(T) A L660 key may be needed to gain entry to C22-P001 and C22-P002.

1. Using a PA235 key,
PERFORM the following:

(T) • Inside C22-P001,
VERIFY Division I ARI TEST SWITCH CH-A (A320-S2) in the Normal position..... ☐

(T) • Inside C22-P001,
VERIFY Division I ARI TEST SWITCH CH-B (A420-S2) in the Normal position..... ☐

(T) • Inside C22-P002,
VERIFY Division II ARI TEST SWITCH CH-A (A320-S2) in the Normal position..... ☐

(T) • Inside C22-P002,
VERIFY Division II ARI TEST SWITCH CH-B (A420-S2) in the Normal position..... ☐ _____/_____

_____/_____
IV

2. At 2CEC*PNL603,
ENSURE the Division I ARI INIT light is out. _____/_____

3. At 2CEC*PNL603,
ENSURE the Division II ARI INIT light is out. _____/_____

Initials/Date

7.2.2. **IF** the Scram Air Header was manually vented,
THEN PERFORM the following,
OTHERWISE MARK this step N/A:

1. On Rx Bldg, EL261 by east end of 2NHS-MCC012, **VERIFY** 2RDS-V43, Scram Air Header Drain, is CLOSED. ..... ☐

2. On Rx Bldg, EL261, on ARI Solenoid Valve Rack, **VERIFY** 2RDS-V595, Scram Pilot Air Header Isol, is OPEN **AND** LEAD SEALED. ..... ☐ _____/_____

_____/_____
IV

7.3. **Additional Manual Scram Restoration**

NOTE

A L660 key may be needed to gain entry to C22-P001 and C22-P002.

(T)

(T) 7.3.1. In Relay Room, C22-P001, Bay B, north side, lower door, **INSTALL** the following 20-amp fuses (Figure 7):

- C22A-F5A ..... ☐
- C22A-F6A ..... ☐ _____/_____

_____/_____
IV

(T) 7.3.2. In Relay Room, C22-P002, Bay B, north side, lower door, **INSTALL** the following 20-amp fuses (Figure 8):

- C22A-F5B ..... ☐
- C22A-F6B ..... ☐ _____/_____

_____/_____
IV

NOTE

Section 7.3 is continued on the next page.

Initials/Date

7.3.3. **RESTORE** RPS interlocks as follows:

(T) • At 2CEC*PNL609,
PLACE switch C72A-S10A, RPS A1 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS A1 SCRAM LOGIC BYPASS C72A-DS24A light extinguished. ☐

(T) • At 2CEC*PNL609,
PLACE switch C72A-S10C, RPS A2 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS A2 SCRAM LOGIC BYPASS C72A-DS24C light extinguished. ☐

(T) • At 2CEC*PNL611,
PLACE switch C72A-S10B, RPS B1 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS B1 SCRAM LOGIC BYPASS C72A-DS24B light extinguished. ☐

(T) • At 2CEC*PNL611,
PLACE switch C72A-S10D, RPS B2 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS B2 SCRAM LOGIC BYPASS C72A-DS24D light extinguished. ☐ _____/_____

_____/_____
IV

7.4. **Individual Control Rod Scram Restoration**

NOTE

A L660 key may be needed to gain entry to C22-P001 and C22-P002.

(T) 7.4.1. In Relay Room, C22-P001, Bay B, north side, lower door, **INSTALL** the following 20-amp fuses (Figure 7):

- C22A-F5A ..... ☐
- C22A-F6A ..... ☐ _____/_____

_____/_____
IV

(T) 7.4.2. In Relay Room, C22-P002, Bay B, north side, lower door, **INSTALL** the following 20-amp fuses (Figure 8):

- C22A-F5B ..... ☐
- C22A-F6B ..... ☐ _____/_____

_____/_____
IV

NOTE

Section 7.4 is continued on the next page.

Initials/Date

7.4.3. **RESTORE** RPS interlocks as follows:

(T) • At 2CEC*PNL609,
PLACE switch C72A-S10A, RPS A1 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS A1 SCRAM LOGIC BYPASS C72A-DS24A light extinguished. ☐

(T) • At 2CEC*PNL609,
PLACE switch C72A-S10C, RPS A2 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS A2 SCRAM LOGIC BYPASS C72A-DS24C light extinguished ☐

(T) • At 2CEC*PNL611,
PLACE switch C72A-S10B, RPS B1 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS B1 SCRAM LOGIC BYPASS C72A-DS24B light extinguished. ☐

(T) • At 2CEC*PNL611,
PLACE switch C72A-S10D, RPS B2 SCRAM LOGIC BYPASS, in NORMAL
AND CONFIRM amber RPS B2 SCRAM LOGIC BYPASS C72A-DS24D light extinguished ☐ _____/_____

_____/_____
IV

7.4.4. **VERIFY** the following on ALL control rod HCUs:

• Upper SRI TEST switch is in the NORM position ☐
• Lower SRI TEST switch is in the NORM position ☐ _____/_____

_____/_____
IV

Initials/Date

7.5. **Manual Control Rod Insertion Restoration**

7.5.1. At Rx Bldg, EL261, RDS Flow Control Station, above 2RDS-FLT2A/2B,
VERIFY 2RDS-V28, CHARGING WATER HEADER ISOL, is OPEN. _____/_____

_____/_____
IV

NOTE

Steps 7.5.2 through 7.5.7 are performed at 2CEC*PNL603.

7.5.2. On 2CEC*PNL603,
VERIFY the Setpoint Tape for 2RDS-FC107, CRD FLOW CONTROL, is set to 63 gpm. _____/_____

_____/_____
IV

7.5.3. **LOWER** Rod Drive System (RDS) flow by depressing the CLOSE pushbutton on 2RDS-FC107
UNTIL the red needle is zeroed with the line over the Setpoint meter. _____/_____

_____/_____
IV

7.5.4. **PLACE** AUTO/MANUAL selector switch for 2RDS-FC107 in the AUTO position. _____/_____

_____/_____
IV

7.5.5. **VERIFY** only one RDS Pump is running. _____/_____

_____/_____
IV

7.5.6. **ADJUST** RDS Drive Water ΔP to about 260 psid by opening 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV. _____/_____

_____/_____
IV

Initials/Date

7.5.7. Using an SHH 5366 key,
PLACE the RWM BYPASS/OPERATE/TEST switch in the OPERATE position. _____/_____

_____/_____
IV

7.6. **Control Rod Overpiston Volume Venting Restoration**

NOTE

The following step is performed at Rx Bldg, EL 261, on mezzanine above the affected HCUs. See Figure 1, RDS HCU LOCATIONS.

CAUTION

Extreme caution should be exercised when removing the high-pressure hose and connections. The draining of Rx coolant will result in the hose and connections being contaminated and may have caused them to be at a high temperature.

7.6.1. **PERFORM** the following for ALL control rod HCUs which had the Over-piston Volume vented:

(T) 1. **REMOVE** the cap from 2RDS*V1 operator ..... ☐

(T) 2. Using T-handled HCU Vent Tool, **VERIFY** 2RDS*V1 is CLOSED. ..... ☐

(T) 3. **DISCONNECT** the high-pressure hose from 2RDS*V1 .. ☐

(T) 4. **REMOVE** the quick disconnect adapter from 2RDS*V1 . ☐

(T) 5. **REPLACE** the cap on the operator for 2RDS*V1 ..... ☐

(T) 6. Using Pipe Sealant, **REPLACE** the cap on the drain for 2RDS*V1..... ☐ _____/_____

_____/_____
IV

7.7. **SM Review**

7.7.1. SM **VERIFY** that restoration is complete..... ☐

7.7.2. **RECORD** comments in Remarks below..... ☐

Remarks: _____

_____

_____

_____

_____

_____/_____/_____
SM Signature Date Time

8.0 BASES

None

9.0 RECORDS

None

FIGURE 1, RDS HCU LOCATIONS

NORTH

2HVR*UC414A

				30 43	30 39	30 35	30 31	34 27	34 23										
		30 51	30 47	34 43	34 39	34 35	34 31	38 27	38 23	34 19									
		34 51	34 47	38 43	38 39	38 35	38 31	42 27	42 23	38 19	34 15			34 11	34 03				
30 59	30 55	38 51	38 47	42 43	42 39	42 35	42 31	46 27	46 23	42 19	38 15			38 11	38 03				
34 59	34 55	42 51	42 47	46 43	46 39	46 35	46 31	50 27	50 23	46 19	42 15			42 11	42 03				
38 59	38 55	46 51	46 47	50 43	50 39	50 35	50 31	54 27	54 23	50 19	46 15			46 11	34 07				
42 59	42 55	50 51	50 47	54 43	54 39	54 35	54 31	58 27	58 23	54 19	50 15			50 11	38 07				
	46 55		54 47	58 43	58 39	58 35	58 31			58 19	54 15			46 07	42 07				

SOUTH

HPU-A

HPU-B

2CES*RAK107

							02 27	02 23	02 19	06 15				14 07					
18 55	14 55		02 43	02 39	02 35	02 31	06 27	06 23	06 19	10 15	10 11			18 07	18 03				
22 55	10 51	06 47	06 43	06 39	06 35	06 31	10 27	10 23	10 19	14 15	14 11			22 07	22 03				
26 55	14 51	10 47	10 43	10 39	10 35	10 31	14 27	14 23	14 19	18 15	18 11			26 07	26 03				
18 59	18 51	14 47	14 43	14 39	14 35	14 31	18 27	18 23	18 19	22 15	22 11			30 07	30 03				
22 59	22 51	18 47	18 43	18 39	18 35	18 31	22 27	22 23	22 19	26 15	26 11								
26 59	26 51	22 47	22 43	22 39	22 35	22 31	26 27	26 23	26 19	30 15	30 11								
		26 47	26 43	26 39	26 35	26 31	30 27	30 23	30 19										

FIGURE 1

FIGURE 2, ROD INSERTION (FIRST SEQUENCE)

NOTE

- Control Rods should be inserted using Figure 2 and Figure 3 as a guide.
- If no rods have inserted, this can be achieved by inserting the CRAM rods per N2-SOP-101D, followed by inserting the Control Rods using the Shutdown Control Rod Sequence.
- Insertion of Deep Control Rods (notch position 00 to 16) will have the greatest impact on reducing power.
- Based on Control Rod Positions and/or LPRM readings, if there is a region of the core that is not or is likely not shutdown, then the operator may insert Control Rod in that area as a priority.

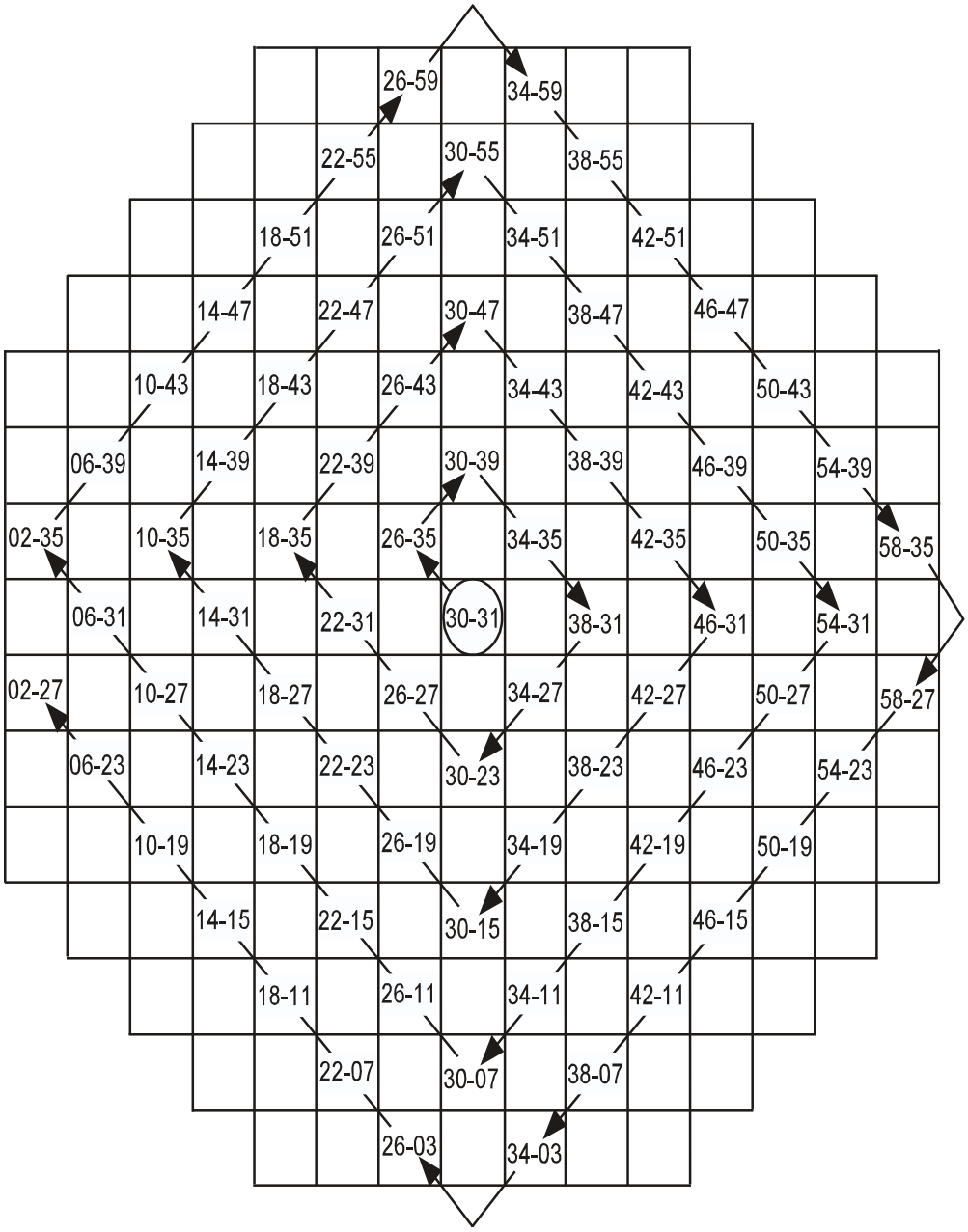


FIGURE 3, ROD INSERTION (SECOND SEQUENCE)

NOTE

- Control Rods should be inserted using Figure 2 and Figure 3 as a guide.
- If no rods have inserted, this can be achieved by inserting the CRAM rods per N2-SOP-101D, followed by inserting the Control Rods using the Shutdown Control Rod Sequence.
- Insertion of Deep Control Rods (notch position 00 to 16) will have the greatest impact on reducing power.
- Based on Control Rod Positions and/or LPRM readings, if there is a region of the core that is not or is likely not shutdown, then the operator may insert Control Rod in that area as a priority.

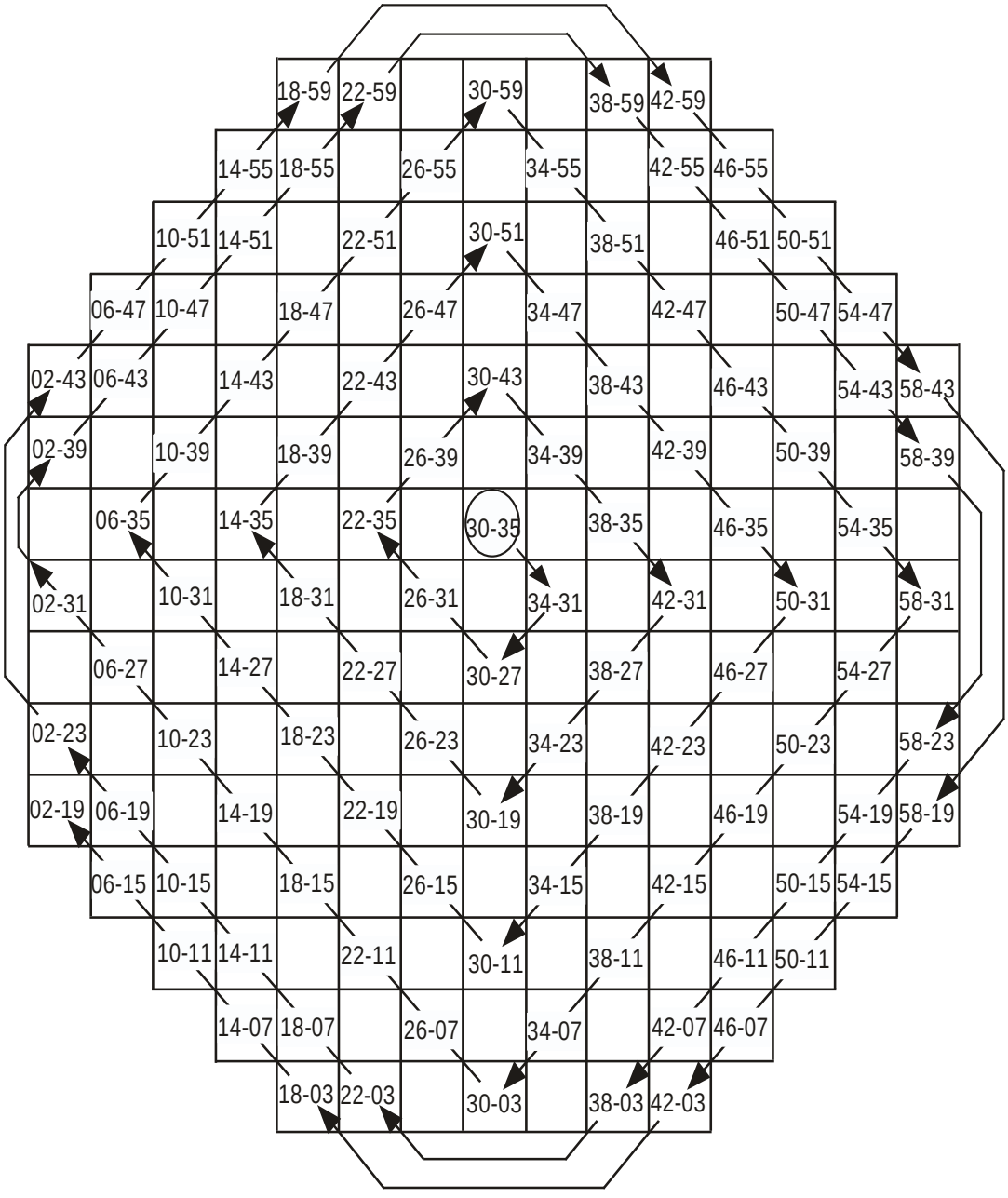


FIGURE 5, 2CEC*PNL609 BAY A

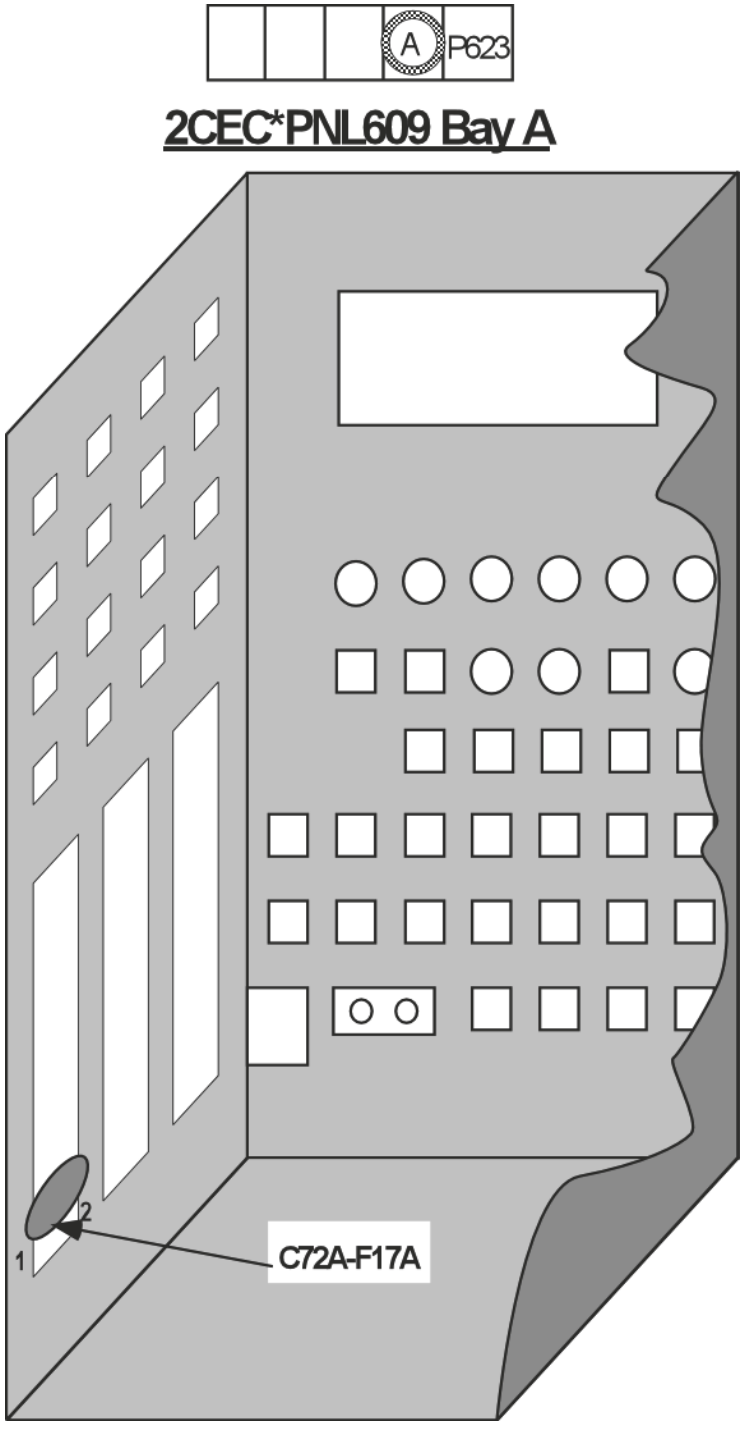


FIGURE 6, 2CEC*PNL611 BAY A

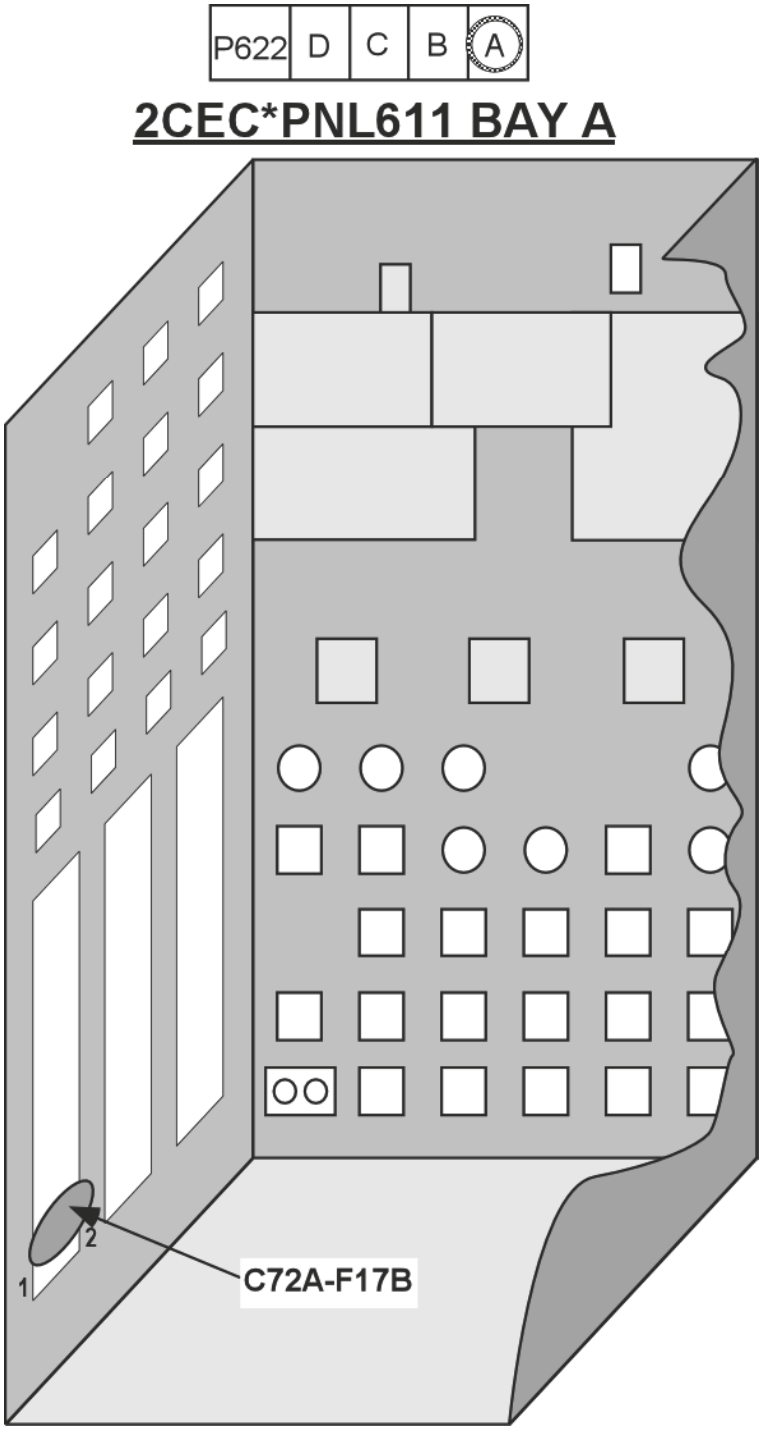


FIGURE 7, C22-P001-2

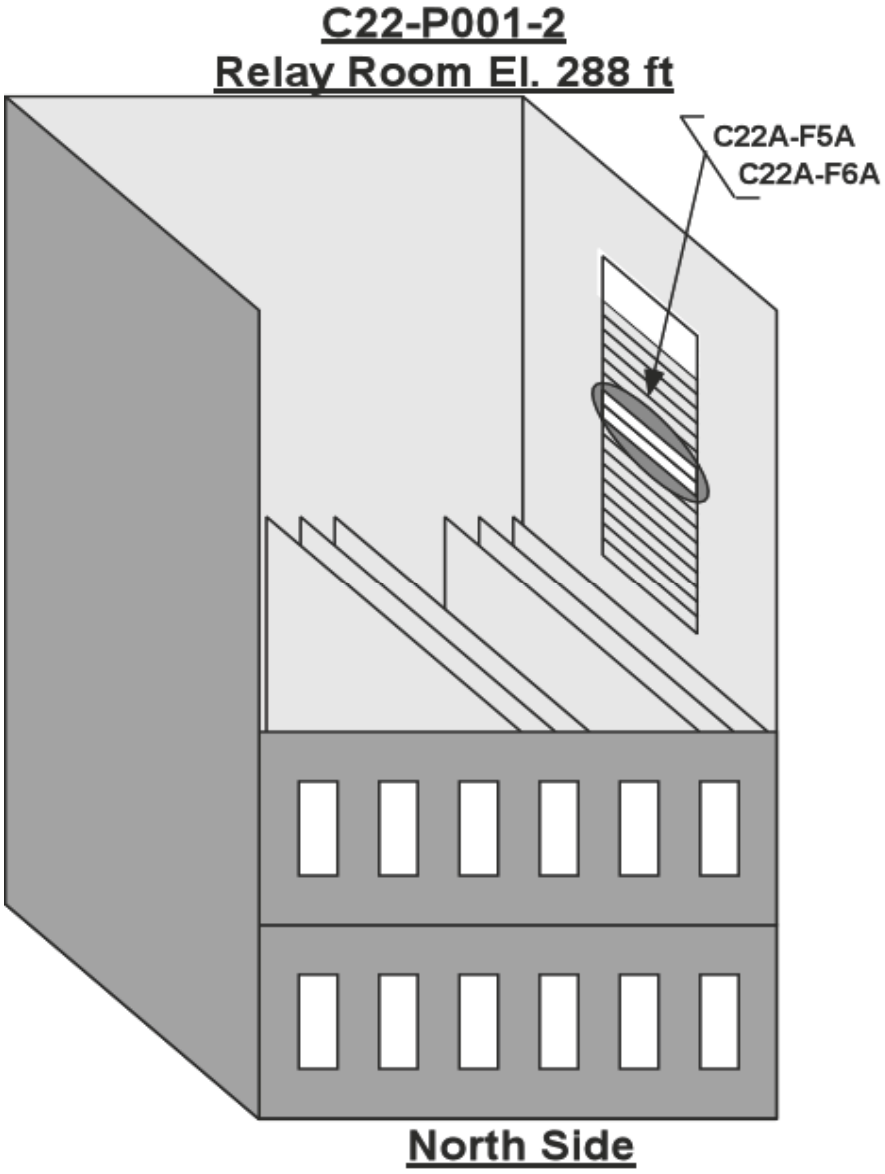


FIGURE 8, C22-P002-2

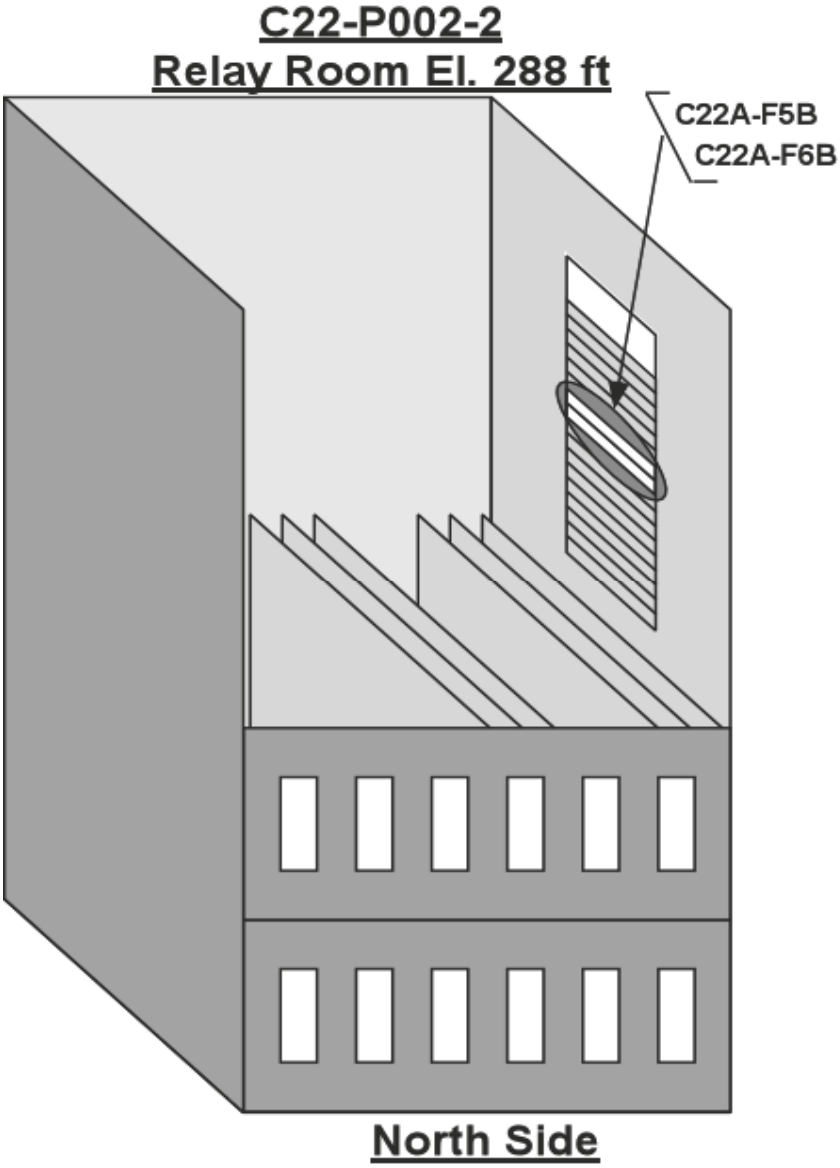


FIGURE 9, 2CEC*PNL609 BAY A

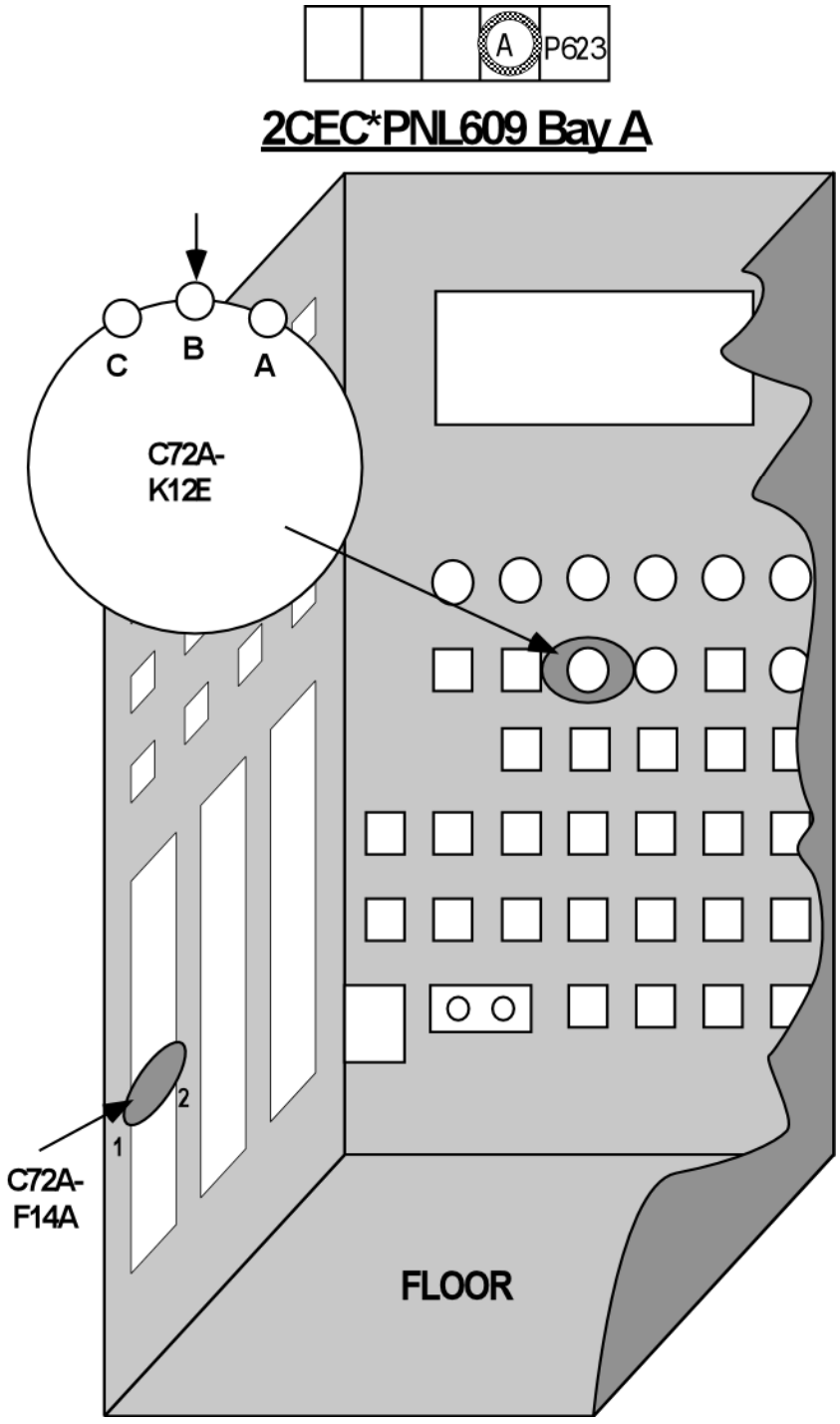


FIGURE 10, 2CEC*PNL609 BAY D

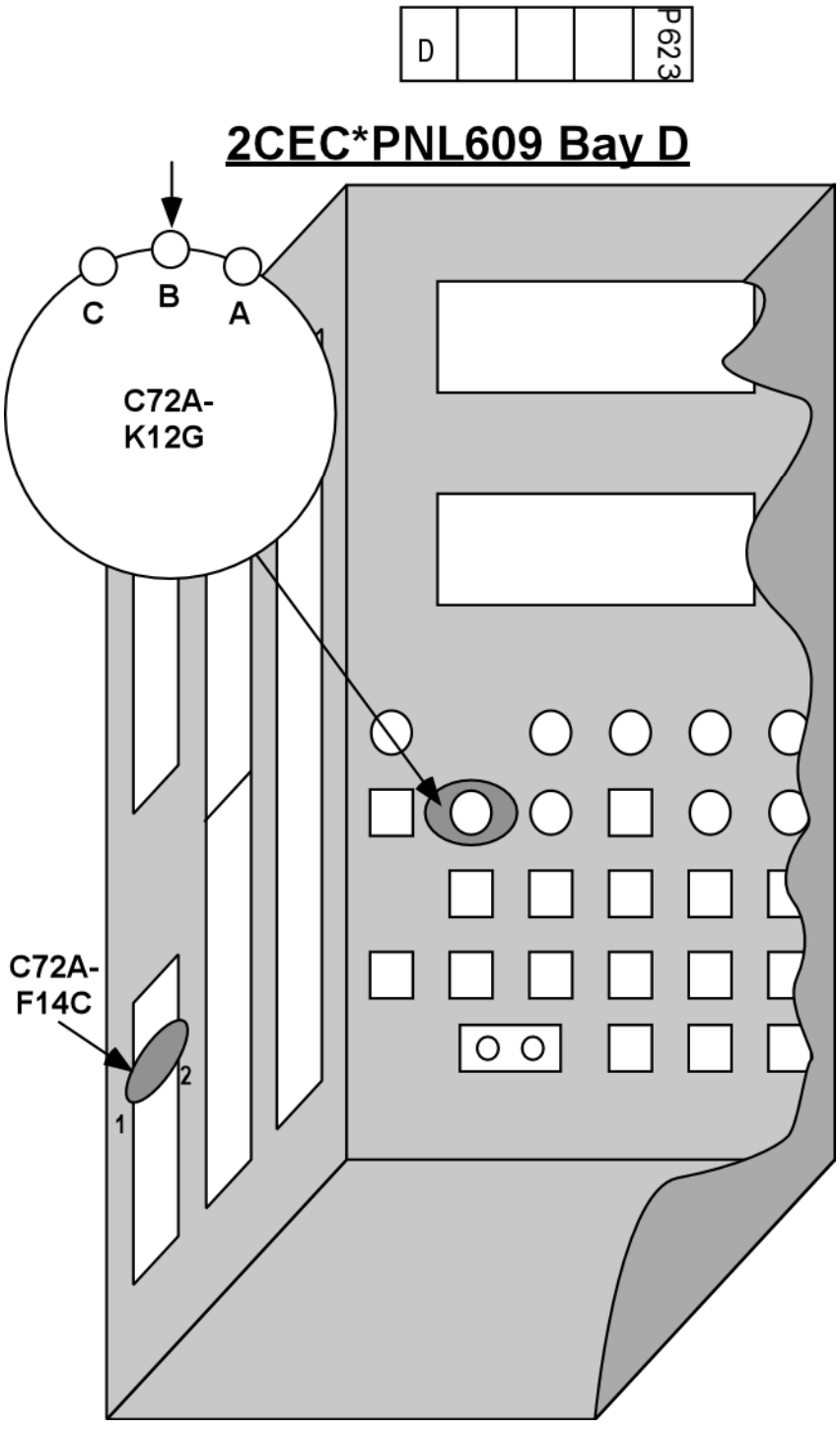


FIGURE 11, 2CEC*PNL611 BAY A

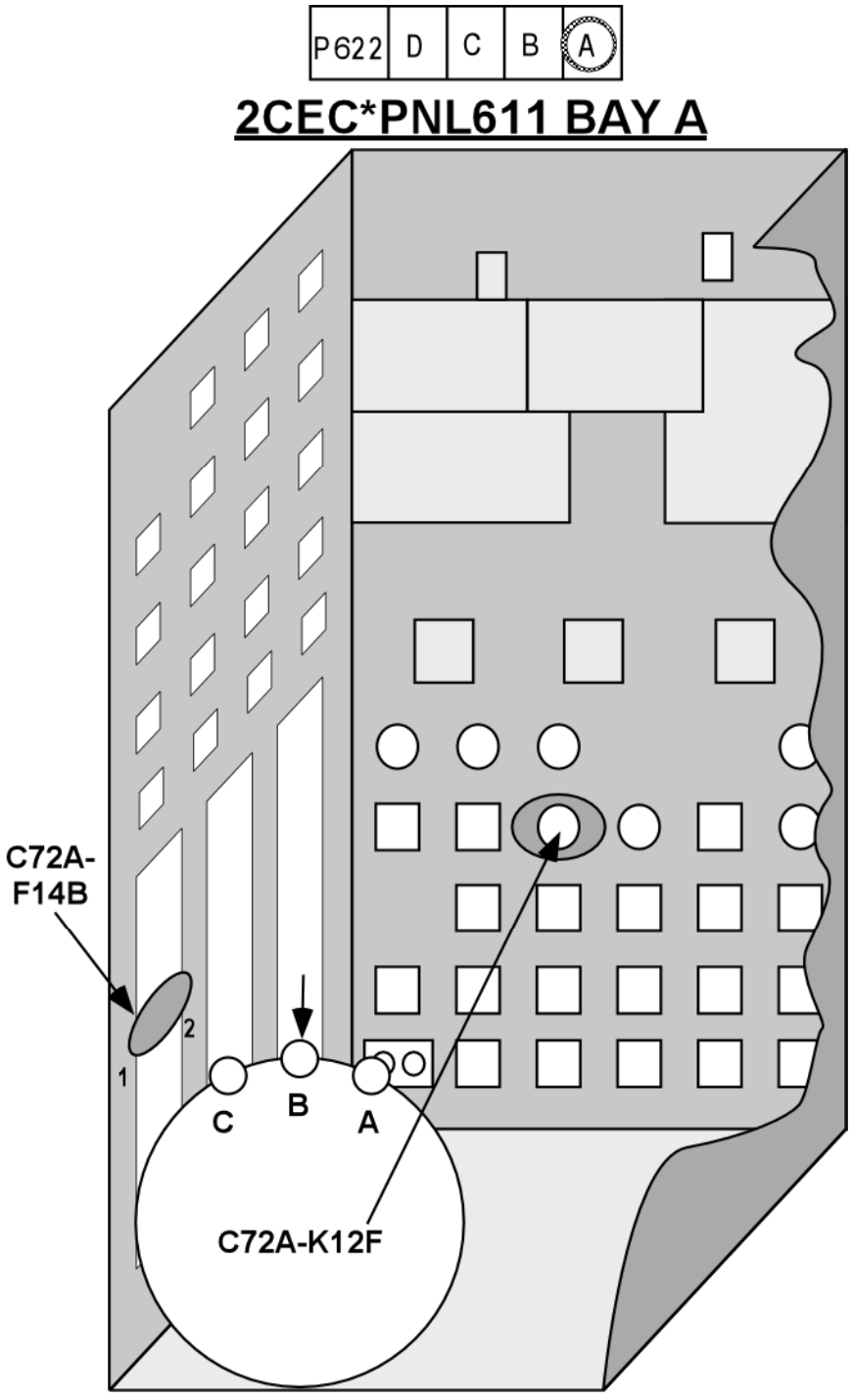


FIGURE 12, 2CEC*PNL611 BAY D

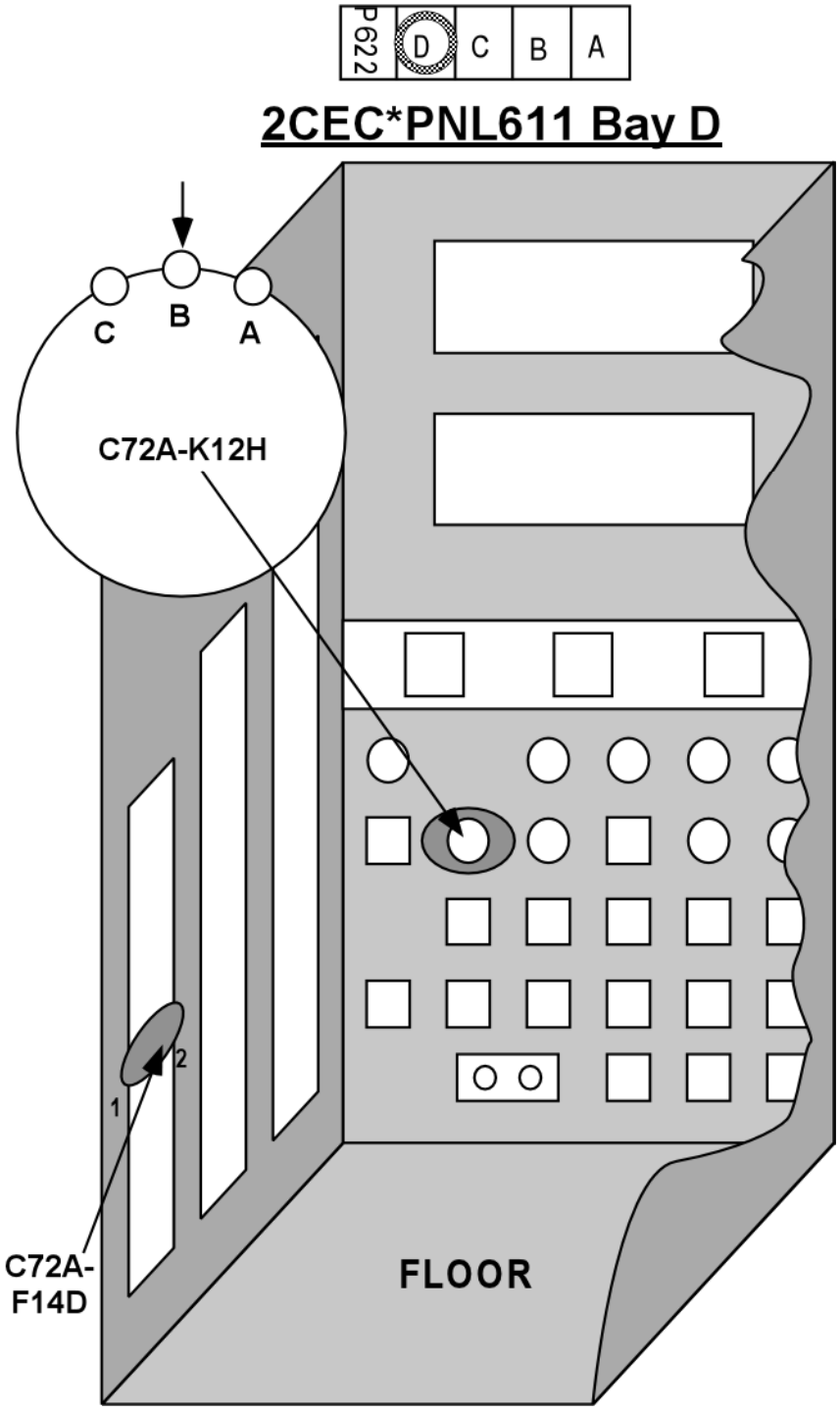
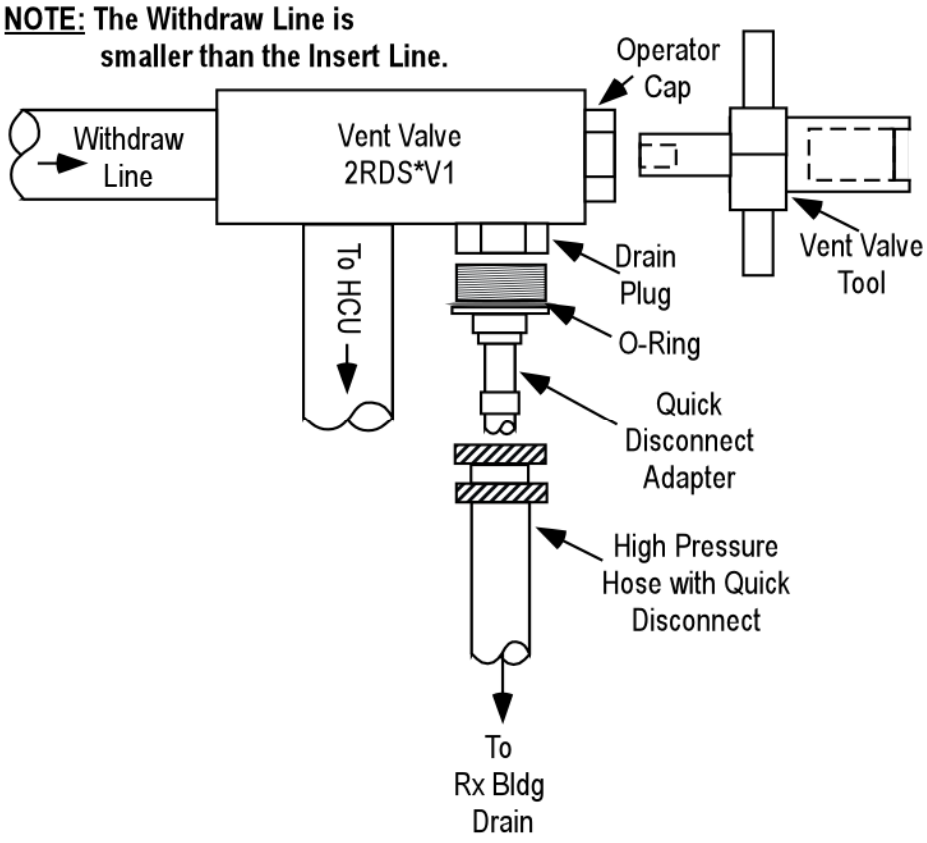


FIGURE 13, CONTROL ROD OVER-PISTON VOLUME VENTING EQUIPMENT



Constellation

Training ID: **Nov 2023 NRC JPM K** Revision: **0.0**

Title: **Recover Offgas after Automatic Shutdown**

Approvals:

	Signature	/	Printed Name	Date
Developed by:	Signature on File	/	Mike Alexander	10/27/23
Validated by:	N/A		Mike Carno	7/14/23
Facility Reviewer:	Signature on File		John Toothaker	10/27/23

Approximate Duration: 25 minutes

Documentation of Performance:

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time: _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

References

1. N2-OP-42, Offgas System
2. NUREG 1123 K/A 271000 A2.11 (3.5/3.0)

Instructor Information

A. JPM Information

1. Description

   a. This JPM tests the Operator's ability to restore the Offgas system after an automatic shutdown.

   b. This JPM is not considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N2-217100-01013, Recover Offgas System Following Automatic Shutdown.

3. Evaluation / Task Criteria

License Level: (Target Audience)	☒ SRO ☒ RO ☐ EO ☐ N/A				
Evaluation Method:	☐ Perform ☒ Simulate				
Evaluation Location:	☒ Plant ☐ Simulator ☐ Classroom				
Time Critical Task:	☐ Yes ☒ No				
Alternate Path:	☐ Yes ☒ No				
Safety Function:	9		Radioactivity Release		
LOD Value: (Must be >1.0) (Ref. CERT 71111.11 Inspection Manual, APP. C)		3			
K/A Statement: (Add justification statement below for K/A's < 3.0)	271000 Offgas System A2.11 - Ability to (a) predict the impacts of the following on the Offgas System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Offgas system low flow				
K/A Importance Rating:	RO	3.5	SRO	3.0	

4. Recommended Start Location
    a. Unit #2 R.P. Access Building
5. Simulator Setup
    a. IC Number
        a. N/A
    b. Presets / With Triggers
        a. Malfunctions
            a) None
        b. Remotes
            a) None
        c. Overrides
            a) None
        d. Annunciators
            a) None
        e. Event Triggers

Event #	Event Action	Command
None	N/A	N/A

        f. Equipment Out of Service
            a) None
        g. Support Documentation
            a) Prepare a copy of N2-OP-42 section H.1.0 with steps H.1.1-H.1.3 placekept as completed. Include a copy of the precautions and limitations.
        h. Miscellaneous
            a) None

6. Strategy Code

    a. None

7. Tools and Equipment

    a. None

8. Commitments

    a. None

9. Prerequisites

    a. None

10. Applicable Operator Fundamental Knowledge Check Question(s)

    a. None

B. Read Before Every JPM Performance

1. For Plant JPM's:

    a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary.

    b. With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

2. For Simulator JPM's:

    a. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will provide cues as necessary.

C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The Unit Supervisor has determined that a verifier is not available and that additional verification will not be provided.

INITIAL CONDITIONS	Given: • The plant is starting up. • Both Offgas Train A and B were operating when a spurious condition caused the Offgas Condensers to isolate. • Reactor power has been reduced to 30%. • The cause of the trip has been determined and fixed. • The crew is executing the ARPs associated with this event. • You are the operator in the field supporting the actions of the loss of Offgas system. • The US has determined that the Off Gas System High Radiation Trips do not need to be defeated. • Recombiner A temperature elements are all indicating 390°F. **Evaluator:** *Ask Operator if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Operator Name)**, *recover Offgas Train 'A' after an automatic shutdown per N2-OP-42, Section H.1.0, starting at H.1.4.*

START TIME	

	PERFORMANCE	**ACT. CODE** P / S / NA	**EVALUATOR**
1.	Provides repeat back of initiating cue **Cue:** *Acknowledge repeat back providing correction if necessary*	P	SAT / UNSAT ***STD:*** *Proper communications used.*
Procedure Note:	All actions in Step H.1.4 are performed at 2OFG-IPNL122 unless otherwise noted.		
Procedure Note:	If OFG Recombiner temperatures are less than 290°F, recovery of the OFG System will not be possible in a timely manner and the system shall be considered non-recoverable.		

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
2.	**WHEN** cause of OFG System automatic shutdown has been identified **AND** corrected, **THEN RECOVER** system as follows: **VERIFY** 2OFG-RBNR1A (B) are ready to resume H2/O2 recombination by observing temperature greater than or equal to 290°F on TEMPERATURE TI-30A (B) RECOMBINER 1A (B) using selector switch TEMPERATURE RBNR-1A (B), TE31A (B), TE32A (B), TE33A (B) ***Cue:*** *Temperatures indicates 390°F*	S (H.1.4.a)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, rotates the RECOMBINER 1A TEMPERATURE selector switch to TI-30A and observes that Recombiner Temperature is 390°F*
3.	VERIFY H2 concentration downstream of Recombiner Train is less than 4% by observing OFF GAS SYSTEM 1A (B) HYD SYS 1A (B) % LFL is reading less than 4% ***Cue:*** *Concentration indicates 1%*	S (H.1.4.b)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, observes H2 concentration on OFF GAS SYSTEM 1A, HYD SYS 1A % LFL (Electronic Yokagowa Recorder). Determines H2 concentration is <4%*
4.	**IF** required, **THEN DEFEAT** Offgas System High Radiation Trip **PER** Attachment 5, Removal and Restoration of Off Gas System High Radiation Monitor Trips **[EOP]**, **OTHERWISE MARK** this step N/A	S (H.1.4.c)	**SAT / UNSAT** ***STD:*** *Determines per the turnover that this step is not required to be performed and marks N/A*
5.	**DEPRESS AND RELEASE** the following Offgas circuits RESET pushbuttons:		
5.a	• **RESET** RE13A&B red pushbutton ***Cue:*** *The indicated pushbutton has been depressed and released*	S (H.1.4.d)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, depresses the RE13A & B RED PUSHBUTTON*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
5.b	• **RESET** SYSTEM A black pushbutton **Cue:** *The component you have identified is in the position you described.*	S (H.1.4.d)	***PASS / FAIL** ***STD:*** *On 2OFG-IPNL122, depresses the RESET A BLACK PUSHBUTTON.* ***Failure =*** *RESET A BLACK PUSHBUTTON NOT depressed*
5.c	• **RESET** SYSTEM B black pushbutton **Cue:** *The component you have identified is in the position you described.*	S (H.1.4.d)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, depresses the RESET B BLACK PUSHBUTTON.*
5.d	• **RESET** VAC PUMP VP-1A black pushbutton **Cue:** *The component you have identified is in the position you described.*	S (H.1.4.d)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, depresses the RESET VAC PUMP VP-1A BLACK PUSHBUTTON*
5.e	• **RESET** VAC PUMP VP-1B black pushbutton **Cue:** *The component you have identified is in the position you described.*	S (H.1.4.d)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, depresses the RESET VAC PUMP VP-1B BLACK PUSHBUTTON*
Procedure Note:			
6.	**OPEN** 2OFG-LV20A (B), CONDENSER 1A (B) LEVEL CONTROLLER ***Cue:*** *The indicated valve is open (red light on, green light off)*	S (H.1.4.e)	***PASS / FAIL** ***STD:*** *On 2OFG-IPNL122, adjusts the 2OFG-LV20A control switch to OPEN 2OFG-LV20A. Observes that LV20A fully opens.* ***Failure =*** *2OFG-LV20A control switch NOT manipulated to OPEN 2OFG-LV20A*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
7.	**PLACE** 2OFG-AOV1A (B), PREHTR E1A (B) INLET ISOL, control switch to STARTUP ***Cue:*** *The indicated control switch is in STARTUP. The indicated valve is open (red light on, green light off)*	S (H.1.4.f)	**SAT / UNSAT** ***STD:*** *Determines on 2OFG-IPNL122, the 2OFG-AOV1A PREHTR E1A INLET ISOL CONTROL SWITCH is in STARTUP. Determines AOV1A is open by observing on the status panel*
8.	**VERIFY** 2OFG-AOV103, OFFGAS EXHAUST TO MAIN STACK open ***Cue:*** *The indicated valve is open (red light on, green light off)*	S (H.1.4.g)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, determines AOV103 is OPEN by observing the position on the status panel.*
Procedure Note:	When both 2OFG-AOV1A and AOV1B are open, 2CCS-MOV45A and MOV45B will each automatically open to mid position. If only 2OFG-AOV1A (B) is open, the associated 2CCS-MOV45A (B) will automatically open fully. The valves are located in the OFG Bldg El 261' Hallway.		
Evaluator Note:	It is not necessary for the operator to locally verify the position of 2CCS-MOV45A. When the operator attempts to go to locally verify the valve is in the proper position, inform the operator that 2CCS-MOV45A is in the proper position.		
9.	Locally, **VERIFY** 2CCS-MOV45A (B), OFFGAS CONDENSER 1A (B) OUTLET ISOLATION, is in proper position ***Cue:*** *When the operator determines 2CCS-MOV45A should be open, cue the valve is in the proper position.*	S (H.1.4.h)	**SAT / UNSAT** ***STD:*** *Per the provided cue, determines 2CCS-MOV45A is in the proper position.*
10.	**VERIFY** one Dryer AND Chiller is in service with its associated valve control switch in OPEN (2OFG-AOV4A/5A, 4B/5B, OR 4C/5C) ***Cue:*** *2OFG-AOV4C/5C are already open (red light on, green light off)*	S (H.1.4.i)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, determines one dryer is in service by observing the CONTROL SWITCH for 2OFG-AOV4C/5C is in OPEN.*

	PERFORMANCE	ACT. CODE P / S / NA	EVALUATOR
11.	IF required, **THEN START** 2OFG-P1A **AND** 2OFG-P1B, VACUUM PUMP VP-1A (B), **OTHERWISE MARK** this step N/A ***Cue:*** *2OFG-P1A is already running (red light on, green light off)*	S (H.1.4.j)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, determines 2OFG-P1A is already running by observing the status panel*
12.	**WHEN** recombination is occurring, as indicated by OFG Recombiner temperature rising, **THEN PLACE** 2OFG-LV20A (B) control switch in AUTO **Cue:** *Recombiner temperatures are 400°F and rising.* **Cue:** *The component you have identified is in the position you described.*	S (H.1.4.k)	***PASS / FAIL** ***STD:*** *On 2OFG-IPNL122, determines OFG RECOMBINER TEMPERATURES are rising by observing temperatures. Places the 2OFG-LV20A control switch to AUTO.* ***Failure =*** *2OFG-LV20A control switch NOT placed in AUTO.*
13.	**PLACE** 2OFG-AOV11A (B), CONDENSER 1A (B) OUTLET ISOLATION, control switch to STARTUP ***Cue:*** *The Control Switch for 2OFG-AOV11A is already in STARTUP*	S (H.1.4.l)	**SAT / UNSAT** ***STD:*** *On 2OFG-IPNL122, observes 2OFG-AOV11A control switch is already in STARTUP.*
Evaluator Note:	Once 2OFG-AOV11A is observed to be in STARTUP, provide the following cue: **Cue:** Your task is complete, another operator will complete remaining actions.		

TASK STANDARD	*After an automatic shutdown of Offgas, Offgas Train A has been recovered.*

STOP TIME	

JPM Handout

INITIAL CONDITIONS	Given: • The plant is starting up. • Both Offgas Train A and B were operating when a spurious condition caused the Offgas Condensers to isolate. • Reactor power has been reduced to 30%. • The cause of the trip has been determined and fixed. • The crew is executing the ARPs associated with this event. • You are the operator in the field supporting the actions of the loss of Offgas system. • The US has determined that the Off Gas System High Radiation Trips do not need to be defeated. • Recombiner A temperature elements are all indicating 390°F.

INITIATING CUE	**(Operator Name)**, *recover Offgas Train 'A' after an automatic shutdown per N2-OP-42, Section H.1.0, starting at H.1.4.*

Plant JPM Handout Reference Packet

JPM Handout

INITIAL CONDITIONS	Given: • The plant is starting up. • Both Offgas Train A and B were operating when a spurious condition caused the Offgas Condensers to isolate. • Reactor power has been reduced to 30%. • The cause of the trip has been determined and fixed. • The crew is executing the ARPs associated with this event. • You are the operator in the field supporting the actions of the loss of Offgas system. • The US has determined that the Off Gas System High Radiation Trips do not need to be defeated. • Recombiner A temperature elements are all indicating 390°F. **Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions*

INITIATING CUE	**(Candidate Name)**, *recover Offgas Train 'A' after an automatic shutdown per N2-OP-42, Section H.1.0, starting at H.1.4.*

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-42

OFFGAS SYSTEM

Revision 02400

This Procedure Requires a 10 CFR 50.59 Review

Safety Related

CONTINUOUS USE

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change	
024	00	PCR-20-01275	Implementation of ECP-18-000435, Modification to Install Offgas Hi Radiation Trip Disconnect Switch, to install a disconnect switch for lead 3001 of TB-6K in 2OFG-IPNL122 to simplify the process of defeating the high radiation trip function. This disconnect switch will be located above the current mounting location of TB-6K on the panel compartment divider plate.
		Reason:	Operations has to lift and land lead 3001 on terminal strip TB-6K in 2OFG-IPNL122 per attachment 5 of N2-OP-42 to defeat the high radiation trip function from 2OFG-CAB13A and 2OFG-CAB13B. The performance of this activity presents greater than acceptable risk due to the potential to cause an automatic isolation of 2OFG-AOV103 Offgas Discharge to Stack Isolation Valve. Should 2OFG-AOV103 isolate, N2-SOP-9 will be entered and power reductions to compensate for loss of vacuum will begin

Changes Implemented in the Procedure

- Attachment 5, Plant Impact #1, deleted since it no longer applies.
- Attachment 5, Step 1.3, delete lifting and taping the lead. Replaced it with opening a knife switch.
- Attachment 5, Step 2.3, delete landing the lead. Replaced it with closing a knife switch.
- Attachment 5, Figure 5-1, deleted figure since it is no longer needed for lifting and landing leads.
- Added developmental reference for ECP-18-000435, Modification to Install Offgas Hi Radiation Trip Disconnect Switch.

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A. REFERENCES AND COMMITMENTS 5
A.1. Developmental References 5
A.2. Performance References 6
B. SYSTEM DESCRIPTION 7
C. OPERATING REQUIREMENTS 9
D. PRECAUTIONS AND LIMITATIONS 10
E. STARTUP 13
E.1. OFG Recombiner Train Heatup 13
E.2. OFG Chiller Unit Startup 14
E.3. OFG Preheater Startup 19
E.4. OFG System Startup 26
F. NORMAL OPERATION 43
F.1. Periodic Operator Checks 43
F.2. Placing the Standby Recombiner Train in Service [SOP] 45
F.3. Placing the Standby (Non-Selected) Dryer in Service 51
F.4. Placing the Standby Vacuum Pump in Service 52
F.5. Placing the Recombiner Train in Standby 54
F.6. Isolating Auxiliary Steam to Preheaters 2OFG-E1A & B 57
F.7. Restoring Auxiliary Steam to Preheaters 2OFG-E1A & B 60
F.8. Isolating Auxiliary Steam to Preheaters 2OFG-E1A & B With 2ASS-SOV142 Unavailable 62
F.9. Restoring Auxiliary Steam to Preheaters 2OFG-E1A & B With 2ASS-SOV142 Unavailable 65
F.10. Placing Recombiner Train B in Standby Following Isolation with Recombiner Train A Remaining in STARTUP 67
F.11. Placing Recombiner Train A in Standby Following Isolation with Recombiner Train B Remaining in STARTUP 70
F.12. Restoring Auxiliary Steam to Preheaters 2OFG-E1A & B with 2ASS-PV125 Available 73
F.13. Filling OFG Chiller Unit Level Tanks 76
F.14. Rotation of Off-Gas Chillers 81
G. SHUTDOWN 83
G.1. Offgas System Shutdown [EOP] 83
G.2. OFG System Hydrogen Analyzer Shutdown 87
H. OFF-NORMAL PROCEDURES 88
H.1. OFG System Recovery After Automatic Shutdown [SOP] [EOP] 88
H.2. Offgas Vacuum Pump High Suction Pressure Trip 92

TABLE OF CONTENTS (Continued)

SECTION TITLE PAGE

H.3. OFG Charcoal Adsorber Fire [SOP] ... 94
H.4. Recovery from OFG Charcoal Adsorber Fire ... 100
H.5. Dryer Compressor Adjustment or Charcoal Adsorber Purge ... 104
H.6. Securing Dryer Compressor Adjustment or Charcoal Adsorber Purge ... 107
H.7. Parallel Operation of OFG Charcoal Adsorbers ... 109
H.8. Providing Preheater Steam from Main Steam When 2ASS-PV125 is Out of Service ... 110
H.9. High Dryer Outlet Dewpoint/Temperature ... 111
H.10. Offgas Pretreatment Radiation Monitor(s) Inoperable ... 112
H.11. Bypassing Offgas Charcoal Adsorber Beds at Power ... 113
H.12. Restoring Offgas Charcoal Adsorber Beds at Power ... 115
H.13. Performing Vacuum Pump Lubrication ... 119
H.14. Draining the Dryer Discharge Line ... 120
H.15. Parallel Vacuum Pump Operation ... 121
H.16. Stroking Recombiner Train A Inlet and Outlet AOVs to Prevent Binding ... 124
H.17. Stroking Recombiner Train B Inlet and Outlet AOVs to Prevent Binding ... 127
H.18. Draining the Dryer Discharge Line for an Out of Service Dryer ... 130
H.19. Raising OFG Preheater Pressure ... 131
H.20. Restoring Preheater Steam from Main Steam When 2ASS-PV125 is Ready for Service 132

Attachment 1, Controller Lineup ... 134
Attachment 2, Charcoal Adsorber Temporary Bypass Line and Pressure Gauges Configuration ... 135
Attachment 3, Valving Inside Hydrogen Monitor Cabinet ... 136
Attachment 4, Defeating/Restoring Annunciator 851316, Off Gas Preheater 1A/1B Stm Hdr Press High/Low ... 137
Attachment 5, Removal and Restoration of Off Gas System High Radiation Monitor Trips **[EOP]** ... 138
Attachment 6, Venting Offgas Flow Transmitters Due to Erratic Indications ... 140

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

- T.S. 3.4.8, RCS Specific Activity
- T.S. 3.7.4, Main Condenser Offgas
- T.S. 5.5.8, Explosive Gas and Storage Tank Radioactivity Monitoring System
- USAR, Section 11.3, Gaseous Waste Management Systems
- USAR, Section 12.2.1.4.3, Offgas System Sources
- USAR, Table 14.2 60, Offgas System
- ODCM, D 3.2.4, Gaseous Radwaste Treatment System
- ODCM, D 3.3.2, Radioactive Gaseous Effluent Monitoring Instrumentation
- TRM 3.3.11, Offgas System Explosive Gas Monitoring
- TRM 3.7.8, Explosive Gas Mixture
- 12177 PID-42A to 42C, Piping and Instrumentation Diagram, Offgas System
- 12177 PID-9A, Piping and Instrumentation Drawing, Condenser Air Removal System
- 12177 PID-25A, Piping and Instrumentation Diagram, Clean Steam Reboiler and Auxiliary Cond.
- File Number 0016.530-078-126, Piping & Instrumentation Diagram Offgas Chiller Skid 2OFG-CHL1A
- File Number 0016.530-078-127, Piping & Instrumentation Diagram Offgas Chiller Skid 2OFG-CHL1B
- File Number 0016.530-078-128, Piping & Instrumentation Diagram Offgas Chiller Skid 2OFG-CHL1C
- CTI Nuclear Dwg 7002013 sheets 1 through 6 (File Number 016.530-078-020 through 025) Offgas System Electric Schematic
- Dwg. 7.510-414-146C (Instrument Rack 2CES-RAK240)
- SIL No. 150 Revision 2, Prevention of Hydrogen Ignitions related to the Offgas System
- SIL No. 150 Revision 2, Supplement 1, Ignition Prevention for Recombiner/Charcoal Adsorber Offgas Systems
- SIL No. 150 Revision 2, Supplement 2, Prevention of Hydrogen Ignitions Related to the Offgas System
- SIL No. 246, Control of Sustained Combustion in Offgas Systems
- SIL No. 497, Hydrogen Ignition in Offgas System
- Instruction Manual, Installation Operation Maintenance Instructions for Gaseous Waste Processing System, N2K18750MISP001 (File Code N20837)
- Simple Design Change SC2-0085-93

A.1 (Continued)

- EDC 2F00788, Bypassing Offgas Charcoal Filters on High Temperature
- EDC 2F00789C, Nitrogen Insertion into the Offgas Charcoal Filters
- ECP-10-000814, Form 7B, input M1, OFG System EPU Instrumentation Changes
- ECP-12-000748, Bailey 771 Replacement to Yokogawa DX364 (2OFG-TR2A (B), 2OFG-MR116)
- CA-1992-001235, Revise N2-OP-42 to include recommendations contained in GE SIL No. 150
- NCTS 503788, Corrective Actions from 1993 Charcoal Fire
- SE 2000-075, MSLRM Trip Removal
- SE 1999-082, Offgas Rad Monitor Delay Pipe Installation
- CA-2005-000869, HWC Isolation as a result of Mode Select Switch being placed in wrong position
- LSK 31-04-B
- Filtrine Manual, Model PCP-5005-62WL-SSD
- PID-14C Piping & Instrumentation Diagram Turbine Building Closed Loop Cooling Water
- PID-16A Piping & Instrumentation Diagram Makeup Water Storage
- ECP-15-000609, NMP2 PPC Upgrade – PPC I/O and System/Network Design
- ECP-18-000435, Modification to Install Offgas Hi Radiation Trip Disconnect Switch

A.2. **Performance References**

- N2-OP-9, Condenser Air Removal
- N2-OP-14, Turbine Building Closed Loop Cooling Water
- N2-OP-101C, Plant Shutdown
- N2-OP-101D, Power Changes
- N2-OP-112, Hydrogen Water Chemistry System
- N2-SOP-21, Turbine Trip
- N2-SOP-29, Loss of Condenser Vacuum
- N2-SOP-101C, Reactor Scram
- N2-SOP-101D, Rapid Power Reduction

B. SYSTEM DESCRIPTION

The function of the Offgas (OFG) System is to reduce the radioactivity of the Main Condenser Steam Jet Air Ejectors (SJAEs) discharge gases to acceptable levels prior to their discharge to the atmosphere. In addition, the OFG System functions to reduce the H_2 concentration in the SJAE discharge stream to lower the possibility of hydrogen detonation within the plant.

During plant operation, non-condensible process gases (Offgas) are continuously removed from the Main Condensers by the SJAEs. The Off-gases, which are composed of hydrogen, oxygen, steam vapor, air in-leakage, and radioactive noble gases, are mixed with driving steam in the SJAE and delivered to the OFG Preheaters. The Preheaters are shell and tube type heat exchangers that heat the Offgas stream using auxiliary steam. Heating the Offgas stream ensures that the steam is superheated prior to entering the OFG Recombiners to prevent wetting of the catalyst, and a subsequent reduction in its activity. This ensures optimum chemical activity within the Recombiner. Once the system is in normal operation, Auxiliary Steam supply to the Preheaters may be secured.

Offgas from the Preheaters goes to the Recombiners where it passes over a catalyst which causes the free hydrogen and oxygen in the Offgas stream to chemically react to form steam. The Offgas mixture then goes to the condensers which condense the majority of the steam present and returns that condensate to the Main Condenser. The condensers are shell and tube type heat exchangers which cool the Offgas stream with Turbine Building Closed Loop Cooling (CCS) water. The Offgas stream passes from the condenser to a 75-second delay pipe to facilitate the decay of short-lived radioactive isotopes. An ap proximate three minute delay pipe in the Offgas Radiation Monitors (2OFG-RE13A/B) sample stream provides sufficient delay time to allow for the decay of most of the short-lived activation products. Then it goes into a common header which is connected to three parallel Dryers.

The Dryers are shell and tube type heat exchangers whose tubes are comprised of two cooling coils. The two coils are condensing coils which lower the Offgas stream temperature to approximately 45°F. The dryers dry the Offgas stream to prevent water “poisoning” of the activated charcoal beds in the Charcoal Adsorbers. The normal Chiller lineup will consist of two units in service and the third unit out of service. One Off-Gas dryer will be in service along with the associated Chiller Unit. One Offgas dryer will be out of service with its associated Chiller in service. This will minimize the time required to shift offgas dryer lineup.

Under normal flow conditions (about 6 to 30 scfm) a single dryer will adequately maintain its outlet temp as indicated by 2OFG-T161A (B, C) less than 50°F. Under high flow conditions it is expected that a single dryer can handle flows of about 50 scfm. Flows greater than 50 scfm may result in dryer outlet temperature degradation. In this case, additional dryers may be put in service to split the flow and load on the dryers to maintain outlet temperature ≤ 50°F.

B (Continued)

The Dryers are provided with downstream dew point instrumentation (2OFG-ME116). It has been determined that this element is susceptible to becoming wetted during system startup which artificially increases its reading which resulted in nuisance annunciation. The system has been modified to use Dryer Outlet Temperature, 2OFG-TI61A (B, C), as indication of dew point (reference PCR #PC2-0241-96).

The dry off gas stream from the dryers enters the two Charcoal Adsorber trains. Each train is comprised of four adsorber columns. The trains are operated in series, parallel or lined up for single train operation dependent upon the Offgas flow rate. The activated charcoal delays the radioactive isotopes in the Offgas stream thereby reducing their radioactivity to acceptable levels.

From the Charcoal Adsorbers, the Offgas stream enters one of two HEPA filters. The two filters are arranged in parallel with one in operation while the other is in standby. The filters remove particulate matter from the Offgas stream, such as solid by-products of radioactive decay. Finally, suction is taken by the vacuum pumps which provide the motive force to the Offgas stream. The two vacuum pumps are arranged in parallel with one operating and the other in standby during normal operations. The operating vacuum pump discharges the effluent to the atmosphere via the Main Stack.

C. OPERATING REQUIREMENTS

C.1. The following systems must be in operation to support the OFG System:

- N2-OP-9, Condenser Air Removal
- N2-OP-14, Turbine Closed Loop Cooling System
- N2-OP-16, Makeup Water Storage System
- N2-OP-19, Instrument and Service Air System
- N2-OP-25, Auxiliary Steam, Auxiliary Condensate and Gland Seal
- N2-OP-71C, 600V AC Power Distribution
- N2-OP-73A, Normal DC Distribution
- N2-OP-79, Radiation Monitoring
- N2-OP-91A, Process Computer

C.2. OFG Flow Rate Measuring Devices

- The primary indicator for measuring OFG System flow rate as referred to in ODCM D 3.3.2 is 2OFG-FI120.
- The OFG System Sampler flow rate measuring device is 2OFG-FT1113A or 2OFG-FT1113B. These devices measure the flow through the OFG System Noble Gas Activity Monitors (2OFG-CAB13A and CAB13B) and provide a local and remote readout.

C.3. The Offgas System Explosive Gas Monitoring system consists of a hydrogen monitor for Recombiner Train A (2OFG –AT16A) and a hydrogen monitor for Recombiner Train B (2OFG-AT16B). If either hydrogen monitor is inoperable with the associated Recombiner Train in service, TRM 3.3.11 requires alternate hydrogen monitoring on an increased frequency. This is due to the degraded ability to monitor hydrogen concentrations throughout the entire Offgas system during dual train operation and lack of Control Room annunciation.

In single train operation, the more frequent readings provided by TRM 3.3.11, Required Actions A.1 and A.2, provide direct indication of hydrogen concentration throughout the Offgas system.

- In dual train operation, TRM 3.3.11 Required Actions B.1 and B.2 include additional actions because direct readings may not be representative of the actual hydrogen concentration throughout the Offgas system. In the event one channel is not operable, the hydrogen concentration for that train can be calculated by using the operable channel reading, system flows and the common hydrogen concentration. In the event that both required channels are inoperable, the hydrogen concentration can be determined to be ≤4% by calculating the maximum hydrogen concentration possible using the system flows and common hydrogen concentration.

D. PRECAUTIONS AND LIMITATIONS

D.1. To prevent cross connecting the Auxiliary Steam System (ASS) with the Auxiliary Boiler System (ABM), 2ASS-SOV138, AUX BOILER TO OFFGAS STEAM SUPPL ISOL, and 2ASS-SOV142, OFFGAS STM SUPPLY ISOL, shall not be open at the same time.

D.2. 2ABM-V89, STEAM ISOL TO AUX STEAM HDR, must be closed when Main Steam is supplying the OFG Preheaters. This will prevent contaminated steam from entering the Aux Boiler System due to leakby of 2ASS-SOV138, AUX BOILER TO OFFGAS STEAM SUPPL ISOL.

D.3. 2ASS-HCV150, PV125 BYPASS, and 2ASS-V37, PV125 ISOL, shall be closed when the Aux Boilers are supplying the OFG Preheaters. This will prevent cross connecting the ASS System with the ABM System through 2ASS-V290, bypass valve around 2ASS-SOV142.

D.4. No open flame, welding or cutting should be performed in the OFG Recombiner Rooms or on Charcoal Adsorber Tanks. These areas are potentially explosive due to possible detonable or flammable hydrogen concentrations or contain extremely flammable material. The need for purging of the applicable system sections should be evaluated prior to performing any maintenance activity.

D.5. It will require approximately three to four days to preheat the OFG inlet piping and the OFG Recombiners to 290°F. Subsection E.1 should be performed at least three days prior to planned startup of the OFG System.

D.6. It will require approximately 30 minutes for the Dryer Chiller Units to cool down to their normal operating range. Subsection E.2 must be performed at least 30 minutes prior to planned startup of the OFG System.

D.7. OFG Hydrogen Analyzers require at least 2 hours to stabilize following startup. OFG System flow to the H_2 Analyzers is controlled by the H_2 Analyzer's educator and is required for proper H_2 Analyzer operation. Chemistry Department shall perform required hydrogen analysis until H_2 Analyzers are operational.

D.8. Due to SDC SC2-0085-93 installation, an operator needs to check the operating H_2 Analyzers at least once every shift. This is done to ensure proper sample flow through the H_2 sample rotameter to prevent analyzer cell damage from high temperature if sample flow is lost. If sample flow is lost and **cannot** be resumed within two hours, the H_2 Analyzer(s) must be shut down in accordance with the applicable actions in Subsection G.2. Refer to TRM 3.3.11 for additional monitoring requirements.

D.9. Plant operation must not continue for greater than one week without an operable H_2 Analyzer in service.

D.10. Due to the non-linear characteristic of H_2 Analyzers, the normal operating reading of H_2 indication may be approximately 0.3 to 0.7%. This is conservative compared to the grab sample result of zero from Chemistry Department. There will be a permanent modification on this non-linear characteristic problem.

D.11. When the H_2 Analyzers are secured during a system outage, Subsection G.2 must be performed within 2 hours of closing 2OFG-AOV1A and 2OFG-AOV1B, PREHTR E1A and E1B INLET ISOL. This is done to prevent H_2 Analyzer cell damage due to overheating.

D.12. On initial system startup, the Charcoal Adsorbers should be bypassed to prevent wetting of the charcoal beds. In addition, the sample line delay pipe should be bypassed so that the Offgas Radiation Monitors (2OFG-RE13A/B) Alert/Trip setpoints remain consistent with the process effluent stream.

D.13. No Recombiner Train isolation valve shall be operated if H_2 concentration is greater than or equal to 4% due to the potential for hydrogen ignition and detonation.

D.14. Loss of the OFG System during power operation will result in a loss of condenser vacuum. Depending on initial plant conditions, this will result in any or all of the following:

- Turbine Trip (and possible Reactor Scram) at 22.1” Hg Vacuum
- MSIV Isolation at 8.5” Hg Vacuum
- Bypass Valve Closure at 7” Hg Vacuum

D.15. The amount of time available to restore the OFG System to service is dependent upon several factors: 1) Reactor power level, 2) Amount of air in-leakage, 3) CWS water temperature, etc. Following a loss of the OFG System, it is recommended that Reactor power be reduced as rapidly as possible in accordance with N2-SOP-101D, Rapid Power Reduction.

D.16. If an Offgas leak has been detected, additional monitoring for H_2 leakage should be performed. When monitoring for H_2 leakage, ensure that the H_2 detector is not a possible ignition source.

D.17. If Computer Point 2OFGFA01 alarms (2OFG-FT120), then Chemistry must be notified to sample Offgas to determine the effect on release rate.

D.18. Prior to placing the Offgas System in service, radiation monitors 2OFG-RE13A, 2OFG-RE13B and Stack Gems (2RMS-CAB170) shall be operable and in service or compensatory actions are being taken.

D.19. Under normal flow conditions (about 6-30 scfm) a single dryer will adequately maintain its outlet temp as indicated by 2OFG-TI61A (B, C) less than 50°F. Under high flow conditions it is expected that a single dryer can handle flows of about 50 scfm. Flows greater than 50 scfm may result in dryer outlet temperature degradation. In this case, additional dryers may be put in service to split the flow and load on the dryers to maintain outlet temperature less than or equal to 50°F.

D.20. The Dryers are provided with downstream dewpoint instrumentation (2OFG-ME116). It has been determined that this element is susceptible to becoming wetted during system startup which artificially increases its reading which resulted in nuisance annunciation. The system has been modified to use Dryer Outlet Temperature, 2OFG-TI61A (B, C), as alarm/indication of dewpoint (reference PCR #PC2-0241-96).

D.21. When the control switches for Offgas Recombiner train inlet and outlet valves (2OFG-AOV1A, 2OFG-AOV1B, 2OFG-AOV11A, and 2OFG-AOV11B) are in “STARTUP”, the automatic isolation signal for low preheater outlet temperature is bypassed.

D.22. When 2OFG-RE13A/13B sample hold up pipe(s) are desired to be bypassed with Reactor at power take actions as described in Section H.10.

D.23. The HWC System will trip from an Offgas Isolation on any automatic or manual closure of the Recombiner inlet valve for the Train selected by the HWC/Offgas System Selector Switch.

D.24. The Charcoal Adsorbers must be in service prior to placing the Offgas Radiation Monitor Delay pipe in service.

D.25. 2CCS-MOV45A/B, OFFGAS COND 1A/1B OUTLET ISOLATION, with both control switches in green flagged position will automatically operate as follows:

- One Recombiner train placed in service – the in-service valve opens fully (about 23% open position)
- Both Recombiner trains in service – both valves open to mid position (about 13% open position)
- Placing a standby Recombiner train in service – the on-line valve travels closed from full open to mid position prior to the standby valve opening to mid position
- One Recombiner train trips or is placed in standby – the tripped/standby train’s valve closes prior to the in-service train’s valve opening fully.

D.26. When the Auxiliary Steam supply to the Preheaters is isolated during normal operations, the following conditions would require restoring Auxiliary Steam to the Preheaters:

- Changes in system configurations, such as dual train to single or swapping SJAE trains, would require restoration before performing manipulations.
- During normal plant shutdown, Auxiliary Steam supply would need to be reestablished prior to reaching a Recombiner differential temperature of less than 100°F as indicated between 2OFG-TI15A/B and 2OFG-TI30A/B at 2OFG-IPNL122.

D.27. The setpoints for 2OFG-RE13A (B) are based on a maximum flow rate of 80 SCFM. If Offgas flow exceeds 80 SCFM, the Offgas Radiation Monitors must be declared nonfunctional until the setpoints are lowered for the higher flow rates. With higher Offgas flow rates, setpoints must be lowered so that an isolation will occur before ODCM limits are exceeded.

Initials

H. OFF-NORMAL PROCEDURES

H.1. OFG System Recovery After Automatic Shutdown [SOP] [EOP]

NOTE

- Loss of the OFG System will result in a rapid loss of condenser vacuum. Depending upon plant operating conditions, this may result in a Turbine Trip, MSIV Closure and Reactor Scram. The time available to correct the situation is dependent on several variables including Reactor power, condenser air in-leakage and CWS temperature.
- Performance of this subsection may be required by the EOPs. Changes to this subsection (including renumbering) are required to be reviewed by the EOP coordinator.

H.1.1. **DISPATCH** an operator to 2OFG-IPNL122 to determine cause of loss of OFG System. OPE

H.1.2. **COMMENCE** reducing Reactor power per N2-SOP-101D, Rapid Power Reduction. OPE

H.1.3. **ENTER** N2-SOP-09, Loss of Condenser Vacuum, **AND EXECUTE** concurrently with this procedure. OPE

NOTE

All actions in Step H.1.4 are performed at 2OFG-IPNL122 unless otherwise noted.

H.1.4. **WHEN** cause of OFG System automatic shutdown has been identified **AND** corrected,
THEN RECOVER system as follows:

NOTE

If OFG Recombiner temperatures are less than 290°F, recovery of the OFG System will not be possible in a timely manner and the system shall be considered non-recoverable.

a. **VERIFY** 2OFG-RBNR1A (B) are ready to resume H2/O2 recombination by observing temperature greater than or equal to 290°F on TEMPERATURE TI-30A (B) RECOMBINER 1A (B) using selector switch TEMPERATURE RBNR-1A (B), TE31A (B), TE32A (B), TE33A (B). ..... ☐

Initials

H.1.4 (Continued)

CAUTION

If H_2 concentrations downstream of the operating Recombiner Train are greater than 4%, no valves are to be operated until H_2 concentrations are less than 4% due to the potential for hydrogen ignition and detonation. This may be overridden in an emergency by SM direction.

b. **VERIFY** H_2 concentration downstream of Recombiner Train is less than 4% by observing OFF GAS SYSTEM 1A (B), HYD SYS 1A (B) % LFL is reading less than 4%. ..... ☐

c. **IF** required,
THEN DEFEAT Offgas System High Radiation Trip **PER** Attachment 5, Removal and Restoration of Off Gas System High Radiation Monitor Trips **[EOP]**,
OTHERWISE MARK this step N/A. _____

d. **DEPRESS AND RELEASE** the following Offgas circuits RESET pushbuttons:

- **RESET** RE13A&B red pushbutton ..... ☐
- **RESET** SYSTEM A black pushbutton ..... ☐
- **RESET** SYSTEM B black pushbutton ..... ☐
- **RESET** VAC PUMP VP-1A black pushbutton ..... ☐
- **RESET** VAC PUMP VP-1B black pushbutton ..... ☐ _____

NOTE

The next step permits process flow to recycle back to the Main Condenser.

e. **OPEN** 2OFG-LV20A (B), CONDENSER 1A (B) LEVEL CONTROLLER. ..... ☐

f. **PLACE** 2OFG-AOV1A (B), PREHTR E1A (B) INLET ISOL, control switch to STARTUP. ..... ☐

g. **VERIFY** 2OFG-AOV103, OFFGAS EXHAUST TO MAIN STACK open. ..... ☐

H.1.4 (Continued)

NOTE

When both 2OFG-AOV1A and AOV1B are open, 2CCS-MOV45A and MOV45B will each automatically open to mid position. If only 2OFG-AOV1A (B) is open, the associated 2CCS-MOV45A (B) will automatically open fully. The valves are located in the OFG Bldg El 261' Hallway.

h. Locally, **VERIFY** 2CCS-MOV45A (B), OFFGAS CONDENSER 1A (B) OUTLET ISOLATION, is in proper position..... ☐

i. **VERIFY** one Dryer **AND** Chiller is in service with its associated valve control switch in OPEN (2OFG-AOV4A/5A, 4B/5B, **OR** 4C/5C). ..... ☐

j. **IF** required,
THEN START 2OFG-P1A **AND** 2OFG-P1B, VACUUM PUMP VP-1A (B),
OTHERWISE MARK this step N/A. ..... _____

k. **WHEN** recombination is occurring, as indicated by OFG Recombiner temperature rising,
THEN PLACE 2OFG-LV20A (B) control switch in AUTO. ..... ☐

l. **PLACE** 2OFG-AOV11A (B), CONDENSER 1A (B) OUTLET ISOLATION, control switch to STARTUP. ..... ☐

m. **WHEN** OFG System stabilizes,
THEN PERFORM the following:

1) **PUSH** RESET SYSTEM A (B) pushbutton..... ☐

2) **RETURN** Recombiner Train Isolation AOV control switches to AUTO. ..... ☐ _____

n. **IF** required,
THEN PERFORM Steps H.1.4.d through H.1.4.l to return second OFG Recombiner Train to service,
OTHERWISE MARK this step N/A. _____

o. **IF** Offgas System High Radiation Trip was defeated,
THEN RESTORE Offgas System High Radiation Trip **PER** Attachment 5, Removal and Restoration of Off Gas System High Radiation Monitor Trips **[EOP]**,
OTHERWISE MARK this step N/A. _____

Initials

H.1.5. **IF** OFG System can **NOT** be returned to service in a timely manner,
THEN CONTINUE to lower Reactor power **PER** N2-SOP-101D, Rapid Power Reduction,
OTHERWISE MARK this step N/A. _____

H.1.6. **IF** a Turbine trip occurs,
THEN refer to N2-SOP-21, Turbine Trip,
OTHERWISE MARK this step N/A. _____

H.1.7. **IF** a Scram occurs,
THEN REFER to N2-SOP-101C, Reactor Scram,
OTHERWISE MARK this step N/A. _____

Initials

H.2. **Offgas Vacuum Pump High Suction Pressure Trip**

NOTE

- All actions in this Subsection are performed at 2OFG-IPNL122 unless otherwise noted.
- When a high suction pressure occurs, the operating vacuum pump trips and the standby pump auto starts. The high pressure signal to the tripped pump seals in and must be reset in order to place the pump in standby or in service.

H.2.1. **VERIFY** cause of high suction pressure trip has been determined **AND** corrected. _____

H.2.2. **IF** tripped pump is to be placed in standby,
THEN PERFORM the following,
OTHERWISE MARK this step N/A:

a. **POSITION** VACUUM PUMP SELECT TRAIN 1A/1B switch to in-service vacuum pump VP-1A **OR** VP-1B..... ☐

b. For tripped pump, **PLACE** VACUUM PUMP VP-1A (B) control switch in AUTO. ..... ☐

c. For tripped pump, **DEPRESS AND RELEASE** RESET VAC PUMP VP-1A (B) pushbutton. ..... ☐ _____

H.2.3. **IF** tripped pump is to be placed in service,
THEN PERFORM the following,
OTHERWISE MARK this step N/A:

a. For tripped pump, **DEPRESS AND RELEASE** RESET VAC PUMP VP-1A (B) pushbutton. ..... ☐

b. **VERIFY** tripped pump restarts as follows:

- 2OFG-P1A (B), VACUUM PUMP VP-1A (B), is running. ..... ☐
- 2OFG-AOV45A (B), VACUUM P1A (B) INLET ISOL, is open. ☐
- 2OFG-AOV52A (B), VACUUM P1A (B) OUTLET ISOL, is open. ..... ☐ _____

c. For pump to be secured, **PLACE** 2OFG-AOV45A (B) /52A (B) control switch to CLOSE. ..... ☐

d. **WHEN** 2OFG-AOV45A (B)/52A (B) are closed,
THEN RETURN control switch to AUTO. ..... ☐

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-1
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 90% power. LPCI pump 2RHS*P1C is out of service for maintenance.		
Turnover:	Start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0. The procedure is in progress up to step F.3.2. Secure CCS pump 2CCS-P1A per N2-OP-14 section F.4.0. Then, raise power to 95% using Recirculation flow per the provided ReMA.		
Critical Tasks:	CT-1: Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5. RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig. CT-2: Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC. Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap CCS Pumps N2-OP-14
2	N/A	R – ATC, SRO	Raise Reactor Power with Recirculation Flow N2-OP-101D
3	RR46A	C – ATC, SRO TS – SRO	Recirculation Pump High Vibration ARP 602219, N2-SOP-29.1, N2-SOP-29, Technical Specifications
4	Overrides	TS – SRO	RCIC Steam Line Isolation Valve 2ICS*MOV128 Breaker Trips ARP 601305, ARP 601319, Technical Specifications
5	Remote CW27	I – BOP, SRO MC – BOP	2CCS-TIK104 Automatic Setpoint Failure N2-OP-14, N2-SOP-14
6	OV0953 OV0954	I – ATC, SRO MC – ATC	Feedwater Level Control Drifts High in Automatic N2-SOP-06
7	MS04 PC10A	M – All	Steam Leak in Drywell, Pair of Drywell/Wetwell Vacuum Breakers Fail Open N2-SOP-101C, N2-EOP-RPV, N2-EOP-PC
8	RP03	I – ATC, SRO MC – ATC	RPS Fails to Scram the Reactor N2-SOP-101C, N2-EOP-C5
9	FW:BA (B)(C): MT:I	C – All MC – BOP	Condensate Booster Pumps Trip N2-SOP-101C, N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-1**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 8 & 9**	2	
2. Abnormal events (2-4) **Events 3, 5, 6**	3	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV, EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)	0	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 1** Revision: **0.0**

Title: **November 2023 NRC Scenario 1**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/11/23
	N/A / Dan Delaney	7/11/23
	N/A Mike Carno	7/11/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-14, Turbine Building Closed Loop Cooling System
2. N2-OP-101D, Power Changes
3. N2-SOP-29.1, Recirc Pump Failures
4. N2-SOP-29, Sudden Reduction in Core Flow
5. N2-SOP-14, Loss or Degraded CCS System
6. N2-SOP-06, Feedwater Failures
7. N2-SOP-101C, Reactor Scram
8. N2-EOP-RPV (1-3), RPV Control - Flowchart
9. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
10. N2-EOP-6.1, Low RPV Water Level/High Drywell Pressure Actuations
11. N2-EOP-6.22, Containment Sprays
12. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
13. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
14. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
15. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 90% power with LPCI pump 2RHS*P1C out of service.

   b. Event 1 is the normal evolution performed by the BOP operator to swap CCS pumps per N2-OP-14.

   c. Event 2 is a reactivity evolution. The ATC operator will raise Reactor power using Recirculation flow to approximately 95% per the provided ReMA.

   d. Event 3 occurs when Recirculation pump 1A develops a high vibration condition. The crew will respond per ARP 602219, execute N2-SOP-29.1, trip the pump, and execute N2-SOP-29. The SRO will evaluate Technical Specifications.

   e. Event 4 occurs when the breaker for RCIC steam line isolation valve 2ICS*MOV128 trips. The crew will respond per ARPs and the SRO will evaluate Technical Specifications.

   f. Event 5 occurs when the CCS temperature controller, 2CCS-TIK104, automatic setpoint fails high. This results in a temperature rise in all CCS cooled components. The crew will perform the associated ARP actions for the failure. The crew will take manual control of 2CCS-TIK104 and control CCS outlet temperature between 80-95°F (ARP) or 80-85°F (SOP).

   g. Event 6 occurs when a Feedwater level control (FWLC) malfunction causes Reactor water level to drift high. The crew will evaluate Feedwater indications, enter N2-SOP-06, determine that FWLC is not responding correctly, and take manual control of the master controller. The crew will then stabilize and restore Reactor water level to the normal band.

h. Events 7, 8, & 9 begin when a steam leak develops in the Drywell. RPS will fail to scram the Reactor (automatically and manually). The crew will insert control rods by manually initiating RRCS (*critical task*). The crew will respond after the scram by executing N2-SOP-101C and N2-EOP-RPV (1-3). Condensate Booster pumps will trip. This will complicate Reactor water level control. A pair of failed open Drywell/Wetwell vacuum breakers will cause elevated Containment bypass leakage will amplify the Containment pressure response to the steam leak. The Drywell pressure rise will require entry into N2-EOP-PC (1-3). The crew will spray the Suppression Chamber and the Drywell (*critical task).*

2. Termination Criteria

   a. All control rods are inserted.

   b. Reactor level is being controlled in assigned band.

   c. Containment pressure is being controlled per N2-EOP-PC (1-3).

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because with Drywell pressure above 1.68 psig, a Reactor scram is required. Continued power generation further challenges the Primary Containment. With RPS failing to scram the Reactor, RRCS provides alternate means to insert control rods and shutdown the Reactor.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Primary Containment pressure is elevated and RPS has failed to scram the Reactor. N2-EOP-PC (1-3) and N2-EOP-RPV (1-3) provide direction to scram the Reactor.*
Success Path:	*Manually arm and depress RRCS pushbuttons*
Performance Feedback:	*Control rod position and Reactor power indications will provide performance feedback regarding the success of RRCS.*
Measurable Performance Standard:	*Arming and depressing of the RRCS pushbuttons will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

   a. CT-1.0, Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually

initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5 (1-3). RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because initiating Drywell spray reduces Primary Containment pressure. This reduces stresses on the Drywell and Suppression Chamber, assists in avoiding "chugging" that may cause fatigue failure of the LOCA downcomers, and avoids the need for a blowdown. These benefits reduce challenges to the fuel cladding, the RPV, and the Primary Containment.*
Initiating Cue:	*Multiple Primary Containment pressure and temperature indications and annunciators will indicate degrading conditions. N2-EOP-PC (1-3) provides direction to initiate Drywell spray.*
Success Path:	*Drywell spray is initiated using N2-EOP-6.22.*
Performance Feedback:	*Drywell spray flow and lowering Primary Containment pressure and temperature indications will provide performance feedback regarding success of crew actions to initiate Drywell spray.*
Measurable Performance Standard:	*Manipulation of RHR control switches will provide observable actions for the evaluation team.*
Bounding Conditions:	*Based on N2-EOP-PC (1-3) requirements*

b. CT-2.0, Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC (1-3). Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

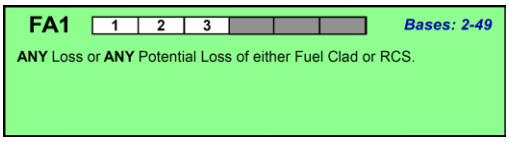
   a. DMS-PC-03, high containment pressure; drywell sprays required; maintain safe region PSP; no blowdown required

6. Technical Specifications (Applicable actions for initial conditions only)

   a. TS 3.5.1 Condition A for LPCI pump 2RHS*P1C (Day 1 of 7)

7. EAL Classification

   a. Alert, EAL FA1:

1) 

B. Initial Conditions

1. IC Number

   a. IC-021 or equivalent (IC-200 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**RH:PC:BK:I**, RHR Pump 1C Breaker Failure, Final = trip	**Inserted**
2)	**RR46A**, RCS PMP1A VIBRATION (MILS), Initial = 3, Final = 25, Ramp = 5 minutes	**TRG1**
3)	**MS04**, Steam Line Rupture Inside Primary Containment, Final = 1.5, Ramp = 15:00	**TRG5**
4)	**PC10A**, Drywell/Wetwell Vacuum Bkr Pair Failed Open, Final = true	**TRG6**
5)	**RP03**, Reactor Protection System Failure To Scram, Final = true	**Inserted**
6)	**FW:BA:MT:I**, Booster Pmp 2A Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
7)	**FW:BB:MT:I**, Booster Pmp 2B Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
8)	**FW:BC:MT:I**, Booster Pmp 2C Motor Failure, Final = short, Delay = 2 minutes	**TRG6**
9)	**RH08**, Group 5 Isolation Failure (RHS*MOV122/115), Final = true	**TRG7**
10)	**OV0953**, FCS-MASTER Setpoint Increment Pushbutton, Final = pressed	**TRG4**
11)	**OV0954**, FCS-MASTER Setpoint Decrement Pushbutton, Final = disabled	**Inserted**

b. Remotes

1) **CW27**, 2CCS-TIK104 Auto Setpoint, Final = 110, Ramp = 2 minutes — **TRG3**

c. Overrides

1) **DO-806**, OFF RHR PUMP P1C GREEN, Final = off — **Inserted**

2) **DO-1023**, OFF MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) GREEN, Final = off — **TRG2**

3) **DO-1024**, ON MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) RED, Final = off — **TRG2**

4) **DI-901**, CLOSE MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD), Final = off — **TRG2**

5) **DO-532**, OFF STEAM SUPPLY INBD ISOL MOV 128 INOP AMBER, Final = on — **TRG2**

d. Annunciators

1) **an601305**, rcic system inoperable, Final = crywolf — **TRG2**

2) **an601319**, rcic valves mtr overload, Final = crywolf — **TRG2**

e. Event Triggers

Event #	Event Action	Command
6	rd_fi==1 (All rods full in)	None
30	Iscla106>187 & ov_slim0spincpb==2 (Reactor water level >187" with event 6 active)	dmf ov0953

f. Equipment Out of Service

1) LPCI pump 2RHS*P1C

g. Support Documentation

1) Provide N2-OP-14 sections F.1.0, F.3.0, and F.4.0. Have section F.3.0 completed up to step F.3.2.

2) Provide ReMA for raising Reactor power from 90% to 95% with Recirc flow.

3) Provide N2-OP-101D completed up to step F.3.7.d.

4) Provide N2-OP-29 section F.4.0

h. Miscellaneous

1) Place clearance reference tag on the LPCI pump 2RHS*P1C control switch in P-T-L.

2) Place Protected pathway signs on 2RHS*P1A and 2RHS*P1B control switches.

C. Shift Turnover Information

1. Reactor Power: 90%
2. Rodline: Above 100%
3. Technical Specification LCOs in effect:
   a. TS 3.5.1 Condition A for LPCI pump 2RHS*P1C (Day 1 of 7)
4. Significant Problems / Abnormalities / Equipment Out of Service:
   a. LPCI pump 2RHS*P1C out of service for maintenance
5. Evolutions / Maintenance Scheduled for this Shift:
   a. Swap CCS pumps per N2-OP-14 sections F.3.0 and F.4.0.
   b. Then, raise Reactor power to 95% using Recirculation flow per the provided ReMA.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 90%.
- LPCI pump 2RHS*P1C is out of service for maintenance. TS 3.5.1 Condition A applies (Day 1 of 7).

PART III: Remarks/Planned Evolutions:

- Start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0. The procedure is in progress up to step F.3.2. Secure CCS pump 2CCS-P1A and place it in standby per N2-OP-14 section F.4.0.
- Then, raise Reactor power to 95% using Recirculation flow per the provided ReMA.

Event #1 – Swap CCS Pumps

Event Information	• Presumed or required initial plant operating conditions - Reactor at power - 2CCS-P1A and 2CCS-P1B running, 2CCS-P1C shutdown • Verifiable Actions - Start 2CCS-P1C - Stop 2CCS-P1A • Final (expected) operating result - 2CCS-P1B and 2CCS-P1C running, 2CCS-P1A in standby

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0 and secure CCS pump 2CCS-P1A per N2-OP-14 section F.4.0.
Role Play: If asked about operating indications after 2CCS-P1C start, report that all indications are SAT. **Role Play:** When directed to slowly open 2CCS-V303C, wait 1 minute, then report 2CCS-V303C is open. **Role Play:** When directed to close 2CCS-V303A, wait 1 minute, then report 2CCS-V303A is closed. **Role Play:** If asked about operating indications after 2CCS-P1A stop, report that all indications are SAT. **Role Play:** When 2CCS-P1A is secured and asked to open 2CCS-V303A, wait 1 minute, then report 2CCS-V303A is open.	**BOP** • Acknowledges direction to start 2CCS-P1C and shutdown 2CCS-P1A • Starts 2CCS-P1C by placing control switch to Normal-After-START • Directs EO to slowly open 2CCS-V303C • Confirms normal operating indications per Subsection F.1.0: ◦ Pump discharge pressure 120-135 psig ◦ Pump current <58.5 amps ◦ 2CCS-TIK104 in Auto controlling ~85°F • Confirms 2CCS-P1A is NOT required for CCS System flow • Directs EO to close 2CCS-V303A • Secures 2CCS-P1A by placing control switch in Normal-After-STOP • Confirms normal operating indications per Subsection F.1.0: ◦ Pump discharge pressure 100-125 psig ◦ Pump current <58.5 amps

Instructor Actions / Plant Response	Operator Actions
	◦ 2CCS-TIK104 in Auto controlling ~85°F

Event Termination Criteria	• 2CCS-P1C is running and 2CCS-P1A is in standby

Event #2 – Raise Reactor Power with Recirculation Flow

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~90% power • Verifiable Actions - Raise Recirculation flow • Final (expected) operating result - Reactor power at ~95%

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to raise Reactor power to 95% using Recirc flow per ReMA and N2-OP-101D
Note: T.S. 3.4.1 requires that loop flow mismatch be maintained ≤ 5% of rated core flow (108.5 mlbm/hr) when operating at an effective core flow ≥ 70% of rated core flow. In this case the loop flow mismatch allowable would be 5.425 mlbm/hr.	**ATC** • Acknowledges direction to raise Reactor power to 95% using Recirc flow • Raises Reactor power to 95% by raising core flow as follows: ◦ Moves 2RCS*HYV17A&B individually in the open direction, maintaining loop flow differential at a minimal value by alternating between the two valves • Monitors APRMs and rate of power change
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• Reactor power has been raised sufficiently as determined by the Lead Evaluator

Event #3 – Recirculation Pump High Vibration

Event Information	• Presumed or required initial plant operating conditions - Plant on-line near rated power with Recirc pump 1A running • Verifiable Actions - Trip Recirc pump 1A • Final (expected) operating result - Operating at reduced power in single loop operation on Recirc pump 1B

Instructor Actions / Plant Response	Operator Actions
When directed by the lead evaluator, **insert** the following **malfunction**: **TRG1** **RCS PMP1A VIBRATION (MILS)**, Initial = 3, Final = 25, Ramp = 5 minutes	
The following annunciator alarms: • *602219, RECIRC 1A/1B VIBRATION HIGH* *The following computer point is generated:* • *RCSNC03, RCRC PMP1A VIBRATION HIGH*	**ATC** • Silences, acknowledges and reports annunciator 602219
Note: N2-RESP-07 may be referenced for Tech Specs. **Role Play:** As RE contacted, acknowledge direction and inform the control room you will start working on it. **Spec** / **Condition** / **Applicable Actions**: 3.4.1 / C / C.1 **Action** / **Description**: C.1 / Satisfy the requirements of the LCO (4 hours)	**SRO** • Acknowledges high vibration on Recirc pump 1A • Directs ATC to enter and execute N2-SOP-29.1 • Directs trip of Recirc pump 1A • Directs ATC to enter and execute N2-SOP-29 • Refers to TS 3.4.1 and determines the LCO is not met because RPS flow instrumentation is not reset for single loop operation. Enters a four hour LCO, Condition C. • Notifies RE to check thermal limits and to make adjustments to the rod pattern based on operation in the Heightened Awareness Region

Instructor Actions / Plant Response	Operator Actions
Role Play: As EO dispatched to monitor vibrations, wait 2 minutes, then report that Recirc pump 1A motor shaft vibrations are 21 mils and rising slowly, motor frame vibrations are 6 mils and stable, Recirc pump 1B vibrations are all normal. **Note:** The first 8 CRAM rods are: • 30-39 • 30-23 • 22-31 • 38-31 • 14-39 • 46-23 • 14-23 • 46-39	**ATC** • Executes ARP 602219 • Dispatches EO to monitor vibrations • Coordinates with BOP to monitor motor temps • Determines / reports N2-SOP-29.1 entry is required • Acknowledges direction to enter N2-SOP-29.1 • Determines / reports that Recirc pump 1A must be tripped • Trips Recirc pump 1A • Acknowledges direction to enter N2-SOP-29 • Determines a Recirc pump is in service • Determines core flow and power are NOT within the Scram Region • Determines Core flow AND power are NOT left of Line K • Inserts the first four cram rods • Performs N2-SOP-29, Attachment 1: • If plant is operating in the EXIT region: • Shifts APRM recorders to fast speed and monitors for core oscillations • Inserts more CRAM rods and/or lowers core flow • Determines cause of reduced core flow is NOT FCV runback • Verifies closed flow control valve for RCS loop A • Verifies RCS loop B is operating <41,800 gpm • Contacts I&C for adjustment to APRMs, Rod Blocks, and Rod Block Monitor • Refers to N2-OP-29, Section H.6.0 for single loop operations. • Posts operating in “Heightened Awareness Region”
	BOP • Coordinates with ATC to monitor motor winding and bearing temperatures

Event Termination Criteria	• Recirc pump 1A tripped • 4 CRAM rods inserted • Technical Specifications have been addressed

Event #4 – RCIC Steam Line Isolation Valve 2ICS*MOV128 Breaker Trips

Event Information	• Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Technical Specifications addressed • Final (expected) operating result - Plant on-line

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **overrides**: **TRG2** **an601305**, rcic system inoperable, Final = crywolf **an601319**, rcic valves mtr overload, Final = crywolf **DO-1023**, OFF MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) GREEN, Final = off **DO-1024**, ON MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD) RED, Final = off **DI-901**, CLOSE MOV 128 VALVE STEAM SUPPLY LINE ISLN (INBOARD), Final = off **DO-532**, OFF STEAM SUPPLY INBD ISOL MOV 128 INOP AMBER, Final = on *The plant responds as follows:* • *2ICS*MOV128 indicating lights extinguish*	
The following annunciators alarm: • *601305, RCIC SYSTEM INOPERABLE* • *601319, RCIC VALVES MOTOR OVERLOAD*	**BOP** • Silences, acknowledges and reports annunciators • Reports 2ICS*MOV128 breaker is tripped

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.6.1.3</td><td>A</td><td>A.1 & A.2</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured (4 hours)</td></tr>
<tr><td>A.2</td><td colspan="2">Verify the affected penetration flow path is isolated (31 days)</td></tr>
</table>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.5.3</td><td>A</td><td>A.1 & A.2</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Verify by administrative means HPCS is operable (Immediately)</td></tr>
<tr><td>A.2</td><td colspan="2">Restore RCIC to operable status (14 days)</td></tr>
</table>

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SRO

- Acknowledges reports
- Reviews Tech. Spec. 3.6.1.3 Condition A, required actions A.1 & A.2
- May direct closing 2ICS*MOV121
- May review Tech Spec. 3.5.3, Condition A, required actions A.1 & A.2

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Role Play:

As EO dispatched to investigate 2ICS*MOV128 breaker (2EHS*MCC302 BKR 14A), wait 2 minutes, then report that the breaker is in the tripped condition and there is a slight acrid odor.

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BOP

- Executes ARP 601305:
  - Determines INOP Status Light is energized for 2ICS*MOV128
- Executes ARP 601319:
  - Determine 2ICS*MOV128 is the affected MOV
- Executes INOP STATUS LIGHT ARP (TURB STM SPLY INBD ISOL VLV ICS*MOV128):
  - Observes 2ICS*MOV128 red and green indicating lights extinguished
  - Dispatches EO to 2EHS*MCC302

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</table>

Event Termination Criteria	• Technical Specifications addressed

Event #5 – 2CCS-TIK104 Auto Setpoint Failure

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Taking manual control of 2CCS-TIK104 and lower temperature setting • Final (expected) operating result - Operating

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **remote**: **TRG3** **CW27**, 2CCS-TIK104 Auto Setpoint, Final = 110, Ramp = 2 minutes	
The following plant response occurs after event initiation: • *2CCS-TIK104 setpoint slowly drifts toward minimum cooling* *After approximately 30 seconds the following computer point is generated:* • *GMLTA01, GN LEADS CLR RTN AIR TEMP (HI)* *The following annunciator alarms ~1:30 after event initiation:* • *601244, TURBINE BLDG CLOSED LOOP COOLING SYS TROUBLE* *The following computer points are generated ~1:30 after event initiation:* • *CCSTC04, TBCLCW HX DISCH TEMP (HIGH)* • *CCSTA01, TBCLCW SYS HX DISCH TEMP (HIGH)*	**RO** • Silences, acknowledges and reports annunciator 601244 • Recognizes / reports 2CCS-TIK104 setpoint rising

Instructor Actions / Plant Response	Operator Actions
Note: Crew may opt to use ARP 601244 actions vs. using N2-SOP-14 guidance. The actions are the same with the exception of the allowed temperature band. In N2-SOP-14 the allowable temperature band is 80-85°F whereas the ARP guidance for computer point CCSTC04 allows 80-95°F.	**SRO** • Acknowledges reports • Directs BOP to execute ARP 601244 and/or N2-SOP-14 • May direct BOP to execute N2-SOP-68 to monitor in the event a generator runback occurs
Role Play: If directed as field operator to investigate stator water coolant flow and temperature at the stator water cooling/hydrogen monitor panel, wait 5 minutes and report the following: • If the CCS-TIK104 setpoint has been dialed down in manual and annunciator 851156 has cleared, then report that stator water inlet temperature is ~38°C, stator water outlet temperature is ~56°C and stator water flow is ~760 gpm • If the CCS-TIK104 setpoint has not yet been dialed down in manual and annunciator 851156 is still in alarm, then report that stator water inlet temperature is ~40°C, stator water outlet temperature is ~59°C and stator water flow is ~760 gpm	**BOP** • Performs N2-SOP-14 actions: ◦ Determines that all CCS Pumps have not tripped ◦ Determines that SWP supply to the turbine building is not isolated ◦ Determines CCS system conditions have not degraded to the point that system loads will NOT have adequate cooling ◦ Determines that main turbine high vibration alarm is not in ◦ Determines that a failure of the standby pump to auto start has not occurred ◦ Determines that the CCS expansion tank is not the problem ◦ Determines at least one CCS pump is operating ◦ Determines that cooling capacity is degraded: - Monitors system loads (Detail A, Major System Loads) - Secures loads as necessary ◦ Performs the following to stabilize the CCS system: - Places CCS-TIK104 in manual - Manually controls temperature using CCS-TIK104 between 80-85°F • Provides update to the control room that manual control has been established and that CCS temperature is lowering • Performs ARP 851112 actions, as required:

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched to CCS temperature control valve, wait 2 minutes and then report that there is nothing abnormal at the valve.	◦ Determines the cause of the event to be rising CCS inlet temperature due to the failed CCS temperature controller ◦ Determines that stator water cooling is lowering • Provides crew update for the requirement of ARP 851112 to enter N2-SOP-68 and the N2-SOP-68 entry, as required • Performs N2-SOP-68 actions, as required: ◦ Reviews N2-SOP-68 and determines that no action is required unless a generator runback has occurred ◦ Notifies the CRS of N2-SOP-68 requirements

Event Termination Criteria	• 2CCS-TIK104 in manual with temperature setting in required band

Event #6: Feedwater Master Controller Failure - High

Event Information	• Presumed or required initial plant operating conditions - Plant on-line with FWLC in master auto • Verifiable Actions - Take manual control of feedwater master controller and stabilize/restore feedwater level back in normal band • Final (expected) operating result - Operating with Feedwater Master controller in manual

Instructor Actions / Plant Response	Operator Actions
When directed by lead evaluator **insert** the following **malfunction**: **TRG4** **OV0953**, FCS-MASTER Setpoint Increment Pushbutton, Final = pressed	
The following annunciator alarms ~30 seconds after event initiation: • *603139, REACTOR WATER LEVEL HIGH/LOW* *The following computer points are generated after event initiation:* • *FWSLC01, REACTOR WTR LEVEL HI/LO*	**Crew** • Recognizes / reports the following: ◦ Annunciator 603139 in alarm ◦ RPV water level slowly rises ◦ Steam flow / feed flow mismatch
Note: The Feedwater master auto setpoint decrement pushbutton is failed off. When Reactor water level rises above ~187", **TRG30** will automatically **delete malfunction OV0953**. Reactor water level will stabilize ~191" and the Feedwater master auto setpoint will NOT be able to be lowered. **Note:** Crew may enter N2-SOP-08 and N2-SOP-101D for unplanned power changes depending on the timing of initial event diagnosis.	**SRO** • Provides crew update for entry into N2-SOP-06 • Directs ATC to enter and execute N2-SOP-06 • Directs RPV level band

Instructor Actions / Plant Response	Operator Actions
	ATC • Monitors the following: ◦ Reactor power ◦ Reactor water level ◦ Reactor pressure • Performs N2-SOP-06 actions and determines the following: ◦ FWLC is NOT responding correctly • Takes manual control of FWLC master controller • Adjusts feedwater master controller in manual to restore feedwater level to directed level band

Event Termination Criteria	• FWLC in master manual • Reactor water level in normal band

Constellation

Event #7, 8, and 9 – Steam Leak in Drywell, Pair of Drywell/Wetwell Vacuum Breakers Fail Open; RPS Fails to Scram the Reactor; Condensate Booster Pumps Trip

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Control rods inserted using RRCS - Reactor water level controlled - Suppression Pool and Drywell sprayed • Final (expected) operating result - All control rods inserted - Reactor level is being controlled in assigned band - Containment pressure is being controlled per N2-EOP-PC (1-3) • Mitigation Strategy code (if applicable) - DMS-PC-03, high containment pressure; drywell sprays required; maintain safe region PSP; no blowdown required
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a steam leak inside the Containment and a failure of RPS to scram the Reactor, the crew will manually initiate RRCS, in accordance with N2-SOP-101C and/or N2-EOP-C5 (1-3). RRCS must be initiated within 10 minutes of Drywell pressure exceeding 1.68 psig. 2. Given a steam leak inside the Containment, the crew will spray the Drywell, in accordance with N2-EOP-PC (1-3). Drywell spray must be initiated in time to avoid the need for an RPV Blowdown due to violating the Pressure Suppression Pressure.

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunction**: **TRG5** **MS04**, Steam Line Rupture Inside Primary Containment, Final = 1.5, Ramp = 15:00 *Drywell pressure rises* *Drywell temperature rises*	

Instructor Actions / Plant Response	Operator Actions
The following malfunction is preset to prevent RPS from inserting control rods: ***RP03**, Reactor Protection System Failure To Scram, Final = true* *The following malfunctions will automatically insert when all control rods are full in:* **TRG6** **PC10A**, Drywell/Wetwell Vacuum Bkr Pair Failed Open, Final = true **FW:BA:MT:I**, Booster Pmp 2A Motor Failure, Final = short, Delay = 2 minutes **FW:BB:MT:I**, Booster Pmp 2B Motor Failure, Final = short, Delay = 2 minutes **FW:BC:MT:I**, Booster Pmp 2C Motor Failure, Final = short, Delay = 2 minutes	**Crew** • Recognizes / reports rising Drywell pressure and temperature
Note: N2-SOP-101C has an immediate action for the ATC to manually initiate ARI once RPS fails to depressurize the scram air header and insert rods. Alternately, the SRO may direct this based on guidance in N2-EOP-C5 (1-3). N2-EOP-C5 (1-3) entry is NOT expected, since the guidance in N2-SOP-101C will likely be carried out prior to EOP entry.	**SRO** • Acknowledges reports • May direct ATC to lower power using Recirc flow IAW with N2-SOP-101D • Directs manual Reactor scram • Directs entry into N2-SOP-101C • Acknowledges failure of RPS to insert control rods • **If required, directs manual initiation of RRCS (CT-1.0)** • Enters N2-EOP-RPV (1-3) on low Reactor water level and high Drywell pressure • Enters N2-EOP-PC (1-3) on high Drywell pressure and temperature • Performs N2-EOP-RPV (1-3) actions: ◦ Directs scram procedure entered (N2-SOP-101C) ◦ Performs level control actions: - Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1

Instructor Actions / Plant Response	Operator Actions
	- Determines that RPV water level is known - Determines that RPV water level can be restored and maintained between 159.3 and 202.3 inches using Feedwater, HPCS, RCIC, CRD, and/or SLC - Directs an RPV level band of 160 – 200 inches - May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: - Determines that no SRVs are cycling - Determines that RPV pressure can be stabilized below 1052 psig using the main turbine bypass valves ◦ Directs a pressure band of 800-1000 psig • Performs N2-EOP-PC (1-3) primary containment pressure leg actions: • Determines cannot maintain containment pressure below 1.68 psig • Determines that suppression chamber spray is required before suppression chamber pressure reaches 10 psig • Determines that suppression pool water level is below EL 217 ft. • Determines that suppression chamber sprays are warranted • Directs RO to place RHR in suppression chamber sprays • Directs pneumatics restored to the Drywell • Establishes a benchmark for closing the outboard MSIVs to control cooldown rate • May direct low pressure ECCS pumps terminated and prevented with a high drywell pressure and reactor pressure lowering (N2-EOP-RPV (1-3), step P-1) • Directs Suppression Chamber Spray secured prior to Suppression Chamber pressure lowering to 0 psig • Waits until suppression chamber pressure is above 10 psig

Instructor Actions / Plant Response	Operator Actions
	• Determines that suppression pool water level is below EL 217 ft. • Directs RO to verify the following: - Both recirculation pumps tripped - All drywell unit coolers tripped • **Directs RO to place RHR in drywell sprays in accordance with N2-EOP-6.22 (CT-2.0)**
Booth Operator: If directed to defeat the HPCS Level 8 interlocks by placing HPCS INJ VLV LEVEL 8 BYPASS TEST SWITCH in the TEST position, **manually insert** the following **remote function**: • **CS14**, OPS-CSH01 PNL625 TST.SW CSH*MOV107, Final = Test Wait 1 minute, then report that the HPCS Level 8 interlocks have been defeated. **Booth Operator:** If directed to defeat the RCIC Level 8 interlocks, **manually insert** the following **remote function**: • **RC02A,** RCIC Level 8 Trip Defeat: Withdraw Trip Units N693A, E, Final = defeated Wait 1 minute, then report that the RCIC Level 8 interlocks have been defeated.	**ATC** • Acknowledges SRO direction to scram the reactor • Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603. • Places the Reactor Mode Switch in the SHUTDOWN position • Recognizes / reports failure of RPS to insert rods • **Manually initiates RRCS (CT-1.0)**: ◦ Arms and depresses pushbuttons: ▪ DIVISION I CHANNEL A MANUAL INITIATION ▪ DIVISION I CHANNEL B MANUAL INITIATION ▪ DIVISION II CHANNEL A MANUAL INITIATION ▪ DIVISION II CHANNEL B MANUAL INITIATION • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs open ◦ Feedwater pumps are running (initially) ◦ All control rods fully inserted • Reports EOP entry conditions on low RPV water level, high Drywell pressure, high Drywell temperature • Performs N2-SOP-101C actions: ◦ Verifies: - All rods full in - Reactor power lowering - Turbine tripped/TSVs & TCVs shut

Instructor Actions / Plant Response	Operator Actions
	- Generator tripped and house loads transferred - SDV V & D Valves closed - RCS pumps downshift - RPV pressure on TBVs or SRVs FWLC controlling level > 159.3 ◦ Fully inserts IRMs and SRMs ◦ Energizes 2WCS-MOV107 ◦ If required, secures makeup to the cooling tower ◦ Shuts down HWC ◦ Performs Level control leg: - Recognizes that level will be stabilized using HPCS, RCIC, CRD, and/or SLC - If directed defeats HPCS Level 8 trips by directing a field operator to perform the actions of N2-EOP-6.20
Note: The candidate may also start HPCS by starting the pump and opening 2CSH*MOV107, rather than using the manual initiation pushbutton. **Booth Operator:** If directed to defeat group 5 isolations for RHR A and B, wait 2 minutes, then **insert malfunction**: **TRG7** **RH08,** Group 5 Isolation Failure – (RHS*MOV122/113), Final = true Then report task completion.	**ATC / BOP** • May manually start RCIC: ◦ Rotate RCIC MANUAL INITIATION pushbutton collar to ARMED position ◦ Depress RCIC MANUAL INITIATION pushbutton ◦ Verify GLAND SEAL SYSTEM AIR COMPRESSOR starts ◦ Verify 2ICS*MOV116, LUBE OIL COOLING WTR SUPPLY, opens ◦ Verify 2ICS*MOV120, TURB STM SUPPLY VLV, opens to start RCIC Turbine ◦ Verify 2ICS*MOV126, PMP 1 DISCH TO REACTOR, opens ◦ WHEN RCIC system total flow exceeds 220 GPM, THEN verify 2ICS*MOV143, PMP MINIMUM FLOW TO SUPPRESSION POOL, closes ◦ WHEN RCIC discharge pressure is greater than reactor pressure, THEN verify 2ICS*V156, REACTOR INJECTION OUTBD CHECK VLV, open ◦ IF at any time, automatic flow control of RCIC is NOT desired, is functioning improperly, OR flow is

Instructor Actions / Plant Response	Operator Actions
	less than 400 GPM, THEN return flow control to manual by placing 2ICS*FC101 to M ◦ Verify RCIC FLOW CONTROLLER establishes AND maintains a RCIC total flow of 600 GPM injecting to Reactor pressure vessel • May manually start HPCS: ◦ At 2CEC*PNL601, ARM AND DEPRESS MANUAL INITIATION ◦ At 2CEC*PNL601, VERIFY the following: ◦ 2CSH*P1, HPCS PUMP 1, started. ◦ 2CSH*MOV107, PMP 1 INJECTION VLV, open. ◦ WHEN system flow is greater than 634 gpm, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV closed. ◦ Flow to RPV as indicated by 2CSH*FI104 flow rising AND indication the following valves are closed: - 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL - 2CSH*MOV110, TEST RETURN TO CONDENSATE TK - 2CSH*MOV112, TEST RETURN TO CONDENSATE TK ◦ At 2CEC*PNL852, VERIFY DIVISION 3 2EGS*EG2 started

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If requested to place SWP radiation monitor RE23A in service, **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = Online **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Power, Final = On If requested to place SWP radiation monitor RE23B in service, **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, Final = Online **RM03-041**, SWP23B Current Radiation Level Power, Final = On Then report 2SWP*RE23A(B) is in service	**ATC / BOP** • Initiates Suppression Chamber spray per N2-EOP-6.22: ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies 2RHS*P1A(B) running. ◦ Opens 2RHS*MOV33A(B) to establish suppression chamber spray flow. ◦ Opens 2RHS*FV38A(B) and establish approximately 7450 gpm – If suppression pool cooling is directed. ◦ Throttles open 2SWP*MOV33A(B) to establish flow not to exceed 7400 gpm - Starts 5th SWP pump (post LOCA). ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay. ◦ Directs RP to place 2SWP*RE23A(B) in service • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE ◦ OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD ◦ OPENS 2IAS*SOV164 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD ◦ OPENS 2IAS*SOV165 • If directed, terminates and prevents Low pressure ECCS injection using N2-EOP-HC attachment 4:

Instructor Actions / Plant Response	Operator Actions
	◦ PLACES 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ IF a LPCI A/LPCS initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI A/LPCS MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24A, LPCI A INJECTION VLV ◦ 2CSL*MOV104, PMP 1 INJECTION VLV ◦ PLACES 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK. ◦ IF a LPCI B & C initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI B & C MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24B, LPCI B INJECTION VLV ◦ 2RHS*MOV24C, LPCI C INJECTION VLV • Closes the outboard MSIVs at pre-established pressure value and shifts pressure control to the SRVs • Recognizes/reports to the SRO that suppression chamber pressure has reached 10 psig • Verifies and reports that both recirculation pumps are tripped • Verifies and reports that all drywell unit coolers are tripped • **Initiates Drywell spray per N2-EOP-6.22 (CT-2.0):** ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies closed 2RHS*FV38A(B) ◦ Verifies RHR A(B) pump running. ◦ Opens 2RHS*MOV33A(B) and verifies flow (If suppression chamber spray is required concurrently with drywell sprays). ◦ Opens 2RHS*MOV15A(B) ◦ Opens 2RHS*MOV25A(B)

Instructor Actions / Plant Response	Operator Actions
	◦ Verifies proper operation of 2RHS*MOV4A(B) ◦ Verifies ~7450 gpm on Drywell Spray Header flow ◦ Verifies open 2SWP*MOV90A(B) ◦ Throttle open 2SWP*MOV33A(B) to establish ~ 7400 gpm ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay.

Event Termination Criteria	• All control rods are inserted. • Reactor level is being controlled in assigned band. • Containment pressure is being controlled per N2-EOP-PC (1-3).

Simulator Scenario Handout Reference Packet

Simulator Scenario Handout Reference Packet

F. NORMAL OPERATION (Continued) **Initials**

NOTE

- The M/A flow controllers have a low and high speed detent to allow speed control of valve movement. Normal operation using M/A flow controllers will be to raise or lower recirculation flow according to power demands. This will be accomplished by operating lever(s) of the appropriate M/A station(s) in control. Loop Manual control of the individual FCV M/A station is the only authorized mode of operation.
- Maximum allowed recirculation flow is 113.9 Mlbm/hr (Computer Point NSSFA01S) (R*Time: (NSSFA101A-01S).
- Do **not** enter the Scram Region or Exit Region of the Power-Flow Operating Map.
- If approaching the loop drive flow limit of 47,200 gpm for two pump high speed operation, use smoothed Computer Point iTARS-20455M (iTARS-20465M) for better control.

F.4. **Normal Flow Control Valve Operation**

F.4.1. **USING** low speed detent of
2RCS-HC1603A(B), RECIRC LOOP A(B) FLOW CONTROL,
RAISE/LOWER output controller as necessary. _____

Constellation

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-101D

POWER CHANGES

Revision 03700

Safety Related

CONTINUOUS USE

10 CFR 50.59 REVIEW REQUIRED

Approval Authority: Director-Operations

SUMMARY OF ALTERATIONS

Minor revision to incorporate the following:

PCR-21-00837:

- **Description:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more.
- **Reason:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more. It makes the procedure consistent with N1-OP-43B. Also, the PSC desires to be notified for all load reductions.
- Step F.3.2, changed bullets to numbers, added notes, and made improvements for notifications.
- Step G.1.1, changed bullets to numbers, added notes, and made improvements for notifications.

PCR-21-00890:

- **Description:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR.
- **Reason:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR. These changes were driven by ECP-20-000231 for the GNF3 fuel modification in RFO18.
- Step D.1.10.b, added MSROOS, PLUOOS, and TCVFASOOS.
- Deleted Allowable Modes of Operation with Inoperable Equipment Table and it’s associated notes.
- Added step D.1.10.c.
- Added P&L D.1.11 added per Reactor Engineering comment.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	4
B.	SYSTEM DESCRIPTION	6
C.	OPERATING REQUIREMENTS	6
D.	PRECAUTIONS AND LIMITATIONS	7
E.	STARTUP	15
	E.1. Power Increase From 45% to 100%	15
F.	NORMAL OPERATIONS	23
	F.1. Rated Power Operation	23
	F.2. Routine Reactor Power Maintenance [C12]	25
	F.3. Power Maneuvers Between 50-100%	27
	F.4. Power Maneuvering for Parameter Margin Control	38
	F.5. Trimming of DEHC Pressure Set	43
G.	SHUTDOWN	44
H.	OFF-NORMAL OPERATIONS	51

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

A.1.1. Technical Specifications

Due to the fact that most, if **not** all, of the technical specifications apply to plant power operation, they are **not** listed individually in this procedure.

A.1.2. Licensee Documentation

- Technical Specification Amendment 123 (ARTS/MELLLA+)
- Updated Safety Analysis Report, USAR
- Consult individual operating procedures as necessary.

A.1.3. Technical Information

- Flow Diagrams

  Consult individual operating procedures as necessary.
- Electrical Diagrams

  Consult individual operating procedures as necessary.
- Vendor Manuals

  Consult individual operating procedures as necessary.
- System Instruction Manuals

  Consult individual operating procedures as necessary.

A.1.4. Supplemental References

- NRC Bulletin 88-07 Supplement 1, Power Oscillations in BWR.
- NMPC 865, GE Project Letter.
- Safety Evaluation 91-049, Revision 1, Extended Load Line Limit Analysis.
- Power-Flow Operating Map (EM-950A) Two RCS Loops Operation.
- Power-Flow Operating Map (EM-950B) One RCS Loop Operation.
- EDC 2M00587 Revision to the Power-Flow Operating Map.
- NMPC File Code ESB2-M95-0193, Condensate/Feedwater System Operation Post Update.
- NMPC File Code ESB2-PM96203, Turbine Pressure Control.
- NER-2E-037, APRM/RBM/Technical Specification – Maximum Extended Load Line Limit Analysis (ARTS/MELLLA+) Project.
- SOER 07-1, Reactivity Management. **[C13]**

A.1.4 (Continued)

- RIS 07-21, Revision 1, Adherence To License Power Limit.
- Amendment 140 to the Facility Operating License, Extended Power Uprate.
- IR 02002103, Unit 2 Required Down Power Due to Offgas Flow Excursions.
- ECP-12-000448, NMP2-Implementation of MELLLA+.
- IR 04296001, Received 2RCS*P1B lower mtr bearing oil level low alarm.

A.2. **Performance References**

- LS-CE-1255, Northeast Site Specific - CE (New York)
- N2-OP-101A, Plant Startup
- N2-OP-101C, Plant Shutdown
- N2-OP-10A, Circulating Water System
- N2-OP-27, Generator H_2 and CO_2 System
- N2-OP-29, Reactor Recirculation System
- N2-OP-3, Condensate and Feedwater System
- N2-OP-30, Control Rod Drive
- N2-OP-42, Offgas System
- N2-OP-5, Condensate Demineralizer System
- N2-OP-8, Feedwater Heaters and Extraction Steam
- N2-OP-9, Condensate Air Removal
- N2-OP-91A, Process Computer
- N2-REP-11, Independent Methods of Determining Core Thermal Power
- N2-RESP-12, APRM/LPRM Noise Level Determination During Low Flow Operation
- N2-RESP-4, LPRM Calibration
- N2-SOP-101C, Reactor Scram
- N2-SOP-101D, Rapid Power Reduction
- N2-SOP-29, Sudden Reduction in Core Flow
- N2-TTP-VIB-V001, Turbine Vibration Readings
- OP-AA-106-101, Significant Event Reporting
- OP-AA-106-101, Significant Event Reporting
- OP-AB-300-1003, BWR Reactivity Maneuver Guidance

A.3. **Definitions**

None

A.4. **Commitments**

Sequence Number	Commitment Number	Description
1	SOER 90-2	Nuclear Fuel Defects
2	SOER 90-3	Nuclear Instrumentation Miscalibration
3	DER 2-92-Q-0050	Control Rod Drive Problems Experienced During 12/13 and 12/16/91 Reactor Startup
4	LER 92-17	Rx Scram Due to Loss of Feedwater Pump
5	LER 95-08	Rx Scram Due to High Turbine Vibration
6	DER 2-95-1737	OE Clarification of Basis for Tech. Spec. Neutron Monitoring Flow Referenced Set Point, Recirculation Drive Flow/Core Flow Correlation
7	DER 2-98-3097	Process Computer Failure
8	DER 2-98-3593	Hot Reheat Steam Temperatures Exceeded 50 °F
9	DER 2-99-0421	OE 9387 differences in definition of TBV, OOS between utilities understanding of operability and GE underlying safety assumptions
10	DER-2-99-3922	OE-SEN 208, Reactor Power Ascension with An Unrecognized Loss of Feedwater Heating
11	DER 2-2000-3818	Potential Recirculation System Jet Pump Cavitation
12	CR-2008-000162	Reactor Power Indications Exceeded License Limit
13	CR 2007-004914	INPO SOER 07-1, Reactivity Management
14	CR 2009-002350	Reactor Power spiked above 1850 MWth during Rod Line Adjustment

B. SYSTEM DESCRIPTION

As all plant systems are affected by the plant power operation, individual system descriptions will **not** be discussed in this procedure. See the individual operating procedures for descriptions of systems that are of interest.

C. OPERATING REQUIREMENTS

All systems and their auxiliaries must be in operation, or in a standby readiness per applicable operating procedure.

D. PRECAUTIONS AND LIMITATIONS

D.1.1. Power can be raised by control rod withdrawal or raising reactor recirculation (recirc) flow. Power changes shall **not** be made by simultaneously using reactor recirc flow and control rod withdrawal.

D.1.2. Control rod movement shall be per OP-AB-300-1003, BWR Reactivity Maneuver Guidance.

D.1.3. During power ascension, for each control rod sequence, fuel pre-conditioning rates shall be maintained and per Reactor Engineering instruction. **[C1]**

D.1.4. Recirculation Pump upshift or pump start shall be performed between the 47% Rodline and the 58% Rodline.

D.1.5. Reactor Engineering will provide and update Rodline information during orderly plant startup and shutdown. If a Reactor Engineer is **not** on site, the Unit Supervisor (US) or the STA will verify the Rodline and update the posting at Panel 2CEC*PNL603.

D.1.6. The Main Turbine should **not** be continuously operated with condenser vacuum less than the low vacuum alarm point (23.8 in. Hg). Turbine load should be reduced under deteriorated vacuum conditions in an attempt to restore vacuum. The Turbine should be manually tripped whenever Turbine load is less than 30% (410 MWe) and annunciator 851359 is in alarm with vacuum less than 24.6 in. Hg.

D.1.7. Under normal conditions hotwell water temperature should be maintained less than or equal to 140 °F or as necessary to ensure CY-AB-120-100, Reactor Water Chemistry, Action 1 limits are **not** exceeded.

During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:

- Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
- RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).

D.1.8. Reactor recirculation flow control shall be maintained in LOOP MANUAL.

D.1.9. Technical Specification Requirements During Power Changes

a. When raising Reactor power, N2-RESP-001, Power Distribution Limits Verification, shall be performed once within 12 hours after exceeding 23% Core Thermal Power (CTP) and every 24 hours thereafter.

b. When Reactor power is greater than or equal to 26% Core Thermal Power (CTP) with any recirculation pump in fast speed, and one or more functions with End of Cycle Recirculation Pump Trip (EOC-RPT) trip capability **not** maintained, then action is required within 2 hours to comply with Tech. Spec. 3.3.4.1. **[C9]**

c. When Reactor power is greater than or equal to 23% Core Thermal Power (CTP), and the Main Turbine Bypass System is found to be INOPERABLE, then action is required within 2 hours to comply with Tech. Spec. 3.7.5. **[C9]**

D.1.10. Extended operating domain and equipment out-of-service analyses have been performed for operations within extended operational boundaries and with various plant equipment out-of-service:

a. Extended Operating Domains

1) For Two Recirculation Loop Operation

Operation is allowed above the Maximum Extended Load Line Limit Analysis (MELLLA) Line, provided 100% rated core thermal power is **not** exceeded.

The region of the Power-Flow Map above the MELLLA Line is known as the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Region. Intentional operation beyond the MELLLA+ Boundary is **not** permitted.

The region of the Power-Flow Map between 100% core flow and 105% core flow is known as the Increased Core Flow (ICF) Region.

2) For Single Recirculation Loop Operation

Single Recirculation Loop Operation (SLO) is restricted by two parameters, namely recirculation flow and rod line. The maximum allowable SLO recirculation drive flow is 41,800 gpm due to recirculation piping vibration limitations and the maximum SLO rod line is 89% (Extended Load Line Limit Analysis (ELLLA) Boundary). These restrictions are also described in the SLO Loop Power-Flow Maps (EM-950B-001 and EM-950B-002). Where these two parameters intersect on the Power-Flow Maps restricts SLO maximum power. Intentional Operation beyond the ELLLA Boundary is **not** permitted.

3) Certain plant transients may result in an operating condition beyond the MELLLA+ Boundary. These transients are analyzed - provided the pre-event conditions were within the normal MELLLA+ Region and actions are taken to exit the region above the MELLLA+ Boundary as part of the transient response procedure.

D.1.10 (Continued)

b. Equipment Out-of-Service (EOOS) Analyses

NMP2 has been analyzed for operation in various plant conditions and with various Equipment Out of Service (EOOS). The following conditions may affect the thermal limits as analyzed in the current COLR:

1) Single Recirculation Loop Operation (SLO)
– Per Tech Spec 3.4.1.

2) Safety Relief Valve Out-of-Service (SRVOOS)
– Per Tech Spec 3.4.4.

3) Automatic Depressurization System Out of Service (ADSOOS)
– Per Tech Spec 3.5.1.

4) MSIV Out-of-Service (MSIVOOS)
– Per Tech Spec 3.6.1.3. Refer to N2-OP-1, Main Steam System

5) Turbine Control Valve Out of Service / Turbine Stop Valve Out of Service (TCV/TSVOOS)
– Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

6) Turbine Bypass Valve Out of Service (TBVOOS)
– Per Tech Spec 3.7.5. Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

7) Recirculation Pump Trip Out of Service, also known as EOC-RPT (RPTOOS)
– Per Tech Spec 3.3.4.1.

8) Pressure Regulator Out of Service (PROOS)
– Per Tech Spec 3.2.2.

9) Moisture Separator Reheaters Out of Service (MSROOS)
– Per the COLR.

10) Power Load Unbalance Out of Service (PLUOOS)
– Per the COLR.

11) Turbine Control Valve Fast Acting Solenoids (TCVFASOOS)
– Per the COLR.

D.1.10 (Continued)

c. GNF3 fuel introduces new equipment out service analyses and combinations of out of service equipment. When out of service equipment is identified, utilize Table 4-2 of the COLR to determine the appropriate penalty to apply for the specific condition that exists. Reactor Engineering assistance may be required to determine the correct penalty to apply for equipment out of service.

d. See the Cycle Specific Core Operating Limits Report for applicable thermal limit impacts for various EOOS conditions [FCMS Search = COLR%].

e. Reference the following to identify the allowable combination of conditions:

- 003N2077-R0, Nine Mile Point 2 GNF NFI T0900 Coincident Equipment out-of-service (EOOS) Report [FCMS Search = 003N2077]
- 006N0918-R0, TSD NF-B756 Nine Mile Point Unit 2 Cycle 18 RPTOOS + TBVOOS Thermal Limits [FCMS Search = 006N0918%]

f. **IF** a specific combination of conditions is **not** identified as allowable, **THEN** it is **not** supported **AND** an orderly plant shutdown to less than 23% power is appropriate.

D.1.11. Operation with Turbine Bypass Valves out of service (TBVOOS) coincident with Turbine Control Valve/Turbine Stop Valve out of service (TCV/TSVOOS) is **not** an allowable condition. It is not supported by the licensing analysis per the COLR.

D.1.12. When Reactor power is greater than 23%, Reactor Engineering notification is required before raising Reactor power with recirculation flow or control rod withdrawal except if power is being raised per Reactor Operating Instructions, Section F.2, Power Maintenance or Section F.4, Power Maneuvering for Parameter Margin Control.

D.1.13. All reactivity changes shall be directly supervised by a Senior Reactor Operator. In order to provide this oversight, the SRO shall be stationed at the Controls Area of the Control Room.

D.1.14. Because of many possible plant conditions, deviation from suggested sequence of operations is permitted as directed by the Shift Manager (SM).

D.1.15. Because the Core Thermal Power Application uses "smooth" feedwater flow inputs, it should **not** be used for Rodline determination during or immediately after power transients. Use APRM power and instantaneous core flow values to compute/plot Rodline from power/flow operating map (Ref. A.1.4, EM-950A or EM-950B Power Flow Operating Map).

D.1.16. If Turbine bearing vibration rises significantly (e.g. to a value of greater than or equal to 5 mils or greater than or equal to 2 mil rise) as a result of Turbine load changes, power should be held steady at the nearest practicable level after the vibration rise. **[C5]**

Once the vibration lowers to an acceptable stable value, Turbine load should be held steady for an additional hour before continuing with load changes. This will allow the Turbine rotor(s) to straighten from any rub induced bowing.

D.1.17. When CTP is greater than or equal to 23% and any APRM indicates outside plus or minus 2% of CTP, an APRM gain adjustment shall be performed prior to additional changes in reactor power.

D.1.18. When power is greater than 90% and prior to any power increase, a Core Thermal Power Report/OD-3 from R*TIME should be demanded to ensure there are **no** unplanned failed sensors inputting into the CTP calculation. Any failed sensor(s) will show on the report under the heading “Failed Sensors List.” (N2-OP-91A has a list of sensors.) **[C6]**

D.1.19. The process computer can fail in ways that are **not** obvious to operators. Operators should **not** add reactivity when the core thermal power calculation is **not** accurate. Operators need to know power level with certainty prior to making adjustments. **[C6] [C7]**

D.1.20. When Reactor Thermal Power is greater than 70% with charcoal adsorbers in service, and Offgas System flow is greater than 25 SCFM the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B sample consistent with the effluent path and maintain proper margin to the trip setpoints.

D.1.21. The Charcoal Adsorber must be in service prior to placing the Offgas Radiation Monitor Delay Pipe in service.

D.1.22. MWth oscillation during steady state operation is a known situation for BWRs using recirculation risers and jet pumps. During steady state operations, if MWth oscillation is experienced, greater than normal expected values, Reactor Engineering should be contacted to adjust rod line or reactor recirculation flow.

D.1.23. The averaging algorithm associated with core thermal power computer points NSSQB001A-01M and NSSQB001A-10M may **not** respond accurately to certain variations in real core thermal power such as oscillations brought about by rapid cycling of recirculation flow. The following computer alarms are provided to ensure that design margins are maintained. Accident analyses provide a 2% margin to account for uncertainties associated with the determination of core thermal power. If any of these alarms are actuated immediate operator action is required to ensure that thermal power is trending below 3988 MWth.

- NSSQB001A_01S, CORE THERMAL POWER – 1-SEC INSTANT Setpoint = 4028 MWt
- NSSQB001A-01M, CORE THERMAL POWER – 1 MIN SMTHD CT POWER Setpoint = 4008 MWt
- NSSQB001A-10M, CORE THERMAL POWER – 10 MIN SMTHD CT POWER Setpoint = 3998 MWt

D.1.24. The R*TIME monitoring above 2CEC*PNL603 is set up to display whole numbers only; decimals are truncated. Caution must be used when monitoring plant parameters using these displays.

D.1.25. Maintain power and flow in the operating domain of the applicable single and two loop power-flow operating maps. Determine expected operating location prior to making reactivity changes. **[C11]**

D.1.26. Primary Containment Oxygen shall be controlled per the requirements of Technical Specification 3.6.3.2.

D.1.27. A CRD Deficiency Log entry should be made at any time during the performance of this procedure when a control rod drive mechanism does **not** perform as expected.

D.1.28. During noble Metals Injection, the plant prerequisite is for core flow greater than or equal to 85% flow and reactor power greater than or equal to 70%. Notify Chemistry Department if outside of these parameters during Noble Metals Injection.

D.1.29. When making core reactivity changes, operators shall monitor redundant nuclear instrumentation as well as other plant indications. Instrumentation to be used but **not** limited to includes SRM count meters, SRM period meters, IRM recorders, APRM recorders/monitors, LPRM meters/monitors, Instantaneous and "Smoothed" Power computer points, Generic Output, Recirc Flow indications, Turbine Bypass and Control Valve positions and Reactor Pressure.

D.1.30. For planned power changes 10% or greater, S.O.S. oversight is required.

D.1.31. Fuel precondition is based on raising fuel bundle nodal power at a rate of less than or equal to 0.35 kW/ft per hour. This translates into raising power at a rate of approximately 3%/hour. This can be accomplished as follows:

a. Raise reactor power by 3% over a one-hour period.

b. Raise reactor power by 3% at any time during the one hour-period, but remain at that power level for the remainder of the hour.

D.1.32. If the 3% rate is exceeded over the first hour, then this can be adjusted for in the next hour. Conversely, if the power increase falls short of the 3% rate over the first hour, this power can be adjusted for over the next hour. However, when adjusting for a shortfall in the first hour, it is **not** permitted to exceed a rate of 3.5% per hour in the second hour.

D.1.33. Operating the reactor in compliance with the Licensed Power Limit is demonstrated by the following process;

a. **No** actions are allowed that would intentionally raise core thermal power above 3988 MWth for any period of time. Small, short-term fluctuations in power that are **not** under the direct control of a licensed Reactor Operator (e.g. fluctuation caused by bi-stable flow) are **not** considered intentional. Due to the change in the reactor bi-stable flow characteristics following N2R16 outage, positive reactivity addition for Power Maintenance shall **not** be made unless instantaneous, one minute and ten-minute thermal power is less than or equal to 3986 MWth and trending lower.

b. Closely monitor thermal power during steady state power operation with the goal of maintaining the rolling 1 hour thermal power average at or below 3987.5 MWth. If the core thermal power average for a rolling 1 hour period is found to exceed 3987.5 MWth, take action to ensure that thermal power is reduced to less than or equal to 3987.5 MWth. Power Maintenance shall be performed before exceeding 3987.5 MWth on the rolling two hour thermal power average.

c. The core thermal power average for a shift is **not** to exceed 3988 MWth. For the purpose of this guidance, a shift is defined as 0600 to 1800 or 1800 to 0600.

d. During End of Cycle Operations, Rx power shall **not** be raised above the Rx power which can be achieved with steady state equilibrium Xenon concentrations in the core (all control rods out, all Recirc Pumps in service, raise Core Flow), as determined by Reactor Engineering.

D.1.34. When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition. When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rate at 3966 MWth and above will be done with recirculation flow. **[C14]**

- The above precaution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, “Reverse Power Effect”, for power maintenance. There should be a minimum of three minute wait prior to any further adjustments with Recirculation Flow.

D.1.35. “Steady State” for the purpose of maintaining licensed power level is defined as:

- **No** control rod motion
- **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
- Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
- Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).

It is **not** intended that “steady state” include equilibrium Xenon or Samarium.

D.1.36. The Feedwater minimum flow valves, 2FWR-FV2ABC, are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:

- Single Pump Operation - 23% to 27% Power
- Two Pump Operation with balanced flows - 46% to 53% Power

D.1.37. Failure to have 3 Condensate Booster Pumps (if **not** inhibited) and 2 Feedwater Pumps in service prior to Steam flow exceeding 11.5 Mlbm per hour (approximately 72% power) will result in a Recirc Flow Control Valve Runback.

D.1.38. With MSRs out of service, special restrictions apply to continuous operation with a heater drain pump out of service. The plant can operate with one heater drain pump out of service provided continuous operation is limited to less than or equal to 69% of rated thermal power. With more than one heater drain pump out of service, reactor power is limited to less than or equal to 56%.

D.1.39. Excessive cycling of the feedwater heater dump valve could damage the associated condenser penetration resulting in an increase in Offgas flow. Immediate actions should be taken to stop/avoid cycling of the heater dump valves.

D.1.40. The following parameters are required to be monitored during each reactivity manipulation:

- Redundant Nuclear Instrumentation
- Core Flow
- Reactor Pressure
- Reactor Temperature

D.1.41. In October 2019, station personnel performed Drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.

D.1.42. During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power), vibration of EHC System Piping and components should be closely monitored. In March 2020, NMP Unit 2 required a manual SCRAM in response to a large EHC leak from to a weld failure caused by piping vibrations after an extended period of operation at this power level. More frequent monitoring of EHC piping and system components should be performed when operating in this “high EHC flow” region. If excessive vibration of EHC piping and system components occur, an evaluation should be made to raise or lower Reactor Power to avoid operation in this region.

Initials

F.3. **Power Maneuvers Between 50-100%**

NOTE

- This section is for performing plant maneuvers between 50% to100%, this section assumes that the plant startup has been completed and that power reduction / ascension is **not** from an outage.
- The steps in this section are a guide for plant maneuvers, if it is determined a plant shutdown is required then entry into the shutdown section of this procedure should occur.
- The steps in this section may be performed out of sequence as determined by the Shift Manager.
- In October 2019, station personnel performed drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.
- During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power) vibration of EHC System Piping and components should be closely monitored. Unit 2 required a manual SCRAM in response to a large EHC leak in March 2020, after a weld failure caused by piping vibrations after an extended period of operation at this power level.
- The Feedwater minimum flow valves, 2FWR-2FVA(B,C), are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:
  - Single Pump Operation 23% to 27% Power
  - Two Pump Operation with balanced flows - 46% to 53% Power

N/A F.3.1 **IF** load reduction is in support of Drywell (DW) entry at power,
THEN NOTIFY Outage Control Center that visual verification of adequate oil level in both Reactor Recirculation Pump (2RCS-M1A and 2RCS-M1B) Motor Bearings (Upper and Lower) must be completed
PRIOR TO Drywell exit,
OTHERWISE MARK this step N/A.

N/A

N/A F.3.2 **PERFORM** the following power reduction preparatory steps, **AND** Steps **not** required may be marked N/A:

NOTE

Steps F.3.2.a thru F.3.2.l may be performed in any order.

a. **IF** prior to a planned shutdown **OR** significant load drop, **THEN INSTRUCT** Plant Performance Group to install vibration data gathering instrumentation at 2CEC-PNL841 per N2-TTP-VIB-V001, **UNLESS** waived by System Engineer..... ☐

NOTE

Per OP-AA-102-101, the following expectations/requirements of timeliness for notifications to Generation Dispatch:

- 2-hour notice is desirable
- 30-minute notice is required
- Immediately before is desirable

b. **VERIFY** the following have been notified of planned power changes:

- Generation Dispatch..... ☐
- Power Control..... ☐
- NDO ..... ☐

NOTE

An unplanned load reduction is one that was **not** previously scheduled.

c. **IF** conducting an unplanned load reduction of 15% **OR** greater, **THEN NOTIFY** Public Service Commission Notification per the following:

1) OP-AA-106-101, Significant Event Reporting ..... ☐

2) LS-CE-1255, Northeast Site Specific - CE (New York) ..... ☐

Initials

N/A F.3.2 (Continued)

d. **IF** containment will be de-inerted,
THEN VERIFY GSN Electric Vaporizers are available ..... ☐

e. **IF** Condensate Demineralizers will be removed,
THEN VERIFY removal order has been developed..... ☐

f. **IF** Clean Steam Reboilers will be swapped to the Aux Boilers,
THEN PREPARE Aux Boilers to supply Clean Steam Reboilers..... ☐

g. **IF** Intermediate Heat Exchangers will be removed from service,
VERIFY Auxiliary Heat Exchangers are in service. ..... ☐

h. **IF** entering Mode 3,
THEN VERIFY all Mode 3 required surveillances are completed
OR ready for completion ..... ☐

i. **IF** desired,
THEN VERIFY N2-OP-25, H.6.0, Preventing Automatic Swapover of Reboiler Steam Supply, is complete..... ☐

j. **IF** planning to reject WCS to the Main Condenser**,**
THEN VERIFY 2NHS-MCC2008-2E, 2WCS-MOV107, RWCU DISCH TO MAIN COND is ON..... ☐

k. **IF** Cooling Tower will be changed to full bypass,
THEN VERIFY Basin Level is adjusted for Cooling Tower Mode Change (Bypass) ..... ☐

l. **IF** performing Scram time testing per N2-OSP-RMC-@001,
THEN prior to commencing power reduction,
VERIFY communication equipment is staged
AND prerequisites are performed. ..... ☐

N/A

Initials

F.3.3. Prior to planned power change,
EVALUATE, ASSIGN AND RECORD below any system critical parameter(s) that may be in an abnormal control lineup which may be adversely affected by the load drop (check AIU book; for example, Gland Sealing Steam):

System Parameter	Operator Responsible (i.e., OATC)
None	N/A

OPE

F.3.4. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation ☑
- Core Flow ☑
- Reactor Pressure ☑
- Reactor Temperature ☑

OPE

F.3.5 **DURING** power change, some of the following systems will require periodic adjustments depending on the significance of the drop.
DIRECT OATC or other available RO(s) to
PERIODICALLY MONITOR the following system parameters
AND NOTIFY the US for adjustment as necessary:

- Reactor Water Cleanup Filter Flows ☑
- Cooling Tower Basin Level/Temperature ☑
- Main Generator Reactive Load (MVARs) ☑
- Turbine Vibrations ☑
- Main Condenser Vacuum ☑
- Gland Sealing Steam Pressure ☑
- Gland Seal Exhaust Pressure ☑
- Clean Steam Reboiler Level Control ☑
- Condensate Demineralizer System/Demin Flow ☑
- Feedwater Level Control Response ☑
- MSR Drain Tank Oscillations ☑
- Containment Parameters ☑

OPE

Initials

N/A F.3.6. Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ☐
- **NOTIFY** NDO if required per OP-AA-102-101. ☐
- **NOTIFY** Chemistry to monitor Offgas chemistry per NF-AB-400-1000, BWR Fuel Integrity Monitoring ☐
- **REVIEW** applicable Precautions and Limitations. ☐
- **OBTAIN** authorization from Shift Manager to reduce power. ☐
- **REVIEW** RMI for planned power reduction. ☐
- **COMPLETE** applicable reactivity brief. ☐

N/A

b. **PERFORM** the following actions as needed **DURING** power reduction:

1) At 2CEC*PNL851, **PERIODICALLY MONITOR** Offgas System Flow Rate using 2OFG-FI3A/B. ☐

2) **IF** automatic make up air to Offgas is **not** available, **THEN WHEN** Offgas flow is less than or equal to 6 SCFM, **PERFORM** the following, **OTHERWISE MARK** these steps N/A:

   a) **REMOVE** pipe cap from 2ARC-V47, ARC SUCTION HDR VENT. ☐

   b) **THROTTLE OPEN** 2ARC-V47 to establish approximately 7 to 10 SCFM on each indicator (2OFG-FI3A/3B). ☐

3) **LOWER** power as directed by approved ReMA. ☐

4) **MONITOR** core flow / reactor power on Power to Flow Map. ☐

5) **IF** core flow / power reduction results in entry into the exit region, **THEN ENTER** N2-SOP-29, **OTHERWISE MARK** this step N/A. ☐

6) **MONITOR** rodline as needed based on core conditions. ☐

7) **MONITOR** Feedwater Pump Seal temperatures **AND ADJUST** seal flows per N2-OP-3. ☐

N/A

N/A F.3.6 (Continued)

c. **PRIOR TO** going below 80% power perform the following:

1) **MAINTAIN** HVH steam supply on Extraction Steam as follows:

NOTE

If plant trip or power reduction below 50% occurs, building heating could be lost because this step prevents the auto swap from Extraction to Aux Steam.

a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in CLOSE. ..... ☐

2) **MAINTAIN** Clean Steam Reboiler steam supply on Extraction Steam at P851 as follows:

NOTE

If plant trip or power reduction below 50% occurs, the Extraction Steam supply pressure could be insufficient to support clean steam reboiler operations and result in an auto swap to the emergency sealing steam supply.

a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in CLOSE. ..... ☐ N/A

d. **WHEN** Steam flow goes below 11.5 Mlbm/hr (approximately 72% power),
CONFIRM RECIRC RUNBACK INHIBITED lit on 2CEC*PNL602. N/A

Initials

N/A F.3.6 (Continued)

e. **CONTINUE** power reduction
AND PERFORM the following as required:

- **REMOVE** Condensate Demineralizers from service as required
AND at recommendations of Chemistry per N2-OP-5, Condensate Demineralizer System. ..... ☐
- At approximately 65%, **PLACE** Heater Drain pumps in recirculation per N2-OP-8, G.1.0. ..... ☐ N/A

f. **DURING** power reduction below 65%,
PERIODICALLY MONITOR reheat steam pressures
AND temperatures to ensure LP Turbine nozzle side-to-side inlet temperature differential is maintained less than or equal to 50 °F. N/A

Initials

F.3.7 Power Ascension

a. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation ..... ☑
- Core Flow ..... ☑
- Reactor Pressure ..... ☑
- Reactor Temperature ..... ☑ OPE

b. **PERFORM** the following actions as needed during the power ascension:

1) **MONITOR** core flow / reactor power on power to flow map. ... ☑

N/A 2) **IF** core flow / power ascension results in entry into the exit region,
THEN ENTER N2-SOP-29,
OTHERWISE MARK this step N/A. ..... ☐ N/A

3) **MONITOR** rodline as needed based on core conditions. ..... ☑

4) **MONITOR** Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☑

5) **WHEN** Reactor power is greater than 65%,
THEN VERIFY the following Reheat Steam Load Control Valves are full open:

- 2MSS-PV28A ..... ☑
- 2MSS-PV28B ..... ☑
- 2MSS-PV29A ..... ☑
- 2MSS-PV29B ..... ☑

6) **WHEN** Steam flow exceeds 11.5 Mlbm/hr (approximately 72% power), **CONFIRM** RECIRC RUNBACK INHIBITED light on 2CEC*PNL602 extinguishes ..... ☑

NOTE

Eight condensate demineralizers are required for 100% power with nine condensate demineralizers preferred for additional operational margin.

7) **PLACE** Condensate Demineralizers in service as required per N2-OP-5
AND Chemistry guidance for restoring demins to service. ..... ☑ OPE

Initials

F.3.7 (Continued)

N/A c. **WHEN** reactor power is greater than 80%,
THEN PERFORM the following:

1) **RESTORE** HVH steam supply automatic swap to Aux Steam as follows:

   a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in AUTO. ..... ☐

   b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in AUTO. ..... ☐

2) **RESTORE** Clean Steam Reboiler automatic swap to Aux Steam at P851 as follows:

   a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in AUTO. ..... ☐

   b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in AUTO. ..... ☐ N/A

F 3.7 (Continued)

d. **WHEN** Power is greater than 90%,
PERFORM the following:

1) **WHEN** Power is greater than 90%
AND PRIOR TO any further power increase,
THEN DEMAND an OD-3 to ensure there are **no** unplanned failed sensors that input to CTP calculation. ..... ☑

NOTE [C14]

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.

- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rated at 3966 MWth and above will be done with recirculation flow. This caution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, "Reverse Power Effect", for power maintenance. There should be a minimum of three-minute wait prior to any further adjustments with Recirculation Flow.

2) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY a smoothed feedwater flow computer point (NSSQB108) is updating,
OTHERWISE MARK this step N/A. ..... ☐

3) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY proper feedwater heating by using N2-OP-8, Feedwater Heaters and Extraction Steam System, Feedwater Temperature / Thermal Power Limit Figure,
OTHERWISE MARK this step N/A. ..... ☐

4) **IF** Offgas make-up air is **no** longer necessary,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a) **CLOSE** 2ARC-V47..... ☐

b) **INSTALL** cap on 2ARC-V47. ..... ☐ _____

F.3.7 (Continued)

NOTE

- "Steady State" for the purpose of maintaining licensed power level is defined as:
  - **No** control rod motion
  - **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
  - Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
  - Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).
- It is **not** intended that "steady state" include equilibrium Xenon or Samarium.

e. **WHEN** reactor has been at constant power for greater than two hours,
THEN DECLARE reactor is at "Steady State". _____

f. **NOTIFY** NDO per OP-AA-102-101. _____

g. **PERFORM** Power Maintenance per section F.2 as necessary. _____

NOTE

- 2ISC*PT108 which outputs to FWSPA101 and TARS-2237, should be used for setting reactor pressure as FWSPA101 inputs into the CTP calculation.
- If 2ISC*PT108 is **not** available or reliable, use PT109 (TARS-2236) or PAM recorders for pressure adjustment. Call Reactor Engineering for guidance regarding substituting Rx pressure value in CTP calculation and System engineering for evaluation.

h. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
ADJUST Digital EHC pressure set **UNTIL ONE** of the following is met, **OTHERWISE MARK** this step N/A.

☐ Reactor dome pressure is 1020 psig ..... ☐

☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. ..... ☐ _____

i. **ADJUST** movement restrictions for 2GMC-TV101 via Side-Mounted Handwheel per N2-OP-26. _____

F.4. **Power Maneuvering for Parameter Margin Control**

NOTE

- This section is intended to be used to maintain condenser vacuum, hotwell temperature, and generator loading parameters within acceptable limits while at or near rated conditions
- Rated generator output at 3988 MWth is 1369 MWe. However, variations in generator output above 1369 MWe are permitted provided generator operation remains within the generator capability curve of N2-OP-68. The 233 MVAR lagging reactive load limit is based on a generator real load of 1380 MWe,
- Minimum condenser vacuum above 55% power is 23.8" Hg Vacuum.
- During Normal Operation, maximum operating hotwell temperature is less than or equal to 140 °F.
- During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:
  - Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
  - RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).
- Power changes using this section may be performed using Reactor Recirculation flow without a Reactivity Maneuvering Instruction (RMI). IF Control Rods are used, an RMI is required.
- This section is written to maneuver the plant between 85% and 100% as needed to control the above parameters in a controlled manner without having to perform a rapid power reduction per N2-SOP-101D. If further power reduction is necessary, use Section F.3.0 of this procedure or N2-SOP-101D as applicable.

F.4.1. Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ..... ☐
- **NOTIFY** NDO if required per OP-AA-102-101. ..... ☐
- **REVIEW** applicable Precautions and Limitations. ..... ☐
- **OBTAIN** authorization from Shift Manager to reduce power. .. ☐
- **IF** a ReMA is to be used for the power reduction,
  THEN REVIEW ReMA**,**
  OTHERWISE MARK this step N/A. ..... ☐
- **COMPLETE** applicable reactivity brief. ..... ☐ _____



Simulator Scenario Handout Reference Packet

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 90%.
- LPCI pump 2RHS*P1C is out of service for maintenance. TS 3.5.1 Condition A applies (Day 1 of 7).

PART III: Remarks/Planned Evolutions:

- Start CCS pump 2CCS-P1C per N2-OP-14 section F.3.0. The procedure is in progress up to step F.3.2. Secure CCS pump 2CCS-P1A and place it in standby per N2-OP-14 section F.4.0.
- Then, raise Reactor power to 95% using Recirculation flow per the provided ReMA.



NINE MILE POINT NUCLEAR STATION UNIT 2
OPERATING PROCEDURE

N2-OP-14

TURBINE BUILDING CLOSED LOOP COOLING SYSTEM

Revision 01400

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director Operations

SUMMARY OF ALTERATIONS

PCR-19-02100

- Description: Change CCS temperature band.
- Reason: Provide a consistent achievable temperature band whenever maintaining CCS temperature in manual.
- The following specific changes made:
  - H.1.5 updated temperature band. **[SOP protected]**
  - H.6.6.b updated temperature band.
  - H.7.1.a updated temperature band.

PCR-21-02572

- Description: Add new Off Normal Sections to allow local manual control of selected Temperature Control Valves in case of control issues.
- Reason: This is an output of Equipment Reliability Vulnerability Review [ERVR] process.
- The following specific changes made:
  - H.9, H.10, and H.11 added per approved PCR content.

Operations Writer Changes

- Updated cover page with updated company logo, **no** revision bar.
- Applied updated Operations Procedure template.
- Updated N/A statements to comply with AD-NM-101-1005 5.4.C.13 and in preparation for the transition to Smart Procedures, by removing N/A check-boxes, made into **IF/THEN** statements and "**OTHERWISE MARK** this step (or these steps) N/A", **no** revision bars.
- Restructured multiple procedure steps to **ACTION VERB** first, **no** revision bars.
- Updated fonts for **ACTION VERBS** and **LOGIC WORDS**, bold and lowercase underline for **not**, **cannot**, and **no** per AD-NM-101-1005 and AD-AA-101-1002, **no** revision bars.
- A.6 updated with technical information associated with changes made.
- A.7 updated commitment numbers to IR's for ease in finding in passport.
- D. PRECAUTIONS AND LIMITATIONS deleted legacy ALARA and RP items per IR 04172818, PIIM 2018-0380 Operations Procedure Excellence.
- H.5 added title of NER 2M-069 document.

Ops System Owner Review Changes

- D.8 removed specific step number and replaced with title of step to be consistent with other procedures.
- H.2.8 removed old step associated with placing legacy caution [configuration] tags.

Operator Functional Review Changes

- Remove new off-normal section for manual control of 2CCS-TV109 as **no** handwheel exist to operate the valve. Change was agreed to by PCR Requestor.
- Add locations of local air isolation valves associated with 2CCS-TV32A/B.
- Add location of 2CES-RAK230.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 4
B. SYSTEM DESCRIPTION 6
C. OPERATING REQUIREMENTS 8
D. PRECAUTIONS AND LIMITATIONS 9
E. STARTUP 10
E.1. Venting and Starting the CCS System [SOP] 10
F. NORMAL OPERATIONS 21
F.1. Normal Operating Indications [SOP] 21
F.2. Filling and Venting an Idle Pump with System Operating 23
F.3. Starting an Idle Pump [SOP] 25
F.4. Secure an Operating Pump 26
F.5. Filling and Venting an Idle Heat Exchanger with System Operating 27
F.6. Placing an Idle Heat Exchanger in Service [SOP] 29
F.7. Removing a Heat Exchanger from Service 31
F.8. Swapping Heat Exchangers with Plant On Line 32
F.9. Verifying CCS Flows to Stator Water and Generator Hydrogen Heat Exchangers 34
G. SHUTDOWN 41
G.1. Securing the CCS System 41
H. OFF-NORMAL OPERATIONS 43
H.1. Local Manual Control of 2CCS-TV104 [SOP] 43
H.2. Operation of Two Pumps with Flow greater than or equal to 16,000 GPM 45
H.3. Bypassing the MWS Makeup Flow Totalizer 47
H.4. Restoring the MWS Makeup Flow Totalizer 48
H.5. Supplying Temporary Cooling to the Condensate Pump Motor Thrust Bearing Oil Coolers During CCS Outages 50
H.6. Bypassing 2CCS-TCV104 for Maintenance 58
H.7. Restoring 2CCS-TCV104 Following Maintenance 59
H.8. CCS Feed and Bleed via Turbine Building Equipment Drains 60
H.9. Bypassing Hydrogen Temperature Control Valve 2CCS-TV109 61
H.10. Local Manual Control of Exciter Air Temperature Control Valve 2CCS-TV32A 62
H.11. Local Manual Control of Exciter Air Temperature Control Valve 2CCS-TV32B 63
Attachment 1, CCS System Cooling Loads 64
Attachment 2, Isolated CCS System Loads 66
Attachment 3, Temporary Cooling To Condensate Pumps During CCS Outages 67

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

None

A.2. Licensee Documentation

Updated Safety Analysis Report (USAR), Volume 18, Section 9.2.7, Turbine Building Closed Loop Cooling Water System

A.3. Standards, Regulations and Codes

None

A.4. Policies, Programs, and Procedures

- N2-OP-3, Condensate and Feedwater System
- N2-OP-5A, Condensate Filtration System
- N2-OP-11, Service Water System
- N2-OP-16, Makeup Water Storage And Transfer
- N2-OP-19, Instrument and Service Air System
- N2-OP-71B, 4.16KV AC Power Distribution
- N2-OP-71C, 600V AC Power Distribution
- N2-OP-73A, Normal DC Distribution
- N2-OP-79, Radiation Monitoring
- N2-OP-91A, Process Computer

A.5. Technical Information

A.5.1. Flow Diagrams

- P&IDs 14A - 14F, Turbine Building Closed Loop Cooling

A.5.2. Electrical Diagrams

- ESK-5CCS01 (02,03), Turb Bldg Cl Loop Clg Wtr Pp 2CCS-P1A(B,C)
- ESK-6CCS03 (04), Offgas Condenser Outlet Valve 2CCS-MOV45A(B)
- ESK-7CCS01, Turb Bldg Cl Loop Clg Wtr Aux

A.5.3. Logic Diagrams

- LSKs 9-7A - 7E, Turbine Bldg Closed Loop Cooling Water

A.6. Supplemental References

- SE 92-083, Temporary Hose Installations on the Makeup Water System to Support Maintenance
- NMP2 Calc A10.1-N-341, Service Water Flow to Control Room Chiller Condensers
- TS Amendment 119, 84 °F UHS Temperature
- DCP N2-05-111, Implement 84 °F UHS Temperature Without 82 °F Averaging
- NMP2 Calc HVK-011, Evaluation of Control Building Chillers with Service Water Temperature of 84 °F
- NMP2 Calc A10.2-Q-005, CCS Line Sizing for Cooling Water Supply to CFS System Air Compressor
- ECP-10-000646, Isolate Retired Loads And Re-Balance CCS Flows
- ECP-10-000082, Remove Existing OFG Refrigeration Units and Install New Packaged Chilled Water Units.
- IR 02632390, ERVR – Generator H_2 Cooling Valve Vulnerability.
- NER 2M-069, Alternate Cooling Sources Refueling Outages.

A.7. Commitments

- **[C1]** IR 02078902 (Legacy DER 2-96-3250), Elevated SWP Temperature Impact of CCP and CCS.
- **[C2]** IR 02077189 (Legacy DER 2-96-1516), SWP Design Bases.

B. SYSTEM DESCRIPTION

The Turbine Building Closed Loop Cooling (CCS) System provides demineralized water to cool auxiliary equipment located in the Turbine and Radwaste Buildings.

Three 50% capacity, horizontal, centrifugal pumps (2CCS-P1A, B and C) are arranged in parallel to circulate cooling water to the closed loop circuit. Each pump is rated at 8,000 GPM at 187 feet TDH. The CCS Pumps will auto trip on sustained bus undervoltage, motor overload/electrical fault or low suction pressure of less than or equal to 25 psig. A standby pump will auto start on low discharge pressure of less than or equal to 95 psig.

A 1,400 gallon Surge And Makeup Tank (2CCS-TK1) is located above the pump suction header on the Turbine floor at Elevation 326. 2CCS-TK1 will reduce pressure surges, permit thermal expansion, provide the required NPSH for the CCS pumps, and provide a low pressure inlet for makeup water to the CCS System. Makeup water is automatically added from the MWS System through 2CCS-AOV105, CCS SURGE TANK MAKEUP WTR LEVEL CONTROLLER, to maintain level 55 - 70 inches. This valve is controlled by level switch 2CCS-LS105 which is located on the surge tank level tree. Makeup to the CCS System is indicated on an integrating flowmeter used to check for system leakage. The tank is provided with high and low level alarms and is normally vented to atmosphere.

Cooling water from the CCS Pumps flows through a common discharge header to 2CCS-TV104, TB CLOSED LOOP COOLING TEMPERATURE CONTROL VALVE. 2CCS-TV104 is a tee-linked butterfly valve which is actually two valves, one in the supply to the CCS Heat Exchangers, and one in the bypass line around the heat exchangers. The two valves making up 2CCS-TV104 are joined by a common operator linkage arranged such that water is directed to or around the heat exchangers depending on outlet temperature. The controller for 2CCS-TV104 receives a signal from temperature element 2CCS-TE104 which monitors CCS common header temperature after mixing from the heat exchanger outlet and bypass headers. 2CCS-TV104 is positioned to maintain supply header temperature 80 °F to 95 °F. On a loss of air, 2CCS-TV104 positions itself to provide maximum cooling. The valve in the heat exchanger supply header will fail open and the valve in the heat exchanger bypass line fails closed.

The three 50% capacity CCS Heat Exchangers are single pass, opposed flow, shell and tube type. CCS water flows through the shell side with service water flowing through the tubes. Piping and valves are provided to permit normal operation with two heat exchangers in service with the third in standby or secured for maintenance. During periods of high Service Water temperature conditions (greater than 79 °F), three CCS Heat Exchanger operation will be required. Service Water flow will be limited under these conditions.

Cooling water is supplied to the components and systems as listed in Attachment 1, CCS System Cooling Loads. All component heat exchangers are equipped with manual inlet and outlet valves to allow isolation for maintenance. The outlet isolation valves provide flow control for heat exchangers without temperature control valves. These valves are also used to balance system flow.

B SYSTEM DESCRIPTION (Continued)

A relief valve is installed between the component isolation valves for over pressure protection when the components are isolated. These relief valves discharge to the Turbine Building Equipment and Miscellaneous Drain Systems.

Cooling water returning from all supplied components joins a common return header which is monitored for pH, conductivity, and total CCS System flow. The CCS return from Radwaste is also monitored for high radiation using radiation monitor 2CCS-RE152. An alarm is provided in the Control Room if a pre-determined radiation level setpoint is exceeded. Computer Point CCSFA01, TBCLCW PMP SUCTION FLW, provides total flow indication.

C. OPERATING REQUIREMENTS

C.1. The following systems must be in operation to support the CCS System:

- N2-OP-11, Service Water System
- N2-OP-16, Makeup Water Storage And Transfer
- N2-OP-19, Instrument and Service Air System
- N2-OP-71B, 4.16KV AC Power Distribution
- N2-OP-71C, 600V AC Power Distribution
- N2-OP-73A, Normal DC Distribution
- N2-OP-79, Radiation Monitoring
- N2-OP-91A, Process Computer

D. PRECAUTIONS AND LIMITATIONS

D.1. CCS Pump start limitations are as follows:

- Two starts in one hour from ambient temperature. A one hour wait is required before additional start attempts.
- One start at rated temperature (obtained after 30 minute run). A one hour wait is required before additional start attempts.

D.2. CCS Pump operating limitations are as follows:

- Maximum operating motor current is 58.5 amps.
- Minimum flow is 1,000 GPM.
- Maximum (runout) flow is 8,500 GPM.

D.3. Flow through a CCS Heat Exchanger shall be maintained less than 8,500 GPM to prevent causing tube vibration.

D.4. Filling, venting or valving in equipment shall be done slowly to prevent exceeding the makeup capability to the Surge Tank.

D.5. Discharge check valves may leak by causing backflow which results in pump/motor reverse rotation.

D.6. During low heat load periods, heat exchanger outlet valves should remain full open to ensure adequate flow in the event the heat load rises.

D.7. Due to the inherent design of the temperature control valve, 2CCS-TV104, approximately 1000 GPM is bypassed around the CCS Heat Exchangers during heat exchanger operations.

D.8. During high lake temperature conditions, SWP flow to the CCS heat exchangers may have to be reduced to support operability of the HVK chillers. See N2-OP-11, High Service Water Temperature Operations.

F.3. Starting an Idle Pump [SOP]

NOTE

- CCS system flow of greater than 8000 GPM requires more than one heat exchanger in service.
- Actions in this Subsection are performed at 2CEC*PNL601 unless otherwise specified.

F.3.1. **IF** time permits,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **DISPATCH** Operator to pump to perform prestart inspection **AND** observe pump during start ..... ☑.

b. For pump to be started,
CLOSE 2CCS-V303A(B,C), TBCLC PUMP 1A(B,C) STOP CHECK.. ☑ OPE

F.3.2. **START** 2CCS-P1A(B,C), PMP 1A(B,C), by placing control switch to Normal-After-START (red flagged). _____

F.3.3. **SLOWLY OPEN** 2CCS-V303A(B,C). _____

F.3.4. **CONFIRM** normal operating indications per Subsection F.1. _____

F.4. **Secure an Operating Pump**

NOTE

- Actions in this Subsection are performed at 2CEC*PNL601 unless otherwise specified.
- CCS Pump runout is 8,500 GPM. If system flow is greater than 8,000 GPM for one pump operations **OR** 16,000 GPM for two pump operations, then starting an additional pump increases the probability of damaging tube vibrations in the CCS heat exchangers. Three CCS pump operation shall be avoided unless shifting of pumps is required.

F.4.1. **CONFIRM** CCS pump is **not** required for CCS System flow as follows, _____

Computer Point CCSFA01, TBCLCW PMP SUCTION FLW	Number of Running Pumps
1,000 - 8,200 GPM	1
8,200 - 16,400 GPM	2

NOTE

2CCS-V303A(B,C) is located at 2CCS-P1A(B,C) on TB Elev 250 Southwest.

CAUTION

The time the CCS Pump is operating with the discharge valve closed shall be minimized due to no minimum flow protection.

F.4.2. **CLOSE** 2CCS-V303A(B,C), TBCLC PUMP 1A(B,C) STOP CHECK. _____

F.4.3. **SECURE** 2CCS-P1A(B,C), PMP 1A(B,C), by placing control switch in Normal-After-STOP (green flagged). _____

F.4.4. **IF** required,
THEN PLACE 2CCS-P1A(B,C) control switch in PULL-TO-LOCK,
otherwise mark this step N/A. _____

F.4.5. **CONFIRM** normal operating indications for the running CCS Pump(s) per Subsection F.1. _____

F.4.6. **IF** 2CCS-P1A(B,C) is to remain in standby,
THEN OPEN 2CCS-V303A(B,C),
OTHERWISE MARK this step N/A. _____

F.5. **Filling and Venting an Idle Heat Exchanger with System Operating**

NOTE

- Actions in this Subsection are performed on TB 250 Southwest unless otherwise specified.
- This Subsection assumes that 2CCS-E1A(B,C) is isolated, drained, **AND** the tagout is ready to be removed.

F.5.1. **VERIFY** the following valves closed:

NOTE

2CCS-V88(V87, V89) is located on top of the heat exchanger inlet line at Elev 262.

- 2CCS-V88(V87, V89), CCS HX1A(B,C) INLET LINE VENT..... ☐
- 2CCS-V109A(B,C), CCS HX1A(B,C) INLET LINE TEST CONN..... ☐
- 2CCS-V11A(B,C), CCS HX1A(B,C) DRAIN..... ☐
- 2CCS-V103A(B,C), CCS HX1A(B,C) VENT..... ☐
- 2CCS-V110A(B,C), TBCLC HEAT EXCHANGER 1A(B,C) OUTLET TEST CONNECTION..... ☐

NOTE

2GMC-V21, V22, and V23 are located on the GMC Skid.

- 2GMC-V21, MAKEUP WTR SUPPLY TO DEIONIZER STOP CHECK. ☐
- 2GMC-V22, DEIONIZER MAKEUP WTR FLUSH CONN..... ☐
- 2GMC-V23, MAKEUP WTR INLET TO DEIONIZER ISOLATION..... ☐ _____

NOTE

A two inch adaptor is required to connect a hose to 2GMC-V22.

F.5.2. **CONNECT** fill hose from 2GMC-V22 to 2CCS-V110A(B,C). _____

F.5.3. **CONNECT** vent hose to 2CCS-V88(V87, V89)
AND ROUTE to appropriate drain. _____

F. NORMAL OPERATIONS

F.1. Normal Operating Indications [SOP]

NOTE

Indications used in this Subsection are located on 2CEC*PNL601 unless otherwise specified.

F.1.1. **VERIFY** system normal operating conditions are as follows:

- Pump discharge pressure is 100 to 125 (120 to 135 psig when three pumps running) psig as indicated on 2CCS-PI149, TBCLC PMPS DISCH HDR PRESS meter. ..... ☐
- 2CCS-P1A(B,C) current is less than 58.5 amps as indicated on ammeter AM-2CCSA51 (B51, C51), CCS-P1A(B,C) CURRENT..... ☐
- 2CCS-TIK104, TBCLC HEAT EXCHANGER TEMP CONTROLLER, is in Auto **AND** controlling temperature at approximately 85 °F. ..... ☐

NOTE

CCS pump runout is 8,500 GPM. If system flow is greater than 8,000 GPM for one pump operations **OR** 16,000 GPM for two pump operations, then starting an additional pump increases the probability of damaging tube vibrations in the CCS heat exchangers. Three CCS pump operation shall be avoided unless shifting of pumps is required.

N/A

- CCS System flow **AND** number of running CCS Pumps is consistent with the following:..... ☑ N/A

Computer Point CCSFA01, TBCLCW PMP SUCTION FLW	Number of Running Pumps
1,000 - 8,200 GPM	1
8,200 - 16,400 GPM	2

F.1.1 (Continued)

NOTE

The following indications are located by the CCS Pumps on TB Elev 250 Southwest.

- Discharge pressure is 100 to 125 (120 to 135 psig WHEN three pumps running) psig as indicated on 2CCS-PI1A(B,C). ..... ☐
- Suction pressure is stable as indicated on 2CCS-PI16A(B,C). ..... ☐
- Packing gland leakoff is maintaining gland cool enough to maintain touch approximately 5 seconds. ..... ☐

NOTE

For 2CCS-P1A(B,C) leakage less than a constant drip could cause packing to overheat.

- For 2CCS-P1A(B,C) packing gland leakoff is reduced to a constant drip..... ☐
- Operating sound is normal. ..... ☐

NOTE

2CCS-LI107 is located by 2CCS-TK1, Surge and Makeup Tank, on Turbine Building Elev 326 Southwest side.

- 2CCS-TK1 level is 55" to 70" as indicated on 2CCS-LI107. ..... ☐ _____

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-2
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 65% power with a plant shutdown in progress. APRM 1 is bypassed due to erratic indication. 2MSS*PSV129 is inoperable due to a failed 'C' solenoid.		
Turnover:	Place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from 48 to 00.		
Critical Tasks:	CT-1: Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5: • Rapidly lowering RPV water level • Tripping RCS pumps • Injecting boron The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram. CT-2: Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5. Control rod insertion must be completed within one hour of the start of the failure to scram.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Place Heater Drain Pump in Recirculation Mode N2-OP-8
2	N/A	R – ATC, SRO	Insert Control Rods N2-OP-96
3	RD11	I – ATC, SRO TS – SRO	Loss of Control Rod Position Indication N2-OP-96, Technical Specifications
4	CW:RA: MT:I CW16	C – BOP, SRO MC – BOP	CCP Pump Trips, Standby Pump Fail to Automatically Start N2-SOP-13
5	NM19	I – ATC, SRO TS – SRO	RBM Fails Inoperable N2-OP-92, Technical Specifications
6	MC01	C – All	Loss of Main Condenser Vacuum N2-SOP-9, N2-SOP-101C, N2-EOP-RPV
7	RP03 RP12	M – All MC – All	Electrical ATWS N2-EOP-C5
8	RP12 SL01	C – BOP, SRO	SLC Fails to Start on RRCS, Delayed SLC Pump Trips N2-EOP-C5

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-2** Op-Test No.: **2023-2**

1. Malfunctions after EOP entry (1-2) **Event 8**	1	
2. Abnormal events (2-4) **Events 4 & 6**	2	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV**	1	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-C5**	1	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 2** Revision: **0.0**

Title: **November 2023 NRC Scenario 2**

	Signature / Printed Name		Date
Developed By	Signature on File	/ Mike Alexander	10/27/23
Validated By	N/A	Mecki Ghabarou	7/11/23
	N/A	/ Dan Delaney	7/11/23
	N/A	/ Mike Carno	7/11/23
Facility Reviewer	Signature on File	/ John Toothaker	10/27/23

References

1. N2-OP-8, Feedwater Heaters and Extraction Steam System
2. N2-OP-101C, Plant Shutdown
3. N2-SOP-13, Loss or Degraded CCP System
4. N2-OP-92, Neutron Monitoring
5. N2-SOP-9, Loss of Condenser Vacuum
6. N2-SOP-101C, Reactor Scram
7. N2-EOP-RPV (1-3), RPV Control - Flowchart
8. N2-EOP-C5 (1-3), RPV Control – ATWS - Flowchart
9. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
10. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
11. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
12. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 65% power during a shutdown with APRM 1 bypassed and 2MSS*PSV129 is inoperable.

   b. Event 1 is the normal evolution performed by the BOP operator to place the “C” heater drain pump in recirculation mode per N2-OP-8.

   c. Event 2 is a reactivity evolution. The ATC operator will continue the shutdown by inserting control rods per the shutdown sequence.

   d. Event 3 occurs when control rod position indication is lost to a control rod. The ATC operator will insert the control rod one notch and control rod indication will be restored. The SRO will evaluate Technical Specifications for the time the control rod position indication was lost.

   e. Event 4 occurs when a running CCP pump trips and the standby pump fails to automatically start. The crew will execute N2-SOP-13 and start the standby pump.

   f. Event 5 occurs when RBM A fails inop. The crew will bypass RBM A. The SRO will evaluate Technical Specifications.

   g. Event 6 occurs when Main Condenser vacuum begins to degrade. The crew will execute N2-SOP-09 and lower Reactor power. Vacuum will continue to degrade. The crew will attempt to scram the Reactor.

   h. Events 7-8 begin immediately following the scram attempt when control rods fail to insert. The crew will enter N2-EOP-RPV (1-3) and transition to N2-EOP-C5 (1-3). The crew will take actions to lower Reactor power *(critical task)* and insert control rods *(critical task)*. SLC pumps fail to automatically start on RRCS initiation. This will require

the crew to take manual actions to start SLC. SLC pumps will trip after 2 minutes of operation, limiting the boron injection and resulting power suppression.

2. Termination Criteria

   a. Reactor power downscale on APRMs.

   b. Control rod insertion completed.

   c. Reactor level being controlled in assigned band.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because, without operator action to lower Reactor power, the risk of damage to fission product barriers is significantly elevated.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that the Reactor has failed to scram and that Reactor power remains above 4%. N2-EOP-RPV (1-3) and N2-EOP-C5 (1-3) provide direction to lower Reactor power.*
Success Path:	*The crew will manipulate injection sources to lower Reactor water level, RCS pump control switches, and/or SLS controls.*
Performance Feedback:	*Reactor water level and injection source flow rates will provide performance feedback regarding the success of rapidly lowering RPV water level. RCS pump and SLS pump indications and flow rates will provide performance feedback regarding the success of operating those systems. Overall, multiple indications will provide performance feedback on the effect of these actions on Reactor power.*
Measurable Performance Standard:	*Manipulating injection sources, RCS pump control switches, and/or SLS controls will provide the evaluation team with observable actions.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

   a. CT-1.0, Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5 (1-3):

      - Rapidly lowering RPV water level
      - Tripping RCS pumps
      - Injecting boron

      The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because, without operator action to insert control rods, the Reactor will not achieve a stable, long-term shutdown condition.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence of the failure of control rods to insert on the scram.*
Success Path:	*The crew will utilize one or more methods in N2-EOP-6.14 to insert control rods. For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.*
Performance Feedback:	*Control rod position indication and Reactor power indication will provide performance feedback regarding the success of control rod insertion.*
Measurable Performance Standard:	*Manipulation of CRD/RMCS components and/or direction to field operators will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

b. CT-2.0, Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5 (1-3). Control rod insertion must be completed within one hour of the start of the failure to scram.

Note: For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-AT-01, high power ATWS (greater than APRM downscale); heat addition to suppression pool or torus requires entry into level power control; RPV level controlled below feedwater spargers; RPV blowdown not required

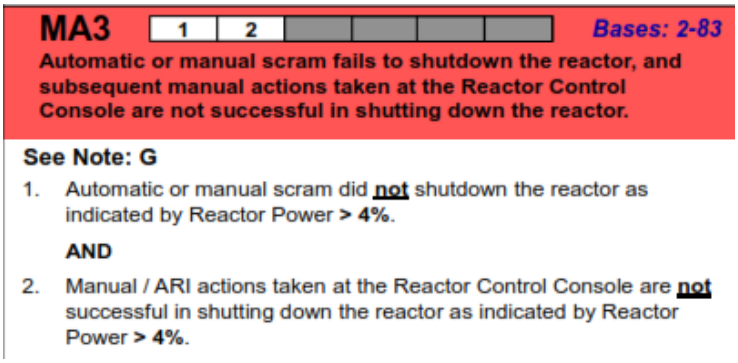
6. Technical Specifications (Applicable actions for initial conditions only)

   a. None

7. EAL Classification

   a. Alert, EAL MA3:

1)



B. Initial Conditions

1. IC Number

   a. IC-023 or equivalent (IC-201 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**AD02U**, Relief Valve Failure PSV129, Final = true	**Inserted**
2)	**RD11-38-39**, Control Rod Failure – RPIS (Any), Final = true	**TRG1**
3)	**CW16C**, CCP-P1C Fails to Auto Start, Final = true	**Inserted**
4)	**CW:RA:MT:I**, RBCLC Pump 1A Motor Failure, Final = short	**TRG2**
5)	**NM19A**, Rod Block Monitor Failure - Inop (A), Final = true	**TRG3**
6)	**MC01**, MAIN CONDENSER AIR INLEAKGE, Final = 50, Ramp = 10 minutes	**TRG4**
7)	**RP03**, Reactor Protection System Failure to Scram, Final = true	**Inserted**
8)	**SL01A**, Standby Liquid Control Pump Trip (P1A), Final = true, Delay = 5:30 minutes	**TRG10**
9)	**SL01B**, Standby Liquid Control Pump Trip (P1B), Final = true, Delay = 5:30 minutes	**TRG11**
10)	**RP12A**, RRCS Division Failure (Div I), Final = true	**Inserted**

11) **RP12B**, RRCS Division Failure (Div II), Final = true — **Inserted**

12) **RP13A**, RRCS Inadvertent Alternate Rod Insertion (Div I), Final = true — **TRG25**

13) **RP13B**, RRCS Inadvertent Alternate Rod Insertion (Div II), Final = true — **TRG26**

b. Remotes

1) **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, Final = 35, Ramp = 45 seconds — **TRG30**

2) **MS06A,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148A), Final = defeated — **TRG20**

3) **MS06B,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148B), Final = defeated — **TRG20**

4) **MS06C,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148C), Final = defeated — **TRG20**

5) **MS06D,** DEFEAT LEVEL ONE ISOLATION OF MSIVS (JUMPER K148D), Final = defeated — **TRG20**

c. Overrides

1) **DO-1225**, OFF SAFETY/RELIEF VALVE PSV 129 GREEN, Final = off — **Inserted**

2) **DO-1226**, OFF SAFETY/RELIEF VALVE PSV 129 RED, Final = off — **Inserted**

3) **DI-3963**, Alt_A RPS Power Selector Switch, Final = off — **Inserted**

4) **DI-3964**, Alt_A RPS Power Selector Switch, Final = off — **Inserted**

d. Annunciators

1) None

e. Event Triggers

Event #	Event Action	Command
1	zdrdserd(125)==1 (Activates when control rod 38-39 is selected)	None

Event #	Event Action	Command
10	hzlslcspump1a(2)==1 (Activates when SLC pump 1A red light energizes)	None
11	hzlslcspump1b(2)==1 (Activates when SLC pump 1B red light energizes)	None
15	rdvposb(125)==736 (Activates when control rod 38-39 reaches position 46)	dmf rd11

f. Equipment Out of Service

1) APRM 1 bypassed

2) 2MSS*PSV129 inop

g. Support Documentation

1) Provide N2-OP-8 section G.1.0, completed up to step G.1.6

2) Provide ReMA for plant shutdown

3) Provide N2-OP-101C completed up to step G.1.9.

4) Provide N2-OP-101D section G completed up through step G.1.19

5) Provide N2-OP-96 section F.3.0

h. Miscellaneous

1) Place clearance reference tag on the 2MSS*PSV129 control switch

2) Place clearance reference tag on APRM bypass switch for APRM 1

C. Shift Turnover Information

1. Reactor Power: 65%
2. Rodline: Below 100%
3. Technical Specification LCOs in effect:

a. None

NOTES

4. Significant Problems / Abnormalities / Equipment Out of Service:

   a. Plant shutdown is in progress.

   b. APRM 1 is bypassed.

   c. 2MSS*PSV129 is inoperable.

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Place the “C” heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Perform through G.1.10 and then hold that evolution.

   b. Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from 48 to 00.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 65% during a plant shutdown
- N2-OP-101C is in progress up to step G.1.9
- N2-OP-101D is in progress up to step G.1.19
- APRM 1 is bypassed due to erratic indication
- 2MSS*PSV129 is inoperable due to a failed 'C' solenoid

PART III: Remarks/Planned Evolutions:

- Place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Perform through G.1.10 and then hold that evolution. An operator is standing by at 2CES-IPNL204.
- Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from position 48 to 00.

Event #1 – Place Heater Drain Pump in Recirculation Mode

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power • Verifiable Actions - Place heater drain pump in recirculation mode • Final (expected) operating result - Heater drain pump in recirculation mode

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1
Role Play As EO directed to lower tape setting (N2-OP-8, G.1.8), immediately **insert** the following **remote function**: **TRG30** **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, Final = 35, Ramp = 45 seconds Once the remote setpoint reaches 35, contact the control room and report that the tape setting is at 35% and 2HDL-LV24C is closed. **Role Play** As EO, observe the position of LV24C on the FW06, Extraction Steam II display and inform the control room when it begins to open and control level. **Role Play** As EO directed to lower tape setting (N2-OP-8, G.1.10), immediately **modify** the following **remote function** as follows: **FW13C**, 2HDL-LIC24C REMOTE SETPOINT, FV=10, RT=1:00 Once the remote setpoint reaches 10, contact the control room and report that the tape setting is at 10%.	**BOP** • Acknowledges direction to place the "C" heater drain pump in recirculation mode per N2-OP-8 section G.1 • At 2CEC*PNL851, slowly raise 2HDL-LV4C, FD WTR HTR 2CNM-E4C WTR LEVEL CONTROL setpoint to 47% • Contacts EO at 2CES-IPNL204 and directs them to slowly lower 2HDL-LV24C, 4TH PT HTR E4C HIGH DR tape setting in Auto UNTIL 2HDL-LV24C begins to open OR tape setting is at 35% • Places 2HDL-LV4C in MAN AND slowly closes the valve WHILE verifying the following: ◦ At 2CES-IPNL204, 2HDL-LV24C, 4TH PT HTR E4C HIGH DR, opens to control heater level. ◦ 2HDL-FV35C, HTR DRAIN P1C RECIRC FV POSN opens AND maintains a minimum system flow of approximately 1400 gpm • Contacts EO and directs them to slowly lower tape setting for 2HDL-LV24C,4th PT HTR E4C controller set to 10%

Event Termination Criteria	• Heater Drain Pump C is in recirculation mode

Event #2 – Insert Control Rods

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power • Verifiable Actions - Insert control rods • Final (expected) operating result - Sufficient control rods inserted for adequate evaluation

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to continue inserting control rods per the shutdown sequence • Provides oversight of reactivity manipulation
Note: The next three control rods to be moved are 38-23, 38-39, and 22-23, going from position 48 to 00. **Note:** The next event (loss of rod position indication) will automatically begin once the 2nd control rod is selected (38-39).	**ATC** • Acknowledges direction to continue inserting control rods per the shutdown sequence • Inserts control rods per N2-OP-96 and the shutdown sequence: ◦ VERIFY there are no rod insert blocks ◦ VERIFY correct rod has been selected per sequence being used ◦ DEPRESS AND HOLD INSERT push button AND OBSERVE INSERT light remains on as long as push button is held down ◦ OBSERVE selected rod moves continuously inward on 4-Rod Display ◦ RELEASE push button AND ALLOW rod to settle into desired notch AND OBSERVE proper cycling of INSERT AND SETTLE lights ◦ MONITOR appropriate nuclear instruments for proper response

Instructor Actions / Plant Response	Operator Actions
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• A sufficient number of control rods have been inserted as determined by the Lead Evaluator

Event #3 – Loss of Control Rod Position Indication

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~65% power - Control rod 38-39 has just been selected • Verifiable Actions - Insert control rod 38-39 to re-establish control rod position indication - Technical Specification 3.1.3 evaluated • Final (expected) operating result - Control rod 38-39 position indication restored

Instructor Actions / Plant Response	Operator Actions
When control rod 38-39 is selected, verify the following **malfunction automatically inserts**: **TRG1 RD11-38-39, Control Rod Failure – RPIS (Any), Final = True** *Control rod 38-39 position indication is lost*	
The following annunciator alarms: • *603443, Control Rod Drift*	**ATC** • Recognize / report control rod 38-39 has lost position indication
Note: During the time that rod position indication is not available (XX indication), TS 3.1.3 Condition C would be applicable. Once valid position indication is achieved, TS 3.1.3 Condition C would be exited. **Spec** | **Condition** | **Applicable Actions** 3.1.3 | C | C.1 & C.2 **Action** | **Description** C.1 | Fully insert inoperable control rod. (3 Hours) C.2 | Disarm the associated CRD. (4 Hours)	**SRO** • Acknowledges report • Directs ATC to execute N2-OP-96 section H.2.0 • Refers to TS 3.1.3 and determines Condition C applies during time position indication is not available • May contact Reactor Engineering and/or plant management

Instructor Actions / Plant Response	Operator Actions
Role Play: If asked to record lit LEDs in 2CEC-PNL615, report that another operator will perform that action. **Role Play:** Once control rod position indication has been restored, if asked as Reactor Engineering and/or plant management about desire to continue control rod insertions, report that it is desired to continue control rod insertions per the shutdown sequence.	**ATC** • Executes N2-OP-96 section H.2.0 • Coordinates with BOP to record lit LEDs in 2CEC-PNL615 • Notifies crew of need to suspend all control rod motion • Determines only one control rod is affected • Inserts control rod 38-39 one notch • Recognizes / reports control rod 38-39 has position indication at position 46 • Continues with control rod insertions per the shutdown sequence, as directed.

Event Termination Criteria	• Control rod 38-39 position indication has been restored • A sufficient number of control rods have been inserted as determined by the Lead Evaluator • Technical Specifications have been addressed

Event #4 – CCP Pump Trips, Standby Pump Fail to Automatically Start

Event Information	• Presumed or required initial plant operating conditions - Plant on-line - CCP system aligned for normal operation with 2CCP-P1A and P1B running, P1C in standby • Verifiable Actions - 2CCP-P1C is started • Final (expected) operating result - CCP system operating with 2CCP-P1B and P1C running

Instructor Actions / Plant Response	Operator Actions
Note: It is desired to wait until no control rods are being moved to insert this event. Wait until operators are signing off ReMA for the rod just moved. **Booth Operator:** When directed by the lead evaluator, **insert** the following **malfunction**: **TRG2** **CW:RA:MT:I**, RBCLC Pump 1A Motor Failure, Final = short The following malfunction is preset: **CW16C**, CCP-P1C Fails to Auto Start, Final = true *The plant responds as follows:* • *2CCP-P1A trips* • *2CCP-P1C fails to automatically start*	

Instructor Actions / Plant Response	Operator Actions
The following annunciators alarm: • *601252, RBCLC Pump 1A/1B/1C Auto Trip/Fail to Start* • *601253, RBCLC Pump 1A/1B/1C Motor Overload* • *601246, Reactor Bldg Closed Loop Cooling Sys Trouble* *The following annunciators may alarm and clear depending upon the time it takes the operators to respond and to manually start the standby RBCLC pump:* • *601254, RBCLC Pump 1A/1B/1C Disch Press Low* • *601255, RBCLCW to RHR Pump 1A/1B/!C Pressure Low* • *601257, RBCLCW to Reactor Recirc Pump A Clrs Pres Low* • *601258, RBCLCW to Reactor Recirc Pump B Clrs Pres Low*	**BOP** • Silences, acknowledges and reports annunciators • Reports trip of 2CCP-P1A • Reports failure of 2CCP-P1C to automatically start
Note: Crew may not enter N2-SOP-13 and simply start 2CCP-P1C due to making an action occur that failed.	**SRO** • Acknowledges reports • Directs entry into N2-SOP-13
Role Play: When contacted by the control room to investigate local indications of pump trip, wait 3 minutes and then report: • If dispatched to pump, no visible problems, however a moderate acrid odor is noticeable in the area of the 2CCP-P1A motor • If dispatched to the breaker, no visible problems, breaker (2NNS-SWG012-5) is tripped, the 50/51 relay is flagged and the 86 Relay on the breaker is rolled	**BOP** • Enters and executes N2-SOP-13 • Performs N2-SOP-13 actions: ◦ Evaluates second set of If/Then statements ◦ Determines that a standby CCP pump failed to auto start ◦ Starts 2CCP-P1C ◦ Reports to SRO that the standby CCP Pump (CCP-P1C) has been manually started • Continues with the actions of N2-SOP-13: ◦ Evaluates all 3 legs of N2-SOP-13 - Determines at least 1 main AND booster pump is operating

Instructor Actions / Plant Response	Operator Actions
	- Determines that cooling capacity is no longer degraded - Determines that no further actions are required to stabilize the CCP system ◦ Verifies CCP cooling is restored and that system parameters have stabilized ◦ Monitors loads cooled by RBCLC ◦ Dispatches an operator to investigate locally (breaker and pump) ◦ Verifies that the system is operating per N2-OP-13, F.1.0 • Perform ARP actions for tripped CCP pump ◦ Place 2CCP-P1A in Pull-to-Lock or Stop

Event Termination Criteria	• 2CCP-P1C started

Event #5 – RBM Fails Inoperable

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Bypass RBM A • Final (expected) operating result - Operating with RBM A bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **remote**: **TRG3** **NM19A**, Rod Block Monitor Failure - Inop (A), Final = true	
The following annunciators alarm immediately after event initiation: • *603204 RBM UPSCALE/ INOPERABLE* • *603442 CONTROL ROD OUT BLOCK*	**ATC** • Silences, acknowledges and reports annunciators • Recognizes / reports inop failure of RBM A
Note: Although RBM A is inoperable, it is only required to be operable when no peripheral (edge) rod is selected. If an edge rod is not already selected, the SRO may direct the RO to select an edge rod to exit the applicability requirements of TS 3.3.2.1. To continue the plant shutdown, it would be required to select a non-edge rod at some point.	**SRO** • Acknowledges reports • Directs execution of ARPs • Directs bypassing RBM A per N2-OP-92 • References TS 3.3.2.1 Condition A and determines a 24 hour LCO applies • May direct selecting an edge rod to change the applicability (If not already selected)

Spec	Condition	Applicable Actions
3.3.2.1	A	A.1
Action	**Description**	
A.1	Restore RBM channel to operable status (24 Hours)	

Instructor Actions / Plant Response	Operator Actions
Role Play: As booth when contacted for indications on RBM A Interface Module Top, provide the following information: LED indications on A3 are as follows: • PWR 1-Lit • PWR 2-Not Lit • PWR A-Not Lit • PWR B-Not Lit LED indications on the cards: • A4-Not Lit • A5- Not Lit • A6- Not Lit • A7-Lit • A8- Not Lit • A9- Not Lit • A10- Not Lit **Note:** If the SRO decides to not bypass RBM A, at the Lead Evaluator's discretion, call up the SRO as the SRO and direct bypassing RBM A. **Role Play:** When contacted to provide indications at P608, inform the operator that BYPASS MANUAL is displayed in inverse video header for RBM A	**ATC** • Executes ARPs • Determines RBM A is inop • Completes N2-OP-92 Attachment 4 • Informs SRO of the results of Attachment 4 • Informs SRO that the RBM may be bypassed per N2-OP-92 • Bypasses RBM A per N2-OP-92 section H.2.0 • Determines no other RBMs are bypassed • Places the RBM bypass joystick to the 'A' position • Verifies RBM A BYPASS light is lit at P603 • Verifies BYPASS MANUAL is displayed in inverse video in the header for RBM A at H13-P608

Event Termination Criteria	• RBM A bypassed • Technical Specifications addressed

Event #6: Loss of Main Condenser Vacuum

Event Information	• Presumed or required initial plant operating conditions - Plant on-line • Verifiable Actions - Lower Reactor power - Attempt manual scram • Final (expected) operating result - Manual scram attempted

Instructor Actions / Plant Response	Operator Actions
When directed by lead evaluator **insert** the following **malfunction**: **TRG4** **MC01**, MAIN CONDENSER AIR INLEAKGE, Final = 50, Ramp = 10 minutes	
• *Offgas flows rise* • *Condenser vacuum begins to lower* • *MWe lowers* • *MWth lowers* *The following annunciator alarms ~5 minutes after event initiation:* • *851358, TURBINE CNSR A/B/C VACUUM LOW* • *851359, LOAD DEPENDENT CNSR VACUUM LOW* **Note:** *Offgas flow rate will be high >65 scfm after 2 minutes. The Main Turbine will trip on low vacuum about 6 minutes after the start of the event.*	**Crew** • Recognizes / reports degrading Main Condenser vacuum

Instructor Actions / Plant Response	Operator Actions
Note: Once the manual Reactor scram is attempted, further actions are described in the next event.	**SRO** • Acknowledges report of degrading condenser vacuum • Directs BOP to enter N2-SOP-09 • May direct ATC to lower power using rods • Directs manual Reactor scram • Directs entry into N2-SOP-101C
	ATC • Lowers Reactor power using Recirc flow • May insert cram rods • Acknowledges SRO direction to scram the reactor • Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603. • Places the Reactor Mode Switch in the SHUTDOWN position • Recognizes / reports failure of RPS to insert rods
	BOP • Acknowledges direction to enter N2-SOP-09 • Determines ATC is lowering power • Verifies proper operation of the following: ◦ SJAE ◦ Offgas ◦ Circulating Water • Determines condenser vacuum cannot be maintained above 22.1”Hg • Informs the SRO that N2-SOP-09 directs tripping the turbine per N2-SOP-21

Event Termination Criteria	• Manual Reactor scram attempted

Event #7 and 8 – Electrical ATWS; SLC Fails to Start on RRCS, Delayed SLC Pump Trips

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power - Manual Reactor scram attempted • Verifiable Actions - Reactor power lowered - Control rods inserted • Final (expected) operating result - Reactor power <4% - Control rod insertion completed • Mitigation Strategy code (if applicable) - DMS-AT-01, high power ATWS (greater than APRM downscale); heat addition to suppression pool or torus requires entry into level power control; RPV level controlled below feedwater spargers; RPV blowdown not required
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a failure to scram with Reactor power above 4%, the crew will lower Reactor power by one or more of the following methods, in accordance with N2-EOP-C5 (1-3): ◦ Rapidly lowering RPV water level ◦ Tripping RCS pumps ◦ Injecting boron The Reactor power reduction must be initiated within ten minutes of the start of the failure to scram. 2. Given a failure to scram, the crew will initiate control rod insertion, in accordance with N2-EOP-C5 (1-3). Control rod insertion must be completed within one hour of the start of the failure to scram. **Note:** For the given failure to scram, it is expected that control rods will be inserted by venting the scram air header.

Instructor Actions / Plant Response	Operator Actions
The following malfunctions are preset to cause the failure to scram: **RP03**, Reactor Protection System Failure to Scram, Final = true **RP12A**, RRCS Division Failure (Div I), Final = true **RP12B**, RRCS Division Failure (Div II), Final = true	

Instructor Actions / Plant Response	Operator Actions
The following malfunctions will automatically insert on TRG10 or TRG11 to cause the trip of the SLC pumps: **TRG10** **SL01A**, Standby Liquid Control Pump Trip (P1A), Final = true, Delay = 5:30 minutes **TRG11** **SL01B**, Standby Liquid Control Pump Trip (P1B), Final = true, Delay = 5:30 minutes	
	Crew • Recognizes / reports failure to scram
Role Play: If requested prevent the main turbine trip from RCIC (N2-EOP-6.2) **manually insert** the following **remote**:	**SRO** • Enters & executes N2-EOP-RPV (1-3) ◦ Informs crew of N2-EOP-RPV (1-3) entry ◦ Determines the following: ▪ Scram was initiated ▪ Failure to scram has occurred • Exits N2-EOP-RPV (1-3) and enters & executes N2-EOP-C5 (1-3) • Informs crew that reactor has failed to scram, exiting N2-EOP-RPV (1-3), and entering N2-EOP-C5 (1-3) • Determines Reactor power is >4% • Directs rapidly lowering RPV water level (Detail J): ◦ Directs inhibiting ADS ◦ **Directs terminating and preventing all injection into the RPV except boron, CRD, and RCIC (CT-1.0)** ◦ Directs maintaining RPV water level between -39” and 100” using Condensate/Feedwater, HPCS, RCIC, SLS, and/or CRD ◦ **Directs boron injection using SLS (CT-1.0)** ◦ May direct preventing Main Turbine trip from RCIC (N2-EOP-6.2) ◦ May direct bypassing the following isolations per N2-EOP-6.10

Instructor Actions / Plant Response	Operator Actions
RC10, Defeat RCIC/MT Trip Intlk (EOP-6 ATT 2), Final = Defeated **Note:** Main Condenser vacuum will stabilize above 8.5" Hgv, such that the Main Condenser remains available. If the crew has concern over Main Condenser vacuum, they may eventually close MSIVs and transfer Reactor pressure control to RCIC/SRVs.	▪ MSIV low water level ▪ Offgas high radiation • Power Leg ◦ Verifies Mode Switch in SHUTDOWN ◦ Directs initiating RRCS (N2-EOP-6.13) ○ Acknowledges RRCS fails to start SLC pumps ○ Directs manual start of SLC pump(s) ◦ Directs reducing Recirc flow to minimum ◦ **Directs tripping Recirc pumps (CT-1.0)** ◦ **Directs inserting control rods per N2-EOP-6.14 (CT-2.0)** • Level Leg: ◦ Directs verification of auto actions per N2-EOP-6.1 for isolations, ECCS, and DGs ◦ Determines the Main Condenser is available ◦ May direct bypassing the following isolations per N2-EOP-6.10 ○ MSIV low water level ○ Offgas high radiation ◦ Directs maintaining RPV water level between -39" and 100" using Condensate/Feedwater, HPCS, RCIC, SLS, and/or CRD • Pressure Leg: ◦ Determines no SRVs are cycling ◦ Directs RPV pressure band of 800 - 1000 psig using EHC ◦ Directs restoring pneumatics to the drywell ◦ Assigns critical parameter for monitoring / reporting the following: ○ Suppression pool temperature (110°F is a key value for BIIT) • May enter N2-EOP-PC (1-3) due to high Suppression Pool temperature ◦ May direct initiation of Suppression Pool Cooling ◦ May direct start of 5th Service Water pump

Instructor Actions / Plant Response	Operator Actions
	ATC • Acknowledges direction to initiate RRCS per N2-EOP-6.13 • Manually initiate RRCS as follows: ◦ At PNL603, arms and depresses the Division I and II A and B RRCS MANUAL INITAITION SWITCHES ◦ Determines the Division I and II ARI initiation lights are lit ◦ Determines SLC did not initiate after 98 seconds and WCS did not isolate as expected ◦ Informs SRO that RRCS failed to start SLC pumps ◦ **Manually places the keylock switches for SLS P1A and P1B in start (CT-1.0)** ◦ Verifies other SLC pump starts and WCS isolates • Informs the SRO that one SLC pump is running and WCS is isolated • May recognize / report trip of SLC pumps (later) • **Terminates and prevents panel 603 (CT-1.0):** ◦ VERIFIES controller 2FWS-HIC1600 is in manual with 0% output ◦ VERIFIES the following controllers are in manual with 0% output: ○ 2FWS-HIC1010A ○ 2FWS-HIC1010B ○ 2FWS-HIC1010C ○ 2FWS-LIC1055A ○ 2FWS-LIC1055B ○ 2CNM-LIK1137 • Informs the SRO that PNL603 is terminated and prevented. • **Allows RPV level to lower to <100 inches and then reestablishes feed flow and an RPV level band as directed (CT-1.0)** • **Trips Recirc pumps (CT-1.0)**

Instructor Actions / Plant Response	Operator Actions
Note: Multiple attempts to scram the Reactor electrically will fail. Control rods will insert when the scram air header is manually vented. Manual insertion of control rods using RMCS will also be possible. **Role Play:** When dispatched as EO to manipulate POWER SOURCE SELECTOR at 2CEC*PNL610, acknowledge direction, wait 2 minutes, then report task completion. **Role Play:** When dispatched as EO to vent scram air header using Div 1 & 2 ARI, acknowledge direction, wait at least 2 minutes and until Lead Examiner is ready for control rods to insert, then **insert TRG25 and TRG26** and report task completion **Role Play:** When dispatched as EO to de-energize scram solenoids with HCU SRI switches, acknowledge direction. **Role Play:** When dispatched as EO to pull fuses in 2CEC*PNL609 and 2CEC*PNL611, acknowledge direction.	• As time permits, verifies auto actions occurred as expected • **Inserts control rods per N2-EOP-6.14 (CT-2.0):** • Determines 8 white scram solenoid power lights are NOT off • Performs the following at 2CEC*PNL610: ◦ Positions POWER SOURCE SELECTOR switch to ALT A ◦ Positions POWER SOURCE SELECTOR switch to ALT B ◦ Positions POWER SOURCE SELECTOR switch to NORM • Determines 8 white scram solenoid power lights are NOT off • Performs concurrently: ◦ Determines blue SCRAM lights NOT lit on Full Core Display: ▪ **Dispatches EO to vent scram air header using Div 1 & 2 ARI** ◦ Continues at Arrow D to de-energize scram solenoids with HCU SRI switches: ◦ Determines scram discharge volume vent and drain valves are open ▪ Dispatches EO to pull fuses in 2CEC*PNL609 and 2CEC*PNL611

Instructor Actions / Plant Response	Operator Actions
Role Play: If directed to defeat the RCIC level 8 logic, wait 1 minute then **manually insert** the following **remote**: **RC02A**, RCIC Level 8 Trip Defeat:Withdraw Trip Units N693A,E, Final = Defeated **Role Play** If contacted to defeat MSIV low water level isolation per N2-EOP-6.10, **insert** the following **remotes**: **TRG20** **MS06A**, Defeat Level One Isolation of MSIVs (Jumper K148A), Final = Defeated **MS06B**, Defeat Level One Isolation of MSIVs (Jumper K148B), Final = Defeated **MS06C**, Defeat Level One Isolation of MSIVs (Jumper K148C), Final = Defeated **MS06D**, Defeat Level One Isolation of MSIVs (Jumper K148D), Final = Defeated Wait 1 minute and report that the MSIV low water level jumpers have been installed. **Role Play** If contacted as field operator to defeat the Offgas high radiation isolation per N2-OP-42 Attachment 6, wait 1 minute and **insert** the following **remote**: **OG03**, Off Gas High Radiation Isolation Defeated, Final = Defeated	**BOP** • Inhibits ADS: ◦ Places DIV I ADS AUTOMATIC INITIATION DISABLE switch to ON and verifies the associated white light is lit ◦ Places DIV II ADS AUTOMATIC INITIATION DISABLE switch to ON and verifies the associated white light is lit ◦ Verifies 601521 DIVISION I ADS AUTOMATIC INITIATION DISABLED is lit ◦ Verifies 601522 DIVISION II ADS AUTOMATIC INITIATION DISABLED is lit • Informs SRO that ADS is inhibited • **Terminates and prevents panel 601 (CT-1.0):** ◦ Places 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ Arms and depresses LPCI A/LPCS MANUAL INITIATION pushbutton ◦ Closes and overrides 2RHS*MOV24A, LPCI A INJECTION VLV and 2CSL*MOV104, PMP 1 INJECTION VLV ◦ Places 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK ◦ Arms and depresses LPCI B & C MANUAL INITIATION pushbutton ◦ Closes and overrides 2RHS*MOV24B, LPCI B INJECTION VLV and 2RHS*MOV24C, LPCI C INJECTION VLV • Informs the SRO that PNL601 is terminated and prevented. • If directed, prevents Main Turbine trip from RCIC ◦ Removes relay E51A-K102 in 2CEC*PNL613 ◦ Delivers relay to SRO

Instructor Actions / Plant Response	Operator Actions
Then report that the Offgas high radiation isolation has been defeated.	• Informs SRO that Main Turbine trip from RCIC is defeated • If directed, defeats MSIV low water level isolations at 2CEC*PNL609 (611) ◦ Installs the following EOP Jumpers: - #20 from relay B22H-K148A to jumper block (P609) - #18 from relay B22H-K148C to jumper block (P609) - #13 from relay B22H-K148B to jumper block (P611) - #12 from relay B22H-K148D to jumper block (P611) ◦ Verifies 2IAS*SOV166, LOCA OVERRIDE VLV switch in OVERRIDE (P851) ◦ Verifies open 2IAS*SOV166 (P851) ◦ IF required, defeats OFG system high radiation isolation AND restores OFG system per N2-OP-42, Attachment 6 - Dispatches field operator to lift lead at 2OFG-IPNL122 • Informs SRO that MSIV low water level isolation is defeated • If defeated, informs SRO that OFG high radiation isolation is defeated • Maintains RPV pressure 800 - 1000 psig using EHC or appropriate pressure band using SRV's if the MSIVs close. • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: - PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE - OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 - PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD - OPENS 2IAS*SOV164

Instructor Actions / Plant Response	Operator Actions
	- PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD - OPENS 2IAS*SOV165
Role Play: If dispatched as field operator to perform pre-start checks on 2SWP-P1A(B-E), wait 3 minutes then report that 2SWP-P1A(B-E) is ready for start and that all pre-start checks are completed. **Role Play:** If requested to place SWP radiation monitor RE23A in service, wait 2 minutes and **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, FINAL=On **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, FINAL=On Then report 2SWP*RE23A is in service. **Role Play:** If requested to place SWP radiation monitor RE23B in service, wait 2 minutes **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, FINAL=On **RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, FINAL=On Then report 2SWP*RE23B is in service.	**ATC/BOP** • If directed, initiates Suppression Pool Cooling per N2-OP-31 section F.4.0: ◦ Notifies Shift Manager to declare RHS A(B) LPCI mode inoperable ◦ Directs radiation protection department to start SWP radiation monitors 2SWP*RE23A(B) ◦ Opens 2SWP*MOV90A(B) ◦ May start 2SWP*P1E(F) ◦ Throttles open 2SWP*MOV33A(B) to establish 7400 gpm service water flow to RHR A(B) ◦ Verifies 2RHS*MOV24A(B) closed ◦ Verifies 2RHS*MOV4A(B) open ◦ Starts 2RHS*P1A(B) ◦ Throttles open 2RHS*FV38A(B) to establish 7450 gpm RHR A(B) total flow ◦ Verifies 2RHS*MOV4A(B) closed ◦ May throttle 2RHS*MOV8A(B) closed

Event Termination Criteria	• Reactor power downscale on APRMs. • Control rods insertion completed. • Reactor level being controlled in assigned band.



Simulator Scenario Handout Reference Packet

UNIT 2 Control Rod Move Sheet Form

FROM: ** **TO:** 00

Sequence ID: NM2C19-SD-1 Rev. 0

Page: 1

Concurrence to Proceed ALL				INSERT						N/A				N/A						
ACD QNE INIT	DJK SRO INIT	Step	Control Rod	From	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	To	Move Complete INIT RO	Move Complete INIT PEER	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	Comments
ACD	DJK	1	30-31	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous, FIRST ROD
ACD	DJK	2	30-47	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous
ACD	DJK	3	30-15	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous
ACD	DJK	4	46-31	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous
ACD	DJK	5	14-31	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous
ACD	DJK	6	22-39	**	00	ODE	VER	N/A	N/A							N/A	N/A	Continuous
ACD	DJK	7	38-23	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	8	38-39	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	9	22-23	**	00			N/A	N/A							N/A	N/A	Continuous

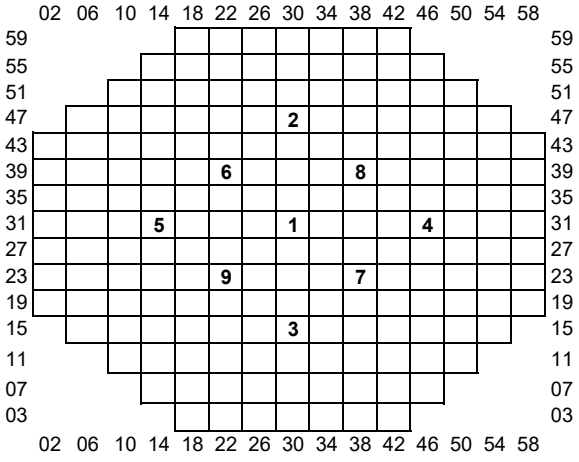
Prepared by: Paul Blejwas

Date: Today

Verified by: Kahn, Gerald J

Date: Today

NOTES
GE Rod Group 9



UNIT 2 Control Rod Move Sheet Form

FROM: ** **TO:** 00

Sequence ID: NM2C19-SD-1 Rev. 0

Page: 2

Concurrence to Proceed ALL				INSERT						N/A				N/A					
ACD QNE INIT	DJK SRO INIT	Step	Control Rod	From	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	To	Move Complete INIT RO	Move Complete INIT PEER	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	Comments
ACD	DJK	1	14-47	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	2	46-15	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	3	46-47	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	4	14-15	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	5	22-55	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	6	38-07	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	7	54-23	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	8	06-39	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	9	38-55	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	10	22-07	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	11	54-39	**	00			N/A	N/A							N/A	N/A	Continuous
ACD	DJK	12	06-23	**	00			N/A	N/A							N/A	N/A	Continuous

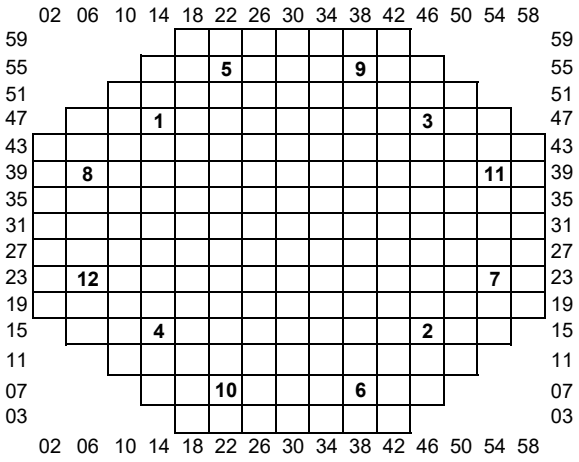
Prepared by: Paul Blejwas

Date: Today

Verified by: Kahn, Gerald J

Date: Today

NOTES
GE Rod Group 9





Simulator Scenario Handout Reference Packet

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 65% during a plant shutdown
- N2-OP-101C is in progress up to step G.1.9
- N2-OP-101D is in progress up to step G.1.19
- APRM 1 is bypassed due to erratic indication
- 2MSS*PSV129 is inoperable due to a failed 'C' solenoid

PART III: Remarks/Planned Evolutions:

- Place the “C” heater drain pump in recirculation mode per N2-OP-8 section G.1. The procedure is complete up to step G.1.6. Perform through G.1.10 and then hold that evolution. An operator is standing by at 2CES-IPNL204.
- Then, lower Reactor power by inserting control rods per the shutdown sequence. The next rod is 38-23 going from position 48 to 00.

Constellation

Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-8

FEEDWATER HEATERS AND EXTRACTION STEAM SYSTEMS

Revision 028

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-23-00777

Description: Added a P&L stating, “While an HDL pump is in service, oil discoloration is expected. (Ref.: IR 04545617).”

Reason: NMP Management Review Committee (MRC) requested the lessons learned be added to the procedure.

- Added IR 04545617 as a reference at the end of step A.3.4 on page 7.
- Added P&L D.23, While an HDL pump is in service, oil discoloration is expected. (Ref.: IR 04545617).

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 5
A.1. Technical Specifications 5
A.2. Licensee Documentation 5
A.3. Technical Information 5
A.3.1. Flow Diagrams 5
A.3.2. Electrical Diagrams 5
A.3.3. Instruction Manuals 6
A.3.4. Other Technical Information 7
A.4. Commitments 8
B. SYSTEM DESCRIPTION 9
C. OPERATING REQUIREMENTS 11
D. PRECAUTIONS AND LIMITATIONS 12
E. STARTUP 15
E.1. System Lineup for Plant Startup 15
E.2. Controller Lineup 23
E.3. Establish Automatic Feedwater Heater Level Control 28
E.4. Start 4TH Point Heater Drain Pumps 38
E.5. Place 4TH Point Heater Drain Pumps In Service 42
E.6. Returning a 4TH Point Heater Drain Pump to Service at Power 46
E.7. Starting a 4TH Point Heater Drain Pump with the Minimum Flow Control Valve Inoperative and Failed Open 50
E.8. Venting HDL Pump Instrumentation 54
F. NORMAL OPERATION 61
F.1. Adjustment of Feedwater Heater Level Controls 61
F.2. Feedwater Temperature/Thermal Power Limit Figure 67
G. SHUTDOWN 68
G.1. Removing a 4TH Point Heater Drain Pump from Service 68
G.2. Feedwater Heater System Shutdown 74
G.3. 4TH Point Heater Drain Pump Shutdown with the Minimum Flow Control Valve Inoperative and Failed Open 77
H. OFF-NORMAL PROCEDURES 80
H.1. Removal of a Low Pressure Feedwater Heater String from Service [SOP] 80
H.2. Removal of a 6TH Point Feedwater Heater from Service [SOP] 85
H.3. Removal of 5TH Point Heater Heating Steam [SOP] 89
H.4. Removal of 4TH Point Heater Heating Steam [SOP] 92

TABLE OF CONTENTS (Continued)

SECTION **PAGE**

H.5. Removal of 3RD Point Heater Heating Steam [SOP] 95
H.6. Power Changes Above 65% Rated Thermal Power when a Heater Drain Pump is Out of Service 96
H.7. Returning 6TH Point Heater to Service at Power 98
H.8. Returning Low Pressure Heater String to Service at Power 112
H.9. Returning 3RD Point Heater Heating Steam to Service 115
H.10. Returning 5TH Point Heater Heating Steam to Service 121
H.11. Returning 4TH Point Heater Heating Steam to Service 129
H.12. Vacuum Drag of Extraction Steam System 137
H.13. Reset of Normal Level Control Valves following Heater High Level 153
H.14. Feedwater Heaters Shell Side Draining, Turbine Offline 155
H.15. Starting a 4TH Point Heater Drain Pump with the Minimum Flow Control Valve Inoperative and Failed Closed 168
H.16. Securing 4TH Point Heater Drain Pump with the Minimum Flow Control Valve Inoperative and Failed Closed 172
H.17. Removal/Restoration of Reheater Level Control Valve from Service 173
H.18. Removal/Restoration of Moisture Separator Level Control Valve from Service 178
H.19. High Debris Strainer Differential Pressure 183
H.20. Adjustment of 2ND Point Drain Receiver Level Controls for ALARA Purposes 184
H.21. Installation and Removal of Temporary Air Conditioning for 4TH Point Heater Drain Pump Motors (2HDL-M1A, 2HDL-M1B, 2HDL-M1C) 189
H.22. Obtaining 2HDL-P1A/B/C Upper and Lower End Bearing Temperature Via Multimeter 191

Attachment 1, Disposition of Previous Revision Attachments 196

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

License Letter 2.C.7, Operation with Reduced Feedwater Temperature is **not** allowed to extend normal fuel cycle.

A.2. Licensee Documentation

USAR Section 10.4.10, Extraction Steam to Feedwater Heater Drain System

A.3. Technical Information

A.3.1. Flow Diagrams

- 12177-PID-8A through 8H, Feedwater Heaters and Extraction Steam Systems
- 12177-PID-8J through 8N, Feedwater Heaters and Extraction Steam Systems
- 12177-PID-8P through 8U, Feedwater Heaters and Extraction Steam Systems
- 12177-PID-8W, Feedwater Heaters and Extraction Steam Systems

A.3.2. Electrical Diagrams

- ESK-5HDL01 through 06, 4th Point Heater Drain Pump 1A (B, C)
- ESK-6DTM01, Crossaround Piping Drain Valves
- ESK-6DTM02, Mstr Sep to Turbine Drain Valves
- ESK-6DTM03 through 06, Cold Reheat Steam Dr Valves
- ESK-6ESS01 through 06, Turbine Extraction Line MOVs
- ESK-7DTM01 through 04, Extraction Line Drain Valves
- ESK-7DTM05, 06 and 13, Drain Valves Aux Controls
- ESK-7DTM07 through 09, 15 and 16, Aux Steam Line Drain Valves
- ESK-7DTM10, Misc. AC Circuits
- ESK-7DTM11 and 12, Extraction Header Drain Valves
- ESK-7DTM14, Aux Stm & Turbine Drain Valves
- ESK-7ESS01, 02 and 11, Extraction ‘A’ (B, C) String Auxiliary Control
- ESK-7ESS03 through 06, Turbine Extraction Line NRV's
- ESK-7ESS07, Turbine Extraction Steamline Isolation Valve
- ESK-7ESS08 through 10, Turbine Extraction Line-Heater Valves
- ESK-7HDL01 through 03, 2nd Point Heater Control
- ESK-7HDL04 through 06, 3rd & 4th Point Heater Control

A. REFERENCES AND COMMITMENTS (Continued)

E.1.1 (Continued)

- ESK-7HDL07 through 09, 5th Point Heater Control
- ESK-7HDL10, 1st & 2nd Point Heaters Auxiliary Cont
- ESK-7HDL11 and 12, 4th Point Htrs Dr Pmp 1A & 1B (1C) Recirc V
- ESK-7SVH01 and 04, 1st & 2nd Point FDW Htr Vents
- ESK-7SVH02 and 05, 3rd & 4th Point FDW Htr Vents
- ESK-7SVH03 and 06, 5th & 6th Point FDW Htr Vents
- ESK-7SVH07, 2nd & 3rd Point FDW Drain Clr Vents
- LSK-4-2.1A through E, LP Feedwater Heater Drains
- LSK-4-2.2, 1st & 2nd Point LP Feedwater Heater Drains
- LSK-6-6A & B, HP Feedwater Heater Drains
- LSK-32-6A through E, Moisture Separator Reheater Drains
- LSK-32-7A & B, Moisture Separator Drains

A.3.3. **Instruction Manuals**

- Instructions for Installation, Operation, and Maintenance of Closed Feedwater Heaters - by - Yuba Heat Transfer Corporation
- Vendor drawings:
    - Byron Jackson IF-7805
    - Byron Jackson 2C-5370

A. REFERENCES AND COMMITMENTS (Continued)

A.3.4. Other Technical Information

- DOC No. NER-2M-024, Heater Drain Pump Maximum Flow Increase
- GE SIL No 370 dated February 1982
- NMP48270 Letter dated November 1989
- Unit 2 Licensing File, dated April 10, 1987, Notes of Telecon with NRC Technical Specification Interpretation, Feedwater Heating
- ESB2-M95-0100, Rev. 1, Letter dated 5/7/95, N2-OP-8, Figure 1
- ESB2-M95-0147, Letter dated 4/7/95, Power Uprate N2-OP-8 questions
- ESB2-M95-0148, Letter dated 4/7/95, Condensate Demineralizer Flow
- CALC #PH-156, Moisture Separator Reheaters Out of Service Heat Balance and Justification for Operation
- GE Report NEDC-32809P Rev 1 (NMPC Doc. # NSSS 01.722-5008), Final Feedwater Temperature
- DER-2-00-3968, HDL Pump Trip On Low Suction Pressure
- EC20090090-000, HDL Pump and Motor Replacement
- ECP-09-000244, Debris Strainers
- EC20090058-000, 3rd Point Heater Replacement
- CR-2013-009678, Unit 2 required Downpower due to Offgas flow excursions
- ECP-15-000396, Trip Logic Improvement for Important BOP Equipment
- ECP-16-000365, Valves 2CNM-V211A and 2CNM-HV58A Are Leaking By Excessively, Request Installing New Valves Downstream To Isolate Leakage
- ECP-15-000610, NMP2 PPC Upgrade - PPC Computer Software Implementation
- ECP-19-000284, Technical Evaluation for PCTCC to N2-OP-8 to Install Portable Cooling to Be Directed to the 4th Point Heater Drain Pump Motors (2HDL-M1A, 2HDL-M1B, 2HDL-M1C)
- EE-008AF, EXTERNAL CONNECT DIAG. 4KV MOTOR
- 0002270036004, OUTLINE 5010 FRAME - 4 POLE - 4000 VOLT -ODPS.O.9822AA
- EE-057CL, Conduit Plan Instrumentation Ventilation Exhaust Equipment Rm El 306'-0"
- https://www.minco.com/resourcecenter/tools/temperatureretable
- CC-AA-415, Maintenance Specification - Measurement and Test Equipment
- IR 0454617, 2HDL-P1A Lower Motor Bearing Oil Is Discolored

A. REFERENCES AND COMMITMENTS (Continued)

A.4. Commitments

- **[C1]**, ESB2-M95-0147, Letter from Engineering Concerning Power Uprate Issues
- **[C2]**, ESB2-M95-0148, Letter from Engineering Concerning Condensate Demin Flow
- **[C3]**, IR 02075324 (legacy CR-1995-003112), Operation of control switches for 2HDL LV35s
- **[C4]**, IR 02077143 (legacy CR-1996-001470), HDL P1A tripped on low level during HDL LIC5A change out
- **[C5]**, N2-TDP-DES-0101, Setpoint control
- **[C6]**, IR 02077143 (legacy CR-1997-002993), 2DTM AOV104, 6th Point Extraction Header Drain Isolated
- **[C7]**, IR 02107505 (legacy CR-2003-000605), OE 15493: Reactor Manual Scram Due to Condenser Tube Leak

B. SYSTEM DESCRIPTION

The purpose of the Feedwater Heaters and Extraction Steam System is to heat the relatively cold condensate to a temperature that is consistent with the reactor vessel inlet requirements. The condensate, passing through the heater tubes, is heated by extraction steam, which is on the shell side of the heaters. This process improves the thermodynamic performance of the power generation cycle.

The feedwater heater and extraction steam system consists of three strings of feedwater heaters. Each string consists of two separate drain coolers, one drain receiver, five low pressure feedwater heaters, one high pressure feedwater heater, one heater drain pump, and the necessary instrumentation and controls.

The condensate booster pump discharge flows into the second point drain cooler, then into the third point drain cooler, then sequentially through the first, second, third, fourth, and fifth point feedwater heaters, then to the feed pumps which discharge to the sixth point feedwater heater.

The sixth point heater shells receive the drains from the reheater drain tanks, 2DSR-TK6A/B, scavaging steam from the reheater and extraction steam from the turbine fourth stage. The condensed steam drains to the fifth point heaters. The fifth point heater shells receive the drains from the sixth point heaters and extraction steam from the cold reheat piping. The fifth point heater drains cascade to the fourth point heaters.

The fourth point heater shells receive the fifth point heater drains, the drains from the moisture separator drain tanks, 2DSM-TK4A/B, and extraction steam from the turbine ninth stage. The fourth point heater drains go directly to the suction of the heater drain pumps, 2HDL-P1A/B/C. The drains are pumped forward through a debris strainer to the condensate system between the fourth and fifth point heaters. The total drain pump flow is approximately one-third of the total feedwater flow.

The third point heater shells receive drains from the clean steam reboilers, HP turbine gland leakoff, and extraction steam from the turbine tenth stage. The resulting drains go to the shell side of the third point drain coolers, 2CNM-DCL3A/B/C. The second point heater shells receive extraction steam from the turbine twelfth stage and pass the condensed steam drains through the second point drain receivers 2HDL-TK2A/B/C to the second point drain cooler shells, 2CNM-DCL2A/B/C.

The first point heater shells receive moisture from the turbine thirteenth stage moisture removal section and extraction steam from the turbine fourteenth stage. The drains from the first point heaters pass through a loop seal to the main condenser.

The extraction steam system has the ability to supply steam to the Building Intermediate Heat Exchangers 2HVH-E1A/E1B, from the cold reheat extraction steam piping. The turbine ninth stage extraction line can supply steam to the Clean Steam Reboilers 2TME-E1A/E1B.

B. SYSTEM DESCRIPTION (Continued)

The extraction steam lines to the building heaters, clean steam reboilers, and the third, fourth and sixth point feedwater heaters have non-return check valves. These check valves prevent the high energy steam from re-entering the turbine, which could cause the turbine to overspeed after a turbine trip. The first and second point feedwater heaters do **not** require non-return check valves since the extraction steam going to these heaters is of a low enough energy that a turbine overspeed would **not** occur after a turbine trip. The fifth point feedwater heaters do **not** require non-return check valves since their extraction point is the eighth HP turbine stage and the isolation of the Main Turbine CIVs on a turbine trip effectively isolate the fifth point heater energy from the low pressure turbine.

The controls for the feedwater heaters and their extraction steam are located on 2CES-IPNL204. The heater drain pumps are controlled from 2CEC*PNL851. The extraction steam to the building heaters and the clean steam reboilers is controlled from 2CES-PNL824.

ECP-15-000396 replaces the existing one-out-of-one trip logic with two-out-of-two trip logic to improve single point vulnerabilities. Additional level switches were installed and they will add the two out of tow trip logic to the high-high trip to the 1st and 2nd stage heaters.

Locally mounted indication panels will provide a status for each level switch. In the event a single switch fails closed, the corresponding LED will illuminate.

C. OPERATING REQUIREMENTS

- N2-OP-2, Moisture Separator Reheaters
- N2-OP-3, Condensate and Feedwater
- N2-OP-4, Condensate Storage and Transfer
- N2-OP-5, Condensate Demineralizer System
- N2-OP-14, Turbine Building Closed Loop Cooling
- N2-OP-19, Instrument and Service Air
- N2-OP-71B, 4160V A.C. Distribution
- N2-OP-71C, 600V A.C. Power Distribution
- N2-OP-75, Station Lighting System

D. PRECAUTIONS AND LIMITATIONS

D.1. When the heater drain pumps are placed in or out of service, closely monitor condensate system flow and condensate demineralizers differential pressure to ensure that adequate feedwater pump suction pressure is maintained.

D.2. Removal of an LP heater string (the first through fifth point heaters and the second and third point drain coolers) or the sixth point heater from service will lower the feedwater temperature and will result in a rise in reactor power.

D.3. All pistol grip control switches must be reset after an automatic trip by momentarily taking the switch to the stop or closed position.

D.4. See section F.2, Feedwater Temperature/Thermal Power Limit Figure, for Feedwater Temperature limits with any feedwater heating out of service.

- Steady state operation with reduced temperature beyond section F.2, Feedwater Temperature/Thermal Power Limit Figure, has **not** been analyzed.
- Reactor power shall be reduced if feedwater temperature is beyond F.2, Feedwater Temperature/Thermal Power Limit Figure, limit until feedwater temperature is restored.

D.5. Operation is limited to 69% reactor power with one high pressure feedwater heater completely isolated (i.e. heating steam and feedwater); provided Moisture-Separator/Reheaters (MSRs) are in service. **[C1]**

D.6. Caution should be taken when closing normally backseated valves with the system pressurized. The valve packing may **not** be leak tight and steam may leak when valve leaves its back seated position.

D.7. When the shellside of a third point heater is isolated, the respective LP heater string must be isolated also. This will include isolating both shell and tubeside per section H.1. **[C1]**

D.8. Flow through a condensate demineralizer should be limited to 3300 gpm and 55 psid (header to header), with a maximum **not** to exceed 3500 gpm and 60 psid (header to header). **[C1]**

D.9. Second point drain tank and fifth and sixth point feedwater heater normal and emergency level controller replacements should be performed at reduced power, usually during a planned downpower. **[C1]**

D.10. With Moisture-Separator/Reheaters (MSRs) out of service, special restrictions apply to continuous operation with any Feedwater Heater out of service. The plant can operate with one low pressure heater or one low pressure heater string, or a sixth point heater, removed from service provided that continuous operation is limited to 43% of rated core thermal power.

D.11. With Moisture-Separator/Reheaters (MSRs) out of service, special restrictions apply to continuous operation with a heater drain pump out of service. The plant can operate with one heater drain pump out of service provided continuous operation is limited to less than or equal to 69% of rated thermal power. With more than one heater drain pump out of service, reactor power is limited to less than or equal to 56%.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.12. The following valves should **not** be isolated or overridden closed during plant operation (Modes 1, 2, or 3): **[C6]**

- 2DTM-AOV111, AUXILIARY STM LINE HDR DRAIN VLV
- 2DTM-AOV143, AUXILIARY SEAL STM HDR LOW POINT DRAIN VLV
- 2DTM-AOV166, AUXILIARY SEAL STM HDR LOW POINT DRAIN VLV
- 2DTM-AOV105, 5TH PT HTR EXTRACTION LINE INL HDR DRAIN VLV
- 2DTM-AOV104, 6TH PT HTR EXTRACTION LINE INL HDR DRAIN VLV

These valves are part of the Turbine Water Induction Prevention System and their associated flowpath should remain in service to allow any condensate to return to the condenser and prevent turbine water induction in case of a turbine trip or power reduction. If isolation is required due to equipment or personnel considerations, Engineering should be consulted prior to isolation.

D.13. If it is necessary to override closed or isolate any of the following valves, operating procedures should be changed to have the valves unisolated or open for plant changes (for example: turbine trip or power reduction). **[C6]**

- 2DTM-AOV106, 5TH PT EXTR STM NRV109 DRAIN VLV
- 2DTM-AOV2A, 5TH POINT HEATER A EXTRACTION LINE DRAIN
- 2DTM-AOV2B, 5TH POINT HEATER B EXTRACTION LINE DRAIN
- 2DTM-AOV2C, 5TH POINT HEATER C EXTRACTION LINE DRAIN
- 2DTM-AOV8A, 6TH POINT HEATER A EXTRACTION LINE DRN
- 2DTM-AOV8B, 6TH POINT HEATER B EXTRACTION LINE DRAIN
- 2DTM-AOV8C, 6TH POINT HEATER C EXTRACTION LINE DRAIN

D.14. Main Steam Line Dose Rates may increase if a Chemistry excursion occurs while H2 Injection System is in service potentially causing a MSLRM alarm.

D.15. To avoid low suction pressure trip during HDL pump start, it is necessary to insure the discharge piping is vented by opening the min flow valve, 2HDL-FV35A (B, C) with verified level in the associated fourth point heater. This is performed in the appropriate heater bay by placing 2HDL-LCS2A (B, C), HTR DRAIN P1A (B, C) FV35A (B, C) control switch, briefly to open.

D.16. Prolonged operation with the Feedwater Heater High Level Dump Valves open, without prior Engineering evaluation to assess the impact on Condenser impingement plates, should be avoided. If such operational conditions exceed one week, contact Operations Management. **[C6]**

D.17. Excessive cycling of the Feedwater Heater Dump Valve could damage the associated condenser penetration resulting in an increase in offgas flow. Immediate actions should be taken to stop/avoid cycling of the heater dump valves.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.18. When operating controllers for the following valves in manual at P851, care should be taken when using the fast detent mode such that controller demand does **not** exceed valve motion (i.e., - demand signal is great enough that valve motion continues after controls are released):

- 2DSR-LVX(Y,Z)65A(B), 2DSR-TK6A(B) LEVEL CONTROL VALVE TO 2FWS-E6A (B, C)
- 2DSM-LVX(Y,Z)75A(B), LEVEL CONTROL VALVE FOR MOISTURE SEPARATOR DRAIN RECIEVER 2DSM-TK4A(B)
- 2HDL-LV4A (B, C), FD WTR HTR CNM-E4A (B, C) WTR LEVEL CONTROL

D.19. Plant transients that affect more than one feedwater heater can magnify system response. Care must be taken when restoring heaters per this procedure under these conditions. This may result in more parameters than one Operator can reasonably monitor and control at IPNL204. A strong pre-job brief and additional personnel should be used to ensure an additional transient is **not** initiated, reference IR 02040897 (legacy CR-2008-005736).

D.20. The following HDL motor starting duty should **not** be exceeded:

- Motor cold – 2 consecutive starts.
- Motor at operating temperature – 1 consecutive start.
- Subsequent starts – with motor running between starts – 20 minutes apart, or with the Motor standing between starts – 45 minutes apart.
- Starts should **not** average more than 6 per day throughout the life of the motor.

D.21. Continued operation of the fourth point Heater Drain Pumps with high strainer differential pressure (greater than 19.0 psid) may result in equipment damage.

D.22. When normal level control for the fourth point heaters is being controlled by HDL-LV24A (B, C), 4TH PT HTR E4A (B, C) HIGH DR, fourth point heater drain pump trips are possible if the controller gain is **not** lowered for fine level control for 2HDL-LIC24A(B, C). If fourth point heater level will be controlled by HDL-LV24A (B, C), 4TH PT HTR E4A (B, C) HIGH DR, for longer than one shift or if fourth point heater parameters are becoming unstable then consider requesting IMD to adjust the gain on the controller to help prevent heater drain pump trips. (Ref.: IR 01963932 / Legacy CR-2010-004993).

D.23. While an HDL pump is in service, oil discoloration is expected. (Ref.: IR 04545617)

G. SHUTDOWN

G.1. Removing a 4TH Point Heater Drain Pump from Service

NOTE

- Steps in this section are performed at 2CEC*PNL851 unless otherwise stated.
- Heater Drain Pumps may be left in operation during short term power reductions below approximately 65% power, however if plant shutdown is in progress, the pumps will be removed from service using this section.
- Operation with the Moisture-Separator/Reheaters (MSRs) out of service and with a Heater Drain Pump out of service and reactor power greater than or equal to 56% is limited to 7 days until further engineering analysis is completed.

CAUTION

- When raising or lowering reactor power during non-emergency conditions, caution must be exercised to prevent occurrence of any of the following:
  - Maximum HDL pump current of 240 amps
  - Maximum HDL pump flow of 4,900 gpm.
  - Cycling of any heater emergency level control valve
  - Exceeding condensate demineralizer maximum and minimum flows
  - With MSRs out of service, exceeding 69% Core Thermal Power
- Removing an HDL pump from service will result in a minor reduction in feed water heating. Power maintenance per N2-OP-101D may be required while performing this procedure.

G.1.1 **IF** MSRs are in service,
THEN CONTINUE at step G.1.3 **[SOP]**,
OTHERWISE MARK this step N/A. OPE

G.1.2 **IF** MSRs are out of service,
THEN VERIFY Reactor power is reduced to less than or equal to 69% per N2-OP-101C or 101D,
OTHERWISE MARK this step N/A. N/A

G. SHUTDOWN (Continued) **Initials**

CAUTION

- Stopping a Heater Drain pump will affect Condensate Demineralizer flows.
- Additional demineralizers may need to be placed into service.

G.1.3 **IF** removing a Heater Drain Pump from service will cause a Condensate Demineralizer's flow to exceed 3300 gpm **OR** 55 psid (header to header),
THEN PLACE additional demineralizers in service per N2-OP-5,
OTHERWISE MARK this step N/A. **[SOP] [C2]** N/A

NOTE

Standby CNM and CNM Booster pumps will automatically start if condensate header flow is greater than 11,000 gpm with less than two HDL pumps running.

G.1.4 **IF** removing a Heater Drain Pump from service (**not** a pump trip),
AND required to maintain Feed Pump suction pressure,
THEN START standby Condensate and Condensate Booster pumps per N2-OP-3,
OTHERWISE MARK this step N/A. **[SOP]** N/A

NOTE

Removal of a Heater Drain Pump from service will result in Condensate Booster Pump Suction Pressure reduction of approximately 25 to 30 psig.

Condensate Booster Pump Suction Pressure should be maintained greater than 132 psig (see N2-OP-5A, section D, PRECAUTIONS AND LIMITATIONS).

G.1.5 **IF** removal of a Heater Drain Pump from service will result in Condensate Booster Pump Suction Pressure less than 132 psig,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **STATION** an Operator at 2CNM-V33, CONDENSATE PREFILTER SYSTEM FULL FLOW BYPASS VALVE ..... ☐

b. **PRIOR TO** HDL pump removal from service,
PERFORM N2-OP-5A, section for Operation of Prefilter 30-inch Manual System Bypass Valve, to
ESTABLISH Condensate Booster Pump Suction Pressure greater than or equal to 160 psig ..... ☐

c. **MAINTAIN** Condensate Booster Pump Suction Pressure greater than 132 psig following pump removal from service ..... ☐ N/A

G. SHUTDOWN (Continued) **Initials**

NOTE

- Steps G.1.6. thru G.1.10 place the Heater Drain pump in the recirculation mode of operation.
- The 4TH PT HEATER 4A (4B/4C) WATER LEVEL HIGH annunciator may be received during performance of the following steps.
- If the Heater Drain Pumps will be in recirculation mode for more than 4 hours, then HDL pumps should be removed from service.
- Per AR 04400874, Operations challenged the basis and whether any flexibility is allowed to this 4-hour recirc requirement. Review of EC20090090-000, Form 5 Design Impact Screen, found the basis for this time requirement to be due to Flow Accelerated Corrosion (FAC) concerns.
- There is **no** concern with exceeding the 4-hour requirement infrequently. An IR will be generated for trending if the 4-hour duration is exceeded.

G.1.6. **IF** an HDL pump runs in the recirculation mode of operation for more than 4 hours during the performance of the remainder of this section,
THEN generate an IR for trending,
OTHERWISE MARK this step N/A. _____

G.1.7. **IF** a planned removal a Heater Drain Pump from service (**not** a pump trip),
THEN at 2CEC*PNL851,
SLOWLY RAISE 2HDL-LV4A (B, C), FD WTR HTR CNM-E4A (B, C) WTR LEVEL CONTROL, setpoint to 47%,
OTHERWISE MARK this step N/A. **[SOP]** _____

G.1.8. **IF** removing a Heater Drain Pump from service (**not** a pump trip),
THEN at 2CES-IPNL204,
SLOWLY LOWER HDL-LV24A (B, C), 4TH PT HTR E4A (B, C) HIGH DR, tape setting in Auto
UNTIL HDL-LV24A (B, C) begins to open
OR tape setting is at 35%,
OTHERWISE MARK this step N/A. **[SOP]** _____

G. SHUTDOWN (Continued) **Initials**

G.1.9. **IF** removing a Heater Drain Pump from service (**not** a pump trip),
THEN PERFORM the following,
OTHERWISE MARK these steps N/A: **[SOP]**

a. At 2CEC*PNL851, **PLACE** 2HDL-LV4A (B, C), FD WTR HTR CNM-E4A (B, C) WTR LEVEL CONTROL, in MAN. ☐

b. At 2CEC*PNL851, **SLOWLY CLOSE** 2HDL-LV4A (B, C), FD WTR HTR CNM-E4A (B, C) WTR LEVEL CONTROL,
WHILE VERIFYING the following:

- At 2CES-IPNL204, HDL-LV24A (B, C), 4TH PT HTR E4A (B, C) HIGH DR, opens to control heater level. ☐
- At 2CEC*PNL851, 2HDL-FV35A (B, C), HTR DRAIN P1A (B, C) RECIRC FV POSN, opens
AND MAINTAINS a minimum system flow of approximately 1,400 gpm. ☐ _____

NOTE

HDL pumps may be left in recirculation mode pending raising Reactor power.

G.1.10. **IF** **not** already performed per ARP,
THEN at 2CES-IPNL204, **SLOWLY LOWER** HDL-LV24A (B, C), 4TH PT HTR E4A (B, C), controller tape setting as follows,
OTHERWISE MARK these steps N/A: **[C5]**

- HDL-LV24A, 4TH PT HTR E4A HIGH DR - **SET** at 15% ☐
- HDL-LV24B, 4TH PT HTR E4B HIGH DR - **SET** at 10% ☐
- HDL-LV24C, 4TH PT HTR E4C HIGH DR - **SET** at 10% ☐ _____

G. SHUTDOWN (Continued) **Initials**

G.1.11. **IF** Heater Drain Pump is to be removed from service **OR** has tripped,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **IF** removing Heater Drain Pump from service (**not** a pump trip),
THEN VERIFY 2HDL-FV35A (B, C), HTR DRAIN P1A (B, C) RECIRC FV POSN is controlling minimum flow,
OTHERWISE MARK this step N/A..... ☐

b. **VERIFY** 2HDL-LV4A (B, C), FD WTR HTR CNM-E4A (B, C) WTR LEVEL CONTROL, closed. ..... ☐

c. **IF** removing Heater Drain Pump from service (**not** a pump trip),
THEN STOP HDL-P1A (B, C), 4TH POINT HEATER DRAIN PMP 1A (B, C)
AND VERIFY 2HDL-FV35A (B, C) closes,
OTHERWISE MARK this step N/A..... ☐

d. **IF** operating with less than three HDL pumps in service
AND HDL pump flow(s) greater than 3,700 gpm,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

- IMD **OBTAIN** initial running HDL pump **AND** motor vibration **AND AFTER** every 2% Core Thermal Power rise. ..... ☐
- **MONITOR** running HDL pump discharge pressures for pump cavitation (possibly: sudden lowering discharge pressure)..... ☐
- **IF** running HDL pumps cavitate,
THEN LOWER pump flow as necessary to stop cavitation,
OTHERWISE MARK this step N/A..... ☐ _____

G. SHUTDOWN (Continued) **Initials**

NOTE

The following is performed as required to limit 4TH Point Heater High Water Level Alarms.

G.1.12. **IF** Reactor is to remain at power,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **IF** MSRs are in service,
THEN SLOWLY ADJUST bias in negative direction on the following flow inputs for stopped Heater Drain Pumps,
OTHERWISE MARK these steps N/A:

- 2DSR-LVX65A (LVY65A, LVZ65A), 6TH PT FW HTR FWS-E6A (B, C) INLET FROM DRAIN TK6A..... ☐
- 2DSR-LVX65B (LVY65B, LVZ65B), 6TH PT FW HTR FWS-E6A (B, C) INLET FROM DRAIN TK6B..... ☐
- 2DSM-LVX75A (LVY75A, LVZ75A), 4TH PT FW HTR CNM-E4A (B, C) INLET FROM DRAIN TK4A..... ☐
- 2DSM-LVX75B (LVY75B, LVZ75B), 4TH PT FW HTR CNM-E4A (B, C) INLET FROM DRAIN TK4B..... ☐ _____

b. **IF** MSRs are **not** in service,
THEN SLOWLY ADJUST bias in negative direction on the following flow inputs for stopped Heater Drain Pumps,
OTHERWISE MARK these steps N/A:

- 2DSM-LVX75A (LVY75A, LVZ75A), 4TH PT FW HTR CNM-E4A (B, C) INLET FROM DRAIN TK4A..... ☐
- 2DSM-LVX75B (LVY75B, LVZ75B), 4TH PT FW HTR CNM-E4A (B, C) INLET FROM DRAIN TK4B..... ☐ _____

G.1.13. **IF** discharge piping of Heater Drain Pump being removed from service will be drained,
THEN in associated heater bay,
PLACE 2HDL-LCS2A (B, C), HTR DRAIN P1A (B, C) FV35A (B, C) control switch, to OPEN,
OTHERWISE MARK this step N/A. **[C3]** _____

G.1.14. **NOTIFY** Chemistry to secure associated Heater Drain Pump continuous sampling. _____

G.2. **Feedwater Heater System Shutdown**

G.2.1. **AFTER** Main Turbine has been tripped,
THEN at 2CEC-PNL824, **VERIFY** the following valves are closed:

- ESS-MOV3A,
  6TH POINT HEATER A EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV28A,
  5TH POINT HEATER A, EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV22A,
  4TH POINT HEATER A, EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV15A,
  3RD POINT HEATER A, EXTRACTION STM ISOL VLV. ..... ☐
- ESS-MOV3B,
  6TH POINT HEATER B EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV28B,
  5TH POINT HEATER B, EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV22B,
  4TH POINT HEATER B, EXTRACTION STM ISOL VLV..... ☐
- ESS-MOV15B,
  3RD POINT HEATER B, EXTRACTION STM ISOL VLV. ..... ☐
- ESS-MOV3C,
  6TH POINT HEATER C EXTRACTION STM ISOL VLV. ..... ☐
- ESS-MOV28C,
  5TH POINT HEATER C EXTRACTION STM ISOL VLV. ..... ☐
- ESS-MOV22C,
  4TH POINT HEATER C EXTRACTION STM ISOL VLV. ..... ☐
- ESS-MOV15C,
  3RD POINT HEATER C EXTRACTION STM ISOL VLV. ..... ☐ _____

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-101D

POWER CHANGES

Revision 03700

Safety Related

CONTINUOUS USE

10 CFR 50.59 REVIEW REQUIRED

Approval Authority: Director-Operations

SUMMARY OF ALTERATIONS

Minor revision to incorporate the following:

PCR-21-00837:

- **Description:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more.
- **Reason:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more. It makes the procedure consistent with N1-OP-43B. Also, the PSC desires to be notified for all load reductions.
- Step F.3.2, changed bullets to numbers, added notes, and made improvements for notifications.
- Step G.1.1, changed bullets to numbers, added notes, and made improvements for notifications.

PCR-21-00890:

- **Description:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR.
- **Reason:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR. These changes were driven by ECP-20-000231 for the GNF3 fuel modification in RFO18.
- Step D.1.10.b, added MSROOS, PLUOOS, and TCVFASOOS.
- Deleted Allowable Modes of Operation with Inoperable Equipment Table and it’s associated notes.
- Added step D.1.10.c.
- Added P&L D.1.11 added per Reactor Engineering comment.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	4
B.	SYSTEM DESCRIPTION	6
C.	OPERATING REQUIREMENTS	6
D.	PRECAUTIONS AND LIMITATIONS	7
E.	STARTUP	15
E.1.	Power Increase From 45% to 100%	15
F.	NORMAL OPERATIONS	23
F.1.	Rated Power Operation	23
F.2.	Routine Reactor Power Maintenance [C12]	25
F.3.	Power Maneuvers Between 50-100%	27
F.4.	Power Maneuvering for Parameter Margin Control	38
F.5.	Trimming of DEHC Pressure Set	43
G.	SHUTDOWN	44
H.	OFF-NORMAL OPERATIONS	51

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

A.1.1. Technical Specifications

Due to the fact that most, if **not** all, of the technical specifications apply to plant power operation, they are **not** listed individually in this procedure.

A.1.2. Licensee Documentation

- Technical Specification Amendment 123 (ARTS/MELLLA+)
- Updated Safety Analysis Report, USAR
- Consult individual operating procedures as necessary.

A.1.3. Technical Information

- Flow Diagrams

  Consult individual operating procedures as necessary.
- Electrical Diagrams

  Consult individual operating procedures as necessary.
- Vendor Manuals

  Consult individual operating procedures as necessary.
- System Instruction Manuals

  Consult individual operating procedures as necessary.

A.1.4. Supplemental References

- NRC Bulletin 88-07 Supplement 1, Power Oscillations in BWR.
- NMPC 865, GE Project Letter.
- Safety Evaluation 91-049, Revision 1, Extended Load Line Limit Analysis.
- Power-Flow Operating Map (EM-950A) Two RCS Loops Operation.
- Power-Flow Operating Map (EM-950B) One RCS Loop Operation.
- EDC 2M00587 Revision to the Power-Flow Operating Map.
- NMPC File Code ESB2-M95-0193, Condensate/Feedwater System Operation Post Update.
- NMPC File Code ESB2-PM96203, Turbine Pressure Control.
- NER-2E-037, APRM/RBM/Technical Specification – Maximum Extended Load Line Limit Analysis (ARTS/MELLLA+) Project.
- SOER 07-1, Reactivity Management. **[C13]**

A.1.4 (Continued)

- RIS 07-21, Revision 1, Adherence To License Power Limit.
- Amendment 140 to the Facility Operating License, Extended Power Uprate.
- IR 02002103, Unit 2 Required Down Power Due to Offgas Flow Excursions.
- ECP-12-000448, NMP2-Implementation of MELLLA+.
- IR 04296001, Received 2RCS*P1B lower mtr bearing oil level low alarm.

A.2. **Performance References**

- LS-CE-1255, Northeast Site Specific - CE (New York)
- N2-OP-101A, Plant Startup
- N2-OP-101C, Plant Shutdown
- N2-OP-10A, Circulating Water System
- N2-OP-27, Generator H_2 and CO_2 System
- N2-OP-29, Reactor Recirculation System
- N2-OP-3, Condensate and Feedwater System
- N2-OP-30, Control Rod Drive
- N2-OP-42, Offgas System
- N2-OP-5, Condensate Demineralizer System
- N2-OP-8, Feedwater Heaters and Extraction Steam
- N2-OP-9, Condensate Air Removal
- N2-OP-91A, Process Computer
- N2-REP-11, Independent Methods of Determining Core Thermal Power
- N2-RESP-12, APRM/LPRM Noise Level Determination During Low Flow Operation
- N2-RESP-4, LPRM Calibration
- N2-SOP-101C, Reactor Scram
- N2-SOP-101D, Rapid Power Reduction
- N2-SOP-29, Sudden Reduction in Core Flow
- N2-TTP-VIB-V001, Turbine Vibration Readings
- OP-AA-106-101, Significant Event Reporting
- OP-AA-106-101, Significant Event Reporting
- OP-AB-300-1003, BWR Reactivity Maneuver Guidance

A.3. **Definitions**

None

A.4. **Commitments**

Sequence Number	Commitment Number	Description
1	SOER 90-2	Nuclear Fuel Defects
2	SOER 90-3	Nuclear Instrumentation Miscalibration
3	DER 2-92-Q-0050	Control Rod Drive Problems Experienced During 12/13 and 12/16/91 Reactor Startup
4	LER 92-17	Rx Scram Due to Loss of Feedwater Pump
5	LER 95-08	Rx Scram Due to High Turbine Vibration
6	DER 2-95-1737	OE Clarification of Basis for Tech. Spec. Neutron Monitoring Flow Referenced Set Point, Recirculation Drive Flow/Core Flow Correlation
7	DER 2-98-3097	Process Computer Failure
8	DER 2-98-3593	Hot Reheat Steam Temperatures Exceeded 50 °F
9	DER 2-99-0421	OE 9387 differences in definition of TBV, OOS between utilities understanding of operability and GE underlying safety assumptions
10	DER-2-99-3922	OE-SEN 208, Reactor Power Ascension with An Unrecognized Loss of Feedwater Heating
11	DER 2-2000-3818	Potential Recirculation System Jet Pump Cavitation
12	CR-2008-000162	Reactor Power Indications Exceeded License Limit
13	CR 2007-004914	INPO SOER 07-1, Reactivity Management
14	CR 2009-002350	Reactor Power spiked above 1850 MWth during Rod Line Adjustment

B. SYSTEM DESCRIPTION

As all plant systems are affected by the plant power operation, individual system descriptions will **not** be discussed in this procedure. See the individual operating procedures for descriptions of systems that are of interest.

C. OPERATING REQUIREMENTS

All systems and their auxiliaries must be in operation, or in a standby readiness per applicable operating procedure.

D. PRECAUTIONS AND LIMITATIONS

D.1.1. Power can be raised by control rod withdrawal or raising reactor recirculation (recirc) flow. Power changes shall **not** be made by simultaneously using reactor recirc flow and control rod withdrawal.

D.1.2. Control rod movement shall be per OP-AB-300-1003, BWR Reactivity Maneuver Guidance.

D.1.3. During power ascension, for each control rod sequence, fuel pre-conditioning rates shall be maintained and per Reactor Engineering instruction. **[C1]**

D.1.4. Recirculation Pump upshift or pump start shall be performed between the 47% Rodline and the 58% Rodline.

D.1.5. Reactor Engineering will provide and update Rodline information during orderly plant startup and shutdown. If a Reactor Engineer is **not** on site, the Unit Supervisor (US) or the STA will verify the Rodline and update the posting at Panel 2CEC*PNL603.

D.1.6. The Main Turbine should **not** be continuously operated with condenser vacuum less than the low vacuum alarm point (23.8 in. Hg). Turbine load should be reduced under deteriorated vacuum conditions in an attempt to restore vacuum. The Turbine should be manually tripped whenever Turbine load is less than 30% (410 MWe) and annunciator 851359 is in alarm with vacuum less than 24.6 in. Hg.

D.1.7. Under normal conditions hotwell water temperature should be maintained less than or equal to 140 °F or as necessary to ensure CY-AB-120-100, Reactor Water Chemistry, Action 1 limits are **not** exceeded.

During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:

- Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
- RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).

D.1.8. Reactor recirculation flow control shall be maintained in LOOP MANUAL.

D.1.9. Technical Specification Requirements During Power Changes

a. When raising Reactor power, N2-RESP-001, Power Distribution Limits Verification, shall be performed once within 12 hours after exceeding 23% Core Thermal Power (CTP) and every 24 hours thereafter.

b. When Reactor power is greater than or equal to 26% Core Thermal Power (CTP) with any recirculation pump in fast speed, and one or more functions with End of Cycle Recirculation Pump Trip (EOC-RPT) trip capability **not** maintained, then action is required within 2 hours to comply with Tech. Spec. 3.3.4.1. **[C9]**

c. When Reactor power is greater than or equal to 23% Core Thermal Power (CTP), and the Main Turbine Bypass System is found to be INOPERABLE, then action is required within 2 hours to comply with Tech. Spec. 3.7.5. **[C9]**

D.1.10. Extended operating domain and equipment out-of-service analyses have been performed for operations within extended operational boundaries and with various plant equipment out-of-service:

a. Extended Operating Domains

1) For Two Recirculation Loop Operation

Operation is allowed above the Maximum Extended Load Line Limit Analysis (MELLLA) Line, provided 100% rated core thermal power is **not** exceeded.

The region of the Power-Flow Map above the MELLLA Line is known as the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Region. Intentional operation beyond the MELLLA+ Boundary is **not** permitted.

The region of the Power-Flow Map between 100% core flow and 105% core flow is known as the Increased Core Flow (ICF) Region.

2) For Single Recirculation Loop Operation

Single Recirculation Loop Operation (SLO) is restricted by two parameters, namely recirculation flow and rod line. The maximum allowable SLO recirculation drive flow is 41,800 gpm due to recirculation piping vibration limitations and the maximum SLO rod line is 89% (Extended Load Line Limit Analysis (ELLLA) Boundary). These restrictions are also described in the SLO Loop Power-Flow Maps (EM-950B-001 and EM-950B-002). Where these two parameters intersect on the Power-Flow Maps restricts SLO maximum power. Intentional Operation beyond the ELLLA Boundary is **not** permitted.

3) Certain plant transients may result in an operating condition beyond the MELLLA+ Boundary. These transients are analyzed - provided the pre-event conditions were within the normal MELLLA+ Region and actions are taken to exit the region above the MELLLA+ Boundary as part of the transient response procedure.

D.1.10 (Continued)

b. Equipment Out-of-Service (EOOS) Analyses

NMP2 has been analyzed for operation in various plant conditions and with various Equipment Out of Service (EOOS). The following conditions may affect the thermal limits as analyzed in the current COLR:

1) Single Recirculation Loop Operation (SLO)
– Per Tech Spec 3.4.1.

2) Safety Relief Valve Out-of-Service (SRVOOS)
– Per Tech Spec 3.4.4.

3) Automatic Depressurization System Out of Service (ADSOOS)
– Per Tech Spec 3.5.1.

4) MSIV Out-of-Service (MSIVOOS)
– Per Tech Spec 3.6.1.3. Refer to N2-OP-1, Main Steam System

5) Turbine Control Valve Out of Service / Turbine Stop Valve Out of Service (TCV/TSVOOS)
– Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

6) Turbine Bypass Valve Out of Service (TBVOOS)
– Per Tech Spec 3.7.5. Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

7) Recirculation Pump Trip Out of Service, also known as EOC-RPT (RPTOOS)
– Per Tech Spec 3.3.4.1.

8) Pressure Regulator Out of Service (PROOS)
– Per Tech Spec 3.2.2.

9) Moisture Separator Reheaters Out of Service (MSROOS)
– Per the COLR.

10) Power Load Unbalance Out of Service (PLUOOS)
– Per the COLR.

11) Turbine Control Valve Fast Acting Solenoids (TCVFASOOS)
– Per the COLR.

D.1.10 (Continued)

c. GNF3 fuel introduces new equipment out service analyses and combinations of out of service equipment. When out of service equipment is identified, utilize Table 4-2 of the COLR to determine the appropriate penalty to apply for the specific condition that exists. Reactor Engineering assistance may be required to determine the correct penalty to apply for equipment out of service.

d. See the Cycle Specific Core Operating Limits Report for applicable thermal limit impacts for various EOOS conditions [FCMS Search = COLR%].

e. Reference the following to identify the allowable combination of conditions:

- 003N2077-R0, Nine Mile Point 2 GNF NFI T0900 Coincident Equipment out-of-service (EOOS) Report [FCMS Search = 003N2077]
- 006N0918-R0, TSD NF-B756 Nine Mile Point Unit 2 Cycle 18 RPTOOS + TBVOOS Thermal Limits [FCMS Search = 006N0918%]

f. **IF** a specific combination of conditions is **not** identified as allowable, **THEN** it is **not** supported **AND** an orderly plant shutdown to less than 23% power is appropriate.

D.1.11. Operation with Turbine Bypass Valves out of service (TBVOOS) coincident with Turbine Control Valve/Turbine Stop Valve out of service (TCV/TSVOOS) is **not** an allowable condition. It is not supported by the licensing analysis per the COLR.

D.1.12. When Reactor power is greater than 23%, Reactor Engineering notification is required before raising Reactor power with recirculation flow or control rod withdrawal except if power is being raised per Reactor Operating Instructions, Section F.2, Power Maintenance or Section F.4, Power Maneuvering for Parameter Margin Control.

D.1.13. All reactivity changes shall be directly supervised by a Senior Reactor Operator. In order to provide this oversight, the SRO shall be stationed at the Controls Area of the Control Room.

D.1.14. Because of many possible plant conditions, deviation from suggested sequence of operations is permitted as directed by the Shift Manager (SM).

D.1.15. Because the Core Thermal Power Application uses "smooth" feedwater flow inputs, it should **not** be used for Rodline determination during or immediately after power transients. Use APRM power and instantaneous core flow values to compute/plot Rodline from power/flow operating map (Ref. A.1.4, EM-950A or EM-950B Power Flow Operating Map).

D.1.16. If Turbine bearing vibration rises significantly (e.g. to a value of greater than or equal to 5 mils or greater than or equal to 2 mil rise) as a result of Turbine load changes, power should be held steady at the nearest practicable level after the vibration rise. **[C5]**

Once the vibration lowers to an acceptable stable value, Turbine load should be held steady for an additional hour before continuing with load changes. This will allow the Turbine rotor(s) to straighten from any rub induced bowing.

D.1.17. When CTP is greater than or equal to 23% and any APRM indicates outside plus or minus 2% of CTP, an APRM gain adjustment shall be performed prior to additional changes in reactor power.

D.1.18. When power is greater than 90% and prior to any power increase, a Core Thermal Power Report/OD-3 from R*TIME should be demanded to ensure there are **no** unplanned failed sensors inputting into the CTP calculation. Any failed sensor(s) will show on the report under the heading “Failed Sensors List.” (N2-OP-91A has a list of sensors.) **[C6]**

D.1.19. The process computer can fail in ways that are **not** obvious to operators. Operators should **not** add reactivity when the core thermal power calculation is **not** accurate. Operators need to know power level with certainty prior to making adjustments. **[C6] [C7]**

D.1.20. When Reactor Thermal Power is greater than 70% with charcoal adsorbers in service, and Offgas System flow is greater than 25 SCFM the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B sample consistent with the effluent path and maintain proper margin to the trip setpoints.

D.1.21. The Charcoal Adsorber must be in service prior to placing the Offgas Radiation Monitor Delay Pipe in service.

D.1.22. MWth oscillation during steady state operation is a known situation for BWRs using recirculation risers and jet pumps. During steady state operations, if MWth oscillation is experienced, greater than normal expected values, Reactor Engineering should be contacted to adjust rod line or reactor recirculation flow.

D.1.23. The averaging algorithm associated with core thermal power computer points NSSQB001A-01M and NSSQB001A-10M may **not** respond accurately to certain variations in real core thermal power such as oscillations brought about by rapid cycling of recirculation flow. The following computer alarms are provided to ensure that design margins are maintained. Accident analyses provide a 2% margin to account for uncertainties associated with the determination of core thermal power. If any of these alarms are actuated immediate operator action is required to ensure that thermal power is trending below 3988 MWth.

- NSSQB001A_01S, CORE THERMAL POWER – 1-SEC INSTANT Setpoint = 4028 MWt
- NSSQB001A-01M, CORE THERMAL POWER – 1 MIN SMTHD CT POWER Setpoint = 4008 MWt
- NSSQB001A-10M, CORE THERMAL POWER – 10 MIN SMTHD CT POWER Setpoint = 3998 MWt

D.1.24. The R*TIME monitoring above 2CEC*PNL603 is set up to display whole numbers only; decimals are truncated. Caution must be used when monitoring plant parameters using these displays.

D.1.25. Maintain power and flow in the operating domain of the applicable single and two loop power-flow operating maps. Determine expected operating location prior to making reactivity changes. **[C11]**

D.1.26. Primary Containment Oxygen shall be controlled per the requirements of Technical Specification 3.6.3.2.

D.1.27. A CRD Deficiency Log entry should be made at any time during the performance of this procedure when a control rod drive mechanism does **not** perform as expected.

D.1.28. During noble Metals Injection, the plant prerequisite is for core flow greater than or equal to 85% flow and reactor power greater than or equal to 70%. Notify Chemistry Department if outside of these parameters during Noble Metals Injection.

D.1.29. When making core reactivity changes, operators shall monitor redundant nuclear instrumentation as well as other plant indications. Instrumentation to be used but **not** limited to includes SRM count meters, SRM period meters, IRM recorders, APRM recorders/monitors, LPRM meters/monitors, Instantaneous and "Smoothed" Power computer points, Generic Output, Recirc Flow indications, Turbine Bypass and Control Valve positions and Reactor Pressure.

D.1.30. For planned power changes 10% or greater, S.O.S. oversight is required.

D.1.31. Fuel precondition is based on raising fuel bundle nodal power at a rate of less than or equal to 0.35 kW/ft per hour. This translates into raising power at a rate of approximately 3%/hour. This can be accomplished as follows:

a. Raise reactor power by 3% over a one-hour period.

b. Raise reactor power by 3% at any time during the one hour-period, but remain at that power level for the remainder of the hour.

D.1.32. If the 3% rate is exceeded over the first hour, then this can be adjusted for in the next hour. Conversely, if the power increase falls short of the 3% rate over the first hour, this power can be adjusted for over the next hour. However, when adjusting for a shortfall in the first hour, it is **not** permitted to exceed a rate of 3.5% per hour in the second hour.

D.1.33. Operating the reactor in compliance with the Licensed Power Limit is demonstrated by the following process;

a. **No** actions are allowed that would intentionally raise core thermal power above 3988 MWth for any period of time. Small, short-term fluctuations in power that are **not** under the direct control of a licensed Reactor Operator (e.g. fluctuation caused by bi-stable flow) are **not** considered intentional. Due to the change in the reactor bi-stable flow characteristics following N2R16 outage, positive reactivity addition for Power Maintenance shall **not** be made unless instantaneous, one minute and ten-minute thermal power is less than or equal to 3986 MWth and trending lower.

b. Closely monitor thermal power during steady state power operation with the goal of maintaining the rolling 1 hour thermal power average at or below 3987.5 MWth. If the core thermal power average for a rolling 1 hour period is found to exceed 3987.5 MWth, take action to ensure that thermal power is reduced to less than or equal to 3987.5 MWth. Power Maintenance shall be performed before exceeding 3987.5 MWth on the rolling two hour thermal power average.

c. The core thermal power average for a shift is **not** to exceed 3988 MWth. For the purpose of this guidance, a shift is defined as 0600 to 1800 or 1800 to 0600.

d. During End of Cycle Operations, Rx power shall **not** be raised above the Rx power which can be achieved with steady state equilibrium Xenon concentrations in the core (all control rods out, all Recirc Pumps in service, raise Core Flow), as determined by Reactor Engineering.

D.1.34. When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition. When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rate at 3966 MWth and above will be done with recirculation flow. **[C14]**

- The above precaution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, “Reverse Power Effect”, for power maintenance. There should be a minimum of three minute wait prior to any further adjustments with Recirculation Flow.

D.1.35. “Steady State” for the purpose of maintaining licensed power level is defined as:

- **No** control rod motion
- **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
- Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
- Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).

It is **not** intended that “steady state” include equilibrium Xenon or Samarium.

D.1.36. The Feedwater minimum flow valves, 2FWR-FV2ABC, are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:

- Single Pump Operation - 23% to 27% Power
- Two Pump Operation with balanced flows - 46% to 53% Power

D.1.37. Failure to have 3 Condensate Booster Pumps (if **not** inhibited) and 2 Feedwater Pumps in service prior to Steam flow exceeding 11.5 Mlbm per hour (approximately 72% power) will result in a Recirc Flow Control Valve Runback.

D.1.38. With MSRs out of service, special restrictions apply to continuous operation with a heater drain pump out of service. The plant can operate with one heater drain pump out of service provided continuous operation is limited to less than or equal to 69% of rated thermal power. With more than one heater drain pump out of service, reactor power is limited to less than or equal to 56%.

D.1.39. Excessive cycling of the feedwater heater dump valve could damage the associated condenser penetration resulting in an increase in Offgas flow. Immediate actions should be taken to stop/avoid cycling of the heater dump valves.

D.1.40. The following parameters are required to be monitored during each reactivity manipulation:

- Redundant Nuclear Instrumentation
- Core Flow
- Reactor Pressure
- Reactor Temperature

D.1.41. In October 2019, station personnel performed Drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.

D.1.42. During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power), vibration of EHC System Piping and components should be closely monitored. In March 2020, NMP Unit 2 required a manual SCRAM in response to a large EHC leak from to a weld failure caused by piping vibrations after an extended period of operation at this power level. More frequent monitoring of EHC piping and system components should be performed when operating in this “high EHC flow” region. If excessive vibration of EHC piping and system components occur, an evaluation should be made to raise or lower Reactor Power to avoid operation in this region.

E. STARTUP

E.1. Power Increase From 45% to 100%

NOTE

- Precautions and Limitations D.1.4 and D.1.16 apply when raising power.
- Power ascents after power reductions are performed per Reactor Engineering Instruction.

E.1.1. **VERIFY** N2-OP-101A, Plant Start-up is completed as indicated by Reactor power approximately 45%. _____

E.1.2. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation..... ☐
- Core Flow ..... ☐
- Reactor Pressure..... ☐
- Reactor Temperature..... ☐ _____

E.1.3. **IF** performing N2-OP-9 Section H.5.0, Main Condenser Vacuum Control for Potential Rub Mitigation,
THEN when power is above 50%,
RESTORE vacuum per N2-OP-9, Section H.5.0,
OTHERWISE MARK this step N/A. _____

NOTE

- All reactivity changes shall be directly supervised by a Senior Reactor Operator. In order to provide this oversight, the SRO shall be stationed at the Controls Area of the Control Room.
- Control Rod withdrawal shall be per Startup Control Rod Sequence or Individual Control Rod Movement Instructions (OP-AB-300-1003). Reactor Engineering will determine when to discontinue use of Startup Control Rod Sequence and begin using Individual Control Rod Movement Instructions.

E.1.4. **RAISE** power per approved ReMA. _____

Initials

E.1.5. **WHILE** raising power,
PERFORM the following:

- **PERIODICALLY MONITOR** Rodline using OD-3..... ☐
- **DURING** Control Rod withdrawal,
  IF any problems operating a specific Control Rod occur,
  THEN REFER to N2-OP-30, Subsection F.12.0
  AND DOCUMENT accordingly,
  OTHERWISE MARK this step N/A. **[C3]**..... ☐
- **WHEN** Core Thermal Power (CTP) is greater than or equal to 23%,
  THEN MONITOR APRMs to ensure indications are within plus or minus 2% of CTP..... ☐ _____

E.1.6. **WHEN** control rod movement specified in each Startup Control Rod Sequence page, Individual Control Rod Movement Instruction, or Rod Worth Minimizer Group has been completed,
THEN a licensed reactor operator
OR other technically qualified member of the technical staff
INDEPENDENTLY VERIFY correct control rod position by selecting each rod in the group moved
AND comparing four rod display indication to Startup Control Rod Sequence. _____

NOTE

The next step is to verify control rod coupling integrity.

E.1.7. **WHEN** a control rod is withdrawn to position 48,
THEN APPLY a continuous withdraw signal
AND OBSERVE the following:

- Position 48 indication remains illuminated on Four Rod Display..... ☐
- Annunciator 603444, CONTROL ROD OVERTRAVEL remains clear… ☐
- Control rod FULL OUT indicating light illuminated on Full Core Display..... ☐
- Any indicated response on nuclear instrumentation while withdrawing control rod..... ☐ _____

E.1.8. **RECORD** coupling integrity check performed by initialing appropriate block on Startup Control Rod Sequence form or Individual Control Rod Movement Instructions as applicable. _____

E.1.9. **IF** deviation from prescribed Startup Control Rod Sequence is required,
THEN CONFIRM SM **AND** Reactor Engineer have initialed working copy of Startup Control Rod Sequence indicating concurrence.
OTHERWISE MARK this step N/A. _____

NOTE

Special restrictions apply to continuous operation with a heater drain pump out of service. Reference D.1.38.

CAUTION

Failure to have 3 Condensate Booster Pumps (if **not** inhibited) or 2 Feedwater pumps in service prior to Steam flow exceeding 11.5 Mlbm per hour (approximately 72% power) will result in a Recirc Flow Control Valve Runback.

E.1.10. **WHEN** Reactor power is 45% to 65%,
THEN PERFORM the following:

a. **PLACE** Fourth Point Heater Drain pumps in operation per N2-OP-8, Feedwater Heaters and Extraction Steam System. ..... ☐

b. **WHEN** Reactor power is approximately 55%,
THEN START standby Condensate pumps
AND Condensate Booster pumps per N2-OP-3, Condensate and Feedwater System. ..... ☐

c. **PLACE** second Feedwater Pump in service in per N2-OP-3, Condensate and Feedwater System, Section E.5.0. ..... ☐

d. **ISOLATE** preheater to 2OFG-E1A & B per N2-OP-42, F.6.0. ..... ☐

e. **VERIFY** Generator Hydrogen addition in automatic per N2-OP-27. ... ☐

f. **DURING** power ascension to 100%,
MONITOR Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☐ _____

Initials

E.1.11. **IF** required **WHEN** Reactor power is approximately 65%,
THEN VERIFY Clean Steam Reboiler Steam Supply auto transfer to Extraction Steam as indicated by the following valve operations,
OTHERWISE MARK these steps N/A:

a. At Panel 851, **VERIFY** 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV open. ..... ☐

b. At Panel 851, **VERIFY** 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV closed. ..... ☐

c. At Panel 824, **VERIFY** 2DTM-AOV119, 2ESS-NRV114 AFTER SEAT DRAIN ISOL closed. ..... ☐

d. At Panel 824, **VERIFY** 2DTM-AOV157, 2ESS-NRV113 UPSTREAM DRAIN ISOLATION closed..... ☐ _____

E.1.12. **IF** a single Heater Drain Pump is out of service,
THEN REFER to N2-OP-8, H.6.0, **WHILE** raising power,
OTHERWISE MARK this step N/A. _____

E.1.13. **WHEN** Reactor power is greater than 65%,
THEN VERIFY the following Reheat Steam Load Control Valves are full open:

- 2MSS-PV28A..... ☐
- 2MSS-PV28B..... ☐
- 2MSS-PV29A..... ☐
- 2MSS-PV29B..... ☐ _____

CAUTION

- When Reactor Thermal Power is greater than 70% with the Charcoal Adsorbers in service and Offgas System flow is greater than 25 SCFM, the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B, Offgas Radiation Monitors sample consistent with the effluent path and maintain proper margin to the trip setpoints.
- The Charcoal Adsorbers must be in service prior to placing the Offgas Radiation Delay Pipe in service.

E.1.14. **BEFORE** reaching 70% Reactor Thermal Power,
VERIFY Offgas System Charcoal Adsorbers are in service per N2-OP-42. _____

Initials

E.1.15. **WHEN** Reactor power is approximately 70%,
THEN at 2CES-IPNL203, **TRANSFER** Bldg. Int. Hx Steam Supply to Extraction Steam as follows:

a. **VERIFY** 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL , is CLOSED. ..... ☐

b. **VERIFY** 2ESS-STV105 BLDG INTMD HT EXCH ST ISOL, is OPEN..... ☐

c. At 2CEC-PNL824,
CLOSE DTM-AOV106, 5TH PT EXTR STM NRV 109 DRAIN VLV. .. ☐ _____

E.1.16. **IF** 2CNM-AOV109 is isolated per N2-OP-3,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

- **RAISE** frequency of first stage shell pressure readings to once per 30 minutes. ..... ☐
- **BEFORE** exceeding 472 psia turbine first stage shell pressure (approximately 71% power), **VERIFY** 2CNM-V223, Condensate Demineralizer Bypass Valve open. ..... ☐ _____

E.1.17. **WHEN** Steam flow exceeds 11.5 Mlbm/hr (approximately 72% power),
THEN CONFIRM on 2CEC*PNL602, "RECIRC RUNBACK INHIBITED" light extinguishes. _____

E.1.18. **IF** plant startup is following refueling cycle outage,
THEN at approximately 75%, **PERFORM** the following,
OTHERWISE MARK these steps N/A:

a. **IF** directed by Reactor Engineer,
THEN N2-RESP-4, LPRM Calibration,
OTHERWISE MARK this step N/A..... ☐

NOTE

For the purpose of satisfying Tech Spec SR 3.1.2.1, equilibrium conditions are met when the reactor is at steady state operations (equilibrium xenon with **no** control rod movement or core flow changes) at or above 75% CTP.

b. **NOTIFY** Reactor Engineering to begin core monitoring for equilibrium conditions..... ☐

c. **WHEN** equilibrium conditions have been established,
THEN PERFORM N2-RESP-02, Reactivity Anomalies within 24 hours to satisfy Tech Spec SR 3.1.2.1. ..... ☐ _____

Initials

E.1.19. **RAISE** Reactor power as directed by Reactor Engineering instruction to maximum recirculation flow
OR rated power using the following power ramp guidelines: **[C1]**

- Control rods may be adjusted as directed by Reactor Engineer **DURING** the power ramp. ..... ☐
- **AFTER** terminating load increase, a soak time of 12 hours at new power level may be required for fuel to be preconditioned at that power level..... ☐
- **IF** it is necessary to interrupt power ramp or soak for a load reduction, **THEN** unit will be returned to power level recommended by Reactor Engineer. ..... ☐
- **IF** fuel has been preconditioned, **THEN** power ascents **AFTER** power decrease shall be made at rate determined by Reactor Engineer..... ☐ _____

NOTE

It is recommended that the Feedwater and Condensate System lineup be in a 3 Condensate, 3 Booster, 2 Feed pump and 3 Heater Drain pump configuration for 100% power. With a 3/2/2/3 lineup the condensate pumps will experience cavitation on transients.

E.1.20. **IF** a plant startup from 0% Reactor power is in progress,
THEN at approximately 90% Reactor power,
PERFORM N2-REP-11, Independent Methods for Determining Core Thermal Power,
OTHERWISE MARK this step N/A. **[C2]** _____

NOTE

N2-OP-91A has a list of sensors.

E.1.21. **WHEN** Power is greater than 90%
AND PRIOR TO any further power increase,
THEN DEMAND an OD-3 to verify there are **no** unplanned failed sensors that input to CTP calculation,
OTHERWISE MARK this step N/A. **[C6]** _____

E.1.22. **WHEN** Power is greater than 90% **PRIOR TO** raising power,
THEN VERIFY an averaged feedwater flow computer point (NSSQB108) is updating. **[C7]** _____

NOTE [C14]

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3 minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.
- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rated at 3966 MWth and above will be done with recirculation flow. This caution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, "Reverse Power Effect", for power maintenance. There should be a minimum of three minute wait prior to any further adjustments with Recirculation Flow.

E.1.23. **WHEN** Power is greater than 90%,
THEN PRIOR TO raising power,
VERIFY proper feedwater heating by using N2-OP-8 (Feedwater Heaters and Extraction Steam System) "Feedwater Temperature/Thermal Power Limit" Figure. **[C10]** _____

E.1.24. **PLACE** additional Condensate Demineralizers into service at the recommendations of Chemistry per N2-OP-5, Condensate Demineralizer System,
OTHERWISE MARK this step N/A. _____

E.1.25. **WHEN** Offgas make up air is **no** longer necessary,
THEN PERFORM the following:

a. **CLOSE** 2ARC-V47..... ☐

b. **INSTALL** cap on 2ARC-V47. ..... ☐ _____

NOTE

IRM "B" indication may be erratic due to localized power oscillations.

E.1.26. **ADJUST** each IRM range switch as necessary to indicate a mid-scale value at P603 as follows:

a. **VERIFY** associated APRM/RBM/IRM recorder switch is in IRM position. ..... ☐

b. Using associated IRM range switch, **SELECT** IRM range which provides an indication as close to mid-scale as possible. ..... ☐

c. **RETURN** associated APRM/RBM/IRM recorder switch to APRM/RBM or OFF as needed. ..... ☐ _____

Initials

E.1.27. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
SLOWLY ADJUST Digital EHC pressure set **UNTIL ONE** of the following is reached:

- ☐ Reactor dome pressure is 1020 psig.
- ☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. _____

F. NORMAL OPERATIONS

F.1. Rated Power Operation

F.1.1. **DURING** late fall, winter, and early spring,
CLOSELY MONITOR Cooling Tower/Circulation Water temperature to **PREVENT FREEZING** of PVC piping in the cooling tower, see N2-OP-10A, Section F.2.0. _____

CAUTION

Steady state operation at core flows greater than 105% (113.925 Mlb/hr) is beyond the upper design boundary of the Increased Core Flow (ICF) region and is **not** permitted.

F.1.2. **MAINTAIN** steady state core flow less than or equal to 105% (113.925 Mlb/hr) of rated core flow where rated core flow is equal to 108.5 Mlb/hr. _____

F.1.3. **DURING** Single Loop Recirculation Pump operation,
REFER to N2-OP-29, Reactor Recirculation System, Subsection H.6.0
AND Tech Spec 3.4.1 for required actions. _____

F.1.4. **IF AFTER** plant startup,
THEN VERIFY N2-REP-11, Independent Methods of Determining Core Thermal Power is performed periodically,
OTHERWISE MARK this step N/A. **[C2]** _____

NOTE

- 2ISC*PT108 which outputs to FWSPA101 and TARS-2237, should be used for setting reactor pressure as FWSPA101 inputs into the CTP calculation.
- If 2ISC*PT108 is **not** available or reliable, use PT109 (TARS-2236) or PAM recorders for pressure adjustment. Call Reactor Engineering for guidance regarding substituting Rx pressure value in CTP calculation and System engineering for evaluation.

CAUTION

The process computer can fail in ways that are **not** obvious to operators. Operators should **not** add reactivity when the process computer core thermal power calculation is **not** accurate. Operators need to know power level with certainty prior to making adjustments. **[C6] [C7]**

F.1.5. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
SLOWLY ADJUST Digital EHC pressure set **UNTIL ONE** of the following is reached:

- ☐ Reactor dome pressure is 1020 psig.
- ☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. _____

F.1.6. **ADJUST** movement restrictions for 2GMC-TV101 via Side-Mounted Handwheel per N2-OP-26. _____

F.2. **Routine Reactor Power Maintenance [C12]**

NOTE

- This section is intended for Power Maintenance only. Changes in power of more than 2% shall be conducted using Section F.3 or F.4.
- CTP trends obtained from the smoothed points NSSQB001A-01M and NSSQB001A-10M will lag slightly behind NSSQB001_01S due to their iterative nature.
- During plant transients or power manipulations CTP should be monitored by either NSSQB001_01S or NSSQB001A-01M.
- If any of the below computer points reach the listed setpoints, immediate operator action is required to ensure that thermal power is trending below 3988 MWT:
  - NSSQB001_01S, CORE THERMAL POWER – 1-SEC INSTANT Setpoint = 4028 MWT
  - NSSQB001A-01M, CORE THERMAL POWER – 1 MIN SMTHD CT POWER Setpoint = 4008 MWT
  - NSSQB001A-10M, CORE THERMAL POWER – 10 MIN SMTHD CT POWER Setpoint = 3998 MWT
- When operating at 98% of rated power or greater, positive reactivity additions with recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition
- "Steady State" for the purpose of maintaining licensed power level is defined as:
  - **No** control rod motion
  - **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
  - Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
  - Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).
- It is **not** intended that "steady state" include equilibrium Xenon or Samarium.
- Power maintenance should maximize steady state power while operating per the limitations described in Section D.1.33.

F.2.1. **CLOSELY MONITOR**
NSSQB001A-60M, CORE THERMAL POWER – 60 MIN AVG,
DURING "Steady State" operation with goal of maintaining rolling two-hour thermal power average between 3984 MWth and 3987.5 MWth. _____

Initials

F.2.2. **IF** NSSQB001A-60M, CORE THERMAL POWER – 60 MIN AVG, is found to exceed 3987.5 MWth,
THEN TAKE ACTION to ensure thermal power is reduced to less than or equal to 3987.5 MWth
AND CLOSELY MONITOR NSSQB001A-60M, CORE THERMAL POWER – 60 MIN AVG, to ensure it remains less than 3987.5 MWth for the shift,
OTHERWISE MARK this step N/A. _____

F.2.3. **WHEN** above 90% power,
AND IF raising power,
THEN DEMAND a Core Thermal Power Report/OD-3 from R*TIME to ensure there are **no** unplanned failed sensors inputting into the CTP calculation; any failed sensor(s) will show on the report under the heading "Failed Sensors List" (N2-OP-91A has a list of sensors),
OTHERWISE MARK this step N/A. _____

F.2.4. **WHEN** above 90% power,
AND IF raising power,
THEN VERIFY a smoothed feedwater flow computer point (NSSQB108) is updating,
OTHERWISE MARK this step N/A. _____

F.2.5. **OBTAIN** Control Room SRO permission to perform power maintenance. _____

F.2.6. **MAKE** a Control Room update regarding planned power maintenance. _____

NOTE

- The M/A flow controllers have low and high-speed detents for valve speed control.
- RCS FCV M/A Station operation is authorized in MANUAL mode only.
- Maximum allowed recirculation flow is 113.9 Mlbm/hr (Computer Point NSSFA01S).
- Do **not** enter the Scram Region, Exit Region, or Heightened Awareness Region of the Power-Flow Operating Map.
- If approaching the RCS Loop Drive Flow Limit of 47,200 gpm for two RCS pump, high-speed operation, use smoothed Computer point iTARS-20455M (iTARS-20465M) for better control.

F.2.7. **ADJUST** 2RCS-HC1603A(B) RECIRC LOOP A(B) FLOW CONTROL by **RAISING OR LOWERING** controller output using low speed detent. _____

F.2.8. **VERIFY** Unit RO logs power adjustment in Control Room Log. _____

F.3. **Power Maneuvers Between 50-100%**

NOTE

- This section is for performing plant maneuvers between 50% to100%, this section assumes that the plant startup has been completed and that power reduction / ascension is **not** from an outage.
- The steps in this section are a guide for plant maneuvers, if it is determined a plant shutdown is required then entry into the shutdown section of this procedure should occur.
- The steps in this section may be performed out of sequence as determined by the Shift Manager.
- In October 2019, station personnel performed drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.
- During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power) vibration of EHC System Piping and components should be closely monitored. Unit 2 required a manual SCRAM in response to a large EHC leak in March 2020, after a weld failure caused by piping vibrations after an extended period of operation at this power level.
- The Feedwater minimum flow valves, 2FWR-2FVA(B,C), are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:
  - Single Pump Operation 23% to 27% Power
  - Two Pump Operation with balanced flows - 46% to 53% Power

F.3.1. **IF** load reduction is in support of Drywell (DW) entry at power,
THEN NOTIFY Outage Control Center that visual verification of adequate oil level in both Reactor Recirculation Pump (2RCS-M1A and 2RCS-M1B) Motor Bearings (Upper and Lower) must be completed
PRIOR TO Drywell exit,
OTHERWISE MARK this step N/A. _____

F.3.2. **PERFORM** the following power reduction preparatory steps, **AND** Steps **not** required may be marked N/A:

NOTE

Steps F.3.2.a thru F.3.2.l may be performed in any order.

a. **IF** prior to a planned shutdown **OR** significant load drop, **THEN INSTRUCT** Plant Performance Group to install vibration data gathering instrumentation at 2CEC-PNL841 per N2-TTP-VIB-V001, **UNLESS** waived by System Engineer..... ☐

NOTE

Per OP-AA-102-101, the following expectations/requirements of timeliness for notifications to Generation Dispatch:

- 2-hour notice is desirable
- 30-minute notice is required
- Immediately before is desirable

b. **VERIFY** the following have been notified of planned power changes:

- Generation Dispatch..... ☐
- Power Control..... ☐
- NDO ..... ☐

NOTE

An unplanned load reduction is one that was **not** previously scheduled.

c. **IF** conducting an unplanned load reduction of 15% **OR** greater, **THEN NOTIFY** Public Service Commission Notification per the following:

1) OP-AA-106-101, Significant Event Reporting ..... ☐

2) LS-CE-1255, Northeast Site Specific - CE (New York) ..... ☐

Initials

F.3.2 (Continued)

d. **IF** containment will be de-inerted,
THEN VERIFY GSN Electric Vaporizers are available ..... ☐

e. **IF** Condensate Demineralizers will be removed,
THEN VERIFY removal order has been developed..... ☐

f. **IF** Clean Steam Reboilers will be swapped to the Aux Boilers,
THEN PREPARE Aux Boilers to supply Clean Steam Reboilers..... ☐

g. **IF** Intermediate Heat Exchangers will be removed from service,
VERIFY Auxiliary Heat Exchangers are in service. ..... ☐

h. **IF** entering Mode 3,
THEN VERIFY all Mode 3 required surveillances are completed
OR ready for completion ..... ☐

i. **IF** desired,
THEN VERIFY N2-OP-25, H.6.0, Preventing Automatic Swapover of Reboiler Steam Supply, is complete..... ☐

j. **IF** planning to reject WCS to the Main Condenser**,**
THEN VERIFY 2NHS-MCC2008-2E, 2WCS-MOV107, RWCU DISCH TO MAIN COND is ON..... ☐

k. **IF** Cooling Tower will be changed to full bypass,
THEN VERIFY Basin Level is adjusted for Cooling Tower Mode Change (Bypass) ..... ☐

l. **IF** performing Scram time testing per N2-OSP-RMC-@001,
THEN prior to commencing power reduction,
VERIFY communication equipment is staged
AND prerequisites are performed. ..... ☐ _____

Initials

F.3.3. Prior to planned power change,
EVALUATE, ASSIGN AND RECORD below any system critical parameter(s) that may be in an abnormal control lineup which may be adversely affected by the load drop (check AIU book; for example, Gland Sealing Steam):

System Parameter	Operator Responsible (i.e., OATC)

_____

F.3.4. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation..... ☐
- Core Flow ..... ☐
- Reactor Pressure..... ☐
- Reactor Temperature..... ☐

_____

F.3.5. **DURING** power change, some of the following systems will require periodic adjustments depending on the significance of the drop.
DIRECT OATC or other available RO(s) to
PERIODICALLY MONITOR the following system parameters
AND NOTIFY the US for adjustment as necessary:

- Reactor Water Cleanup Filter Flows..... ☐
- Cooling Tower Basin Level/Temperature ..... ☐
- Main Generator Reactive Load (MVARs) ..... ☐
- Turbine Vibrations..... ☐
- Main Condenser Vacuum..... ☐
- Gland Sealing Steam Pressure..... ☐
- Gland Seal Exhaust Pressure ..... ☐
- Clean Steam Reboiler Level Control ..... ☐
- Condensate Demineralizer System/Demin Flow ..... ☐
- Feedwater Level Control Response ..... ☐
- MSR Drain Tank Oscillations ..... ☐
- Containment Parameters ..... ☐

_____

F.3.6. Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ..... ☐
- **NOTIFY** NDO if required per OP-AA-102-101. ..... ☐
- **NOTIFY** Chemistry to monitor Offgas chemistry per NF-AB-400-1000, BWR Fuel Integrity Monitoring ..... ☐
- **REVIEW** applicable Precautions and Limitations. ..... ☐
- **OBTAIN** authorization from Shift Manager to reduce power. .. ☐
- **REVIEW** RMI for planned power reduction. ..... ☐
- **COMPLETE** applicable reactivity brief. ..... ☐ _____

b. **PERFORM** the following actions as needed **DURING** power reduction:

1) At 2CEC*PNL851, **PERIODICALLY MONITOR** Offgas System Flow Rate using 2OFG-FI3A/B. ..... ☐

2) **IF** automatic make up air to Offgas is **not** available, **THEN WHEN** Offgas flow is less than or equal to 6 SCFM, **PERFORM** the following, **OTHERWISE MARK** these steps N/A:

a) **REMOVE** pipe cap from 2ARC-V47, ARC SUCTION HDR VENT..... ☐

b) **THROTTLE OPEN** 2ARC-V47 to establish approximately 7 to 10 SCFM on each indicator (2OFG-FI3A/3B)..... ☐

3) **LOWER** power as directed by approved ReMA. ..... ☐

4) **MONITOR** core flow / reactor power on Power to Flow Map... ☐

5) **IF** core flow / power reduction results in entry into the exit region, **THEN ENTER** N2-SOP-29, **OTHERWISE MARK** this step N/A. ..... ☐

6) **MONITOR** rodline as needed based on core conditions. ..... ☐

7) **MONITOR** Feedwater Pump Seal temperatures **AND ADJUST** seal flows per N2-OP-3. ..... ☐ _____

F.3.6 (Continued)

c. **PRIOR TO** going below 80% power perform the following:

1) **MAINTAIN** HVH steam supply on Extraction Steam as follows:

NOTE

If plant trip or power reduction below 50% occurs, building heating could be lost because this step prevents the auto swap from Extraction to Aux Steam.

a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in CLOSE..... ☐

2) **MAINTAIN** Clean Steam Reboiler steam supply on Extraction Steam at P851 as follows:

NOTE

If plant trip or power reduction below 50% occurs, the Extraction Steam supply pressure could be insufficient to support clean steam reboiler operations and result in an auto swap to the emergency sealing steam supply.

a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in CLOSE. ..... ☐ _____

d. **WHEN** Steam flow goes below 11.5 Mlbm/hr (approximately 72% power),
CONFIRM RECIRC RUNBACK INHIBITED lit on 2CEC*PNL602. _____

Initials

F.3.6 (Continued)

e. **CONTINUE** power reduction
AND PERFORM the following as required:

- **REMOVE** Condensate Demineralizers from service as required
AND at recommendations of Chemistry per N2-OP-5, Condensate Demineralizer System. ..... ☐
- At approximately 65%, **PLACE** Heater Drain pumps in recirculation per N2-OP-8, G.1.0. ..... ☐ _____

f. **DURING** power reduction below 65%,
PERIODICALLY MONITOR reheat steam pressures
AND temperatures to ensure LP Turbine nozzle side-to-side inlet temperature differential is maintained less than or equal to 50 °F. _____

F.3.7. Power Ascension

a. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation ..... ☐
- Core Flow ..... ☐
- Reactor Pressure ..... ☐
- Reactor Temperature ..... ☐ _____

b. **PERFORM** the following actions as needed during the power ascension:

1) **MONITOR** core flow / reactor power on power to flow map. ... ☐

2) **IF** core flow / power ascension results in entry into the exit region,
THEN ENTER N2-SOP-29,
OTHERWISE MARK this step N/A. ..... ☐

3) **MONITOR** rodline as needed based on core conditions. ..... ☐

4) **MONITOR** Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☐

5) **WHEN** Reactor power is greater than 65%,
THEN VERIFY the following Reheat Steam Load Control Valves are full open:

- 2MSS-PV28A ..... ☐
- 2MSS-PV28B ..... ☐
- 2MSS-PV29A ..... ☐
- 2MSS-PV29B ..... ☐

6) **WHEN** Steam flow exceeds 11.5 Mlbm/hr (approximately 72% power), **CONFIRM** RECIRC RUNBACK INHIBITED light on 2CEC*PNL602 extinguishes ..... ☐

NOTE

Eight condensate demineralizers are required for 100% power with nine condensate demineralizers preferred for additional operational margin.

7) **PLACE** Condensate Demineralizers in service as required per N2-OP-5
AND Chemistry guidance for restoring demins to service. ..... ☐ _____

F.3.7 (Continued)

c. **WHEN** reactor power is greater than 80%,
THEN PERFORM the following:

1) **RESTORE** HVH steam supply automatic swap to Aux Steam as follows:

a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in AUTO. ..... ☐

b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in AUTO. ..... ☐

2) **RESTORE** Clean Steam Reboiler automatic swap to Aux Steam at P851 as follows:

a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in AUTO. ..... ☐

b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in AUTO. ..... ☐ _____

F.3.7 (Continued)

d. **WHEN** Power is greater than 90%,
PERFORM the following:

1) **WHEN** Power is greater than 90%
AND PRIOR TO any further power increase,
THEN DEMAND an OD-3 to ensure there are **no** unplanned failed sensors that input to CTP calculation. ..... ☐

NOTE [C14]

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.
- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rated at 3966 MWth and above will be done with recirculation flow. This caution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, "Reverse Power Effect", for power maintenance. There should be a minimum of three-minute wait prior to any further adjustments with Recirculation Flow.

2) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY a smoothed feedwater flow computer point (NSSQB108) is updating,
OTHERWISE MARK this step N/A. ..... ☐

3) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY proper feedwater heating by using N2-OP-8, Feedwater Heaters and Extraction Steam System, Feedwater Temperature / Thermal Power Limit Figure,
OTHERWISE MARK this step N/A. ..... ☐

4) **IF** Offgas make-up air is **no** longer necessary,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a) **CLOSE** 2ARC-V47..... ☐

b) **INSTALL** cap on 2ARC-V47. ..... ☐ _____

F.3.7 (Continued)

NOTE

- "Steady State" for the purpose of maintaining licensed power level is defined as:
  - **No** control rod motion
  - **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
  - Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
  - Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).
- It is **not** intended that "steady state" include equilibrium Xenon or Samarium.

e. **WHEN** reactor has been at constant power for greater than two hours,
THEN DECLARE reactor is at "Steady State". _____

f. **NOTIFY** NDO per OP-AA-102-101. _____

g. **PERFORM** Power Maintenance per section F.2 as necessary. _____

NOTE

- 2ISC*PT108 which outputs to FWSPA101 and TARS-2237, should be used for setting reactor pressure as FWSPA101 inputs into the CTP calculation.
- If 2ISC*PT108 is **not** available or reliable, use PT109 (TARS-2236) or PAM recorders for pressure adjustment. Call Reactor Engineering for guidance regarding substituting Rx pressure value in CTP calculation and System engineering for evaluation.

h. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
ADJUST Digital EHC pressure set **UNTIL ONE** of the following is met, **OTHERWISE MARK** this step N/A.

☐ Reactor dome pressure is 1020 psig ..... ☐

☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. ..... ☐ _____

i. **ADJUST** movement restrictions for 2GMC-TV101 via Side-Mounted Handwheel per N2-OP-26. _____

F.4. **Power Maneuvering for Parameter Margin Control**

NOTE

- This section is intended to be used to maintain condenser vacuum, hotwell temperature, and generator loading parameters within acceptable limits while at or near rated conditions
- Rated generator output at 3988 MWth is 1369 MWe. However, variations in generator output above 1369 MWe are permitted provided generator operation remains within the generator capability curve of N2-OP-68. The 233 MVAR lagging reactive load limit is based on a generator real load of 1380 MWe,
- Minimum condenser vacuum above 55% power is 23.8" Hg Vacuum.
- During Normal Operation, maximum operating hotwell temperature is less than or equal to 140 °F.
- During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:
  - Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
  - RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).
- Power changes using this section may be performed using Reactor Recirculation flow without a Reactivity Maneuvering Instruction (RMI). IF Control Rods are used, an RMI is required.
- This section is written to maneuver the plant between 85% and 100% as needed to control the above parameters in a controlled manner without having to perform a rapid power reduction per N2-SOP-101D. If further power reduction is necessary, use Section F.3.0 of this procedure or N2-SOP-101D as applicable.

F.4.1. Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ..... ☐
- **NOTIFY** NDO if required per OP-AA-102-101. ..... ☐
- **REVIEW** applicable Precautions and Limitations. ..... ☐
- **OBTAIN** authorization from Shift Manager to reduce power. .. ☐
- **IF** a ReMA is to be used for the power reduction,
  THEN REVIEW ReMA**,**
  OTHERWISE MARK this step N/A. ..... ☐
- **COMPLETE** applicable reactivity brief. ..... ☐ _____

F.4.1 (Continued)

b. **PERFORM** the following actions as needed **DURING** power reduction:

1) At 2CEC*PNL851, **PERIODICALLY MONITOR** Offgas System Flow Rate using 2OFG-FI3A/B. ..... ☐

2) **IF** automatic make up air to Offgas is **not** available, **AND WHEN** Offgas flow is less than or equal to 6 SCFM, **THEN PERFORM** the following, **OTHERWISE MARK** these steps N/A:

a) **REMOVE** pipe cap from 2ARC-V47, ARC SUCTION HDR VENT. ..... ☐

b) **THROTTLE OPEN** 2ARC-V47 to establish approximately 6 to 10 SCFM on each indicator (2OFG-FI3A/3B). ..... ☐

3) **LOWER** power using Reactor Recirculation flow **OR** as directed by ReMA **UNTIL** applicable parameter is within acceptable limits. ..... ☐

4) **MONITOR** Feedwater Pump Seal temperatures **AND ADJUST** seal flows per N2-OP-3. ..... ☐ _____

Initials

F.4.2. Power Ascension

a. **MONITOR** the following parameters
DURING each reactivity manipulation:

- Redundant Nuclear Instrumentation ..... ☐
- Core Flow ..... ☐
- Reactor Pressure ..... ☐
- Reactor Temperature ..... ☐ _____

b. **PERFORM** the following actions as needed
DURING power ascension:

1) **WHEN** sufficient margin to limiting parameters is restored,
THEN RAISE power using Reactor Recirculation flow
OR as directed by ReMA. ..... ☐

2) **MONITOR** core flow / reactor power on power to flow map. ... ☐

3) **MONITOR** Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☐ _____

c. **WHEN** Power is greater than 90%
AND PRIOR TO any further power increase,
THEN DEMAND an OD-3 to ensure there are **no** unplanned failed sensors that input to CTP calculation. _____

NOTE [C14]

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.
- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rated at 3966 MWth and above will be done with recirculation flow. This caution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, “Reverse Power Effect”, for power maintenance. There should be a minimum of three-minute wait prior to any further adjustments with Recirculation Flow.

d. **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY a smoothed feedwater flow computer point (NSSQB108) is updating,
OTHERWISE MARK this step N/A. _____

F.4.2 (Continued)

e. **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY proper feedwater heating by using N2-OP-8, Feedwater Heaters and Extraction Steam System, Feedwater Temperature / Thermal Power Limit Figure,
OTHERWISE MARK this step N/A. _____

f. **IF** Offgas make-up air is **no** longer necessary,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **CLOSE** 2ARC-V47. ..... ☐

2) **INSTALL CAP** on 2ARC-V47. ..... ☐ _____

NOTE

- "Steady State" for the purpose of maintaining licensed power level is defined as:
  - **No** control rod motion
  - **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
  - Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
  - Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).
- It is **not** intended that "steady state" include equilibrium Xenon or Samarium.

g. **WHEN** reactor has been at constant power for greater than two hours,
THEN DECLARE reactor is at "Steady State". _____

h. **NOTIFY** NDO per OP-AA-102-101. _____

i. **PERFORM** Power Maintenance per section F.2 as necessary. _____

F.4.2 (Continued)

NOTE

- 2ISC*PT108 which outputs to FWSPA101 and TARS-2237, should be used for setting reactor pressure as FWSPA101 inputs into the CTP calculation.
- If 2ISC*PT108 is **not** available or reliable, use PT109 (TARS-2236) or PAM recorders for pressure adjustment. Call Reactor Engineering for guidance regarding substituting Rx pressure value in CTP calculation and System engineering for evaluation.

j. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
SLOWLY ADJUST Digital EHC pressure set
UNTIL ONE of the following is met,
OTHERWISE MARK this step N/A:

☐ Reactor dome pressure is 1020 psig

☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. _____

F.5. **Trimming of DEHC Pressure Set**

F.5.1. **ACCESS** the PRESSURE CONTROL screen 5532 on Panel 2CEC*PNL851 DEHC HMI. _____

F.5.2. **VERIFY** the MANUAL RATE button is selected. _____

F.5.3. **ENTER OR VERIFY** desired TARGET RATE
AND VERIFY RATE is correct. _____

NOTE

The pressure regulator setpoint may be trimmed at a DEHC HMI, using the PRESSURE SETPOINT trim buttons (+0.1 PSIG, -0.1 PSIG, +1.0 PSIG, -1.0 PSIG).

F.5.4. **SELECT** a desired Pressure Set trim button. _____

F.5.5. **VERIFY** message on popup window (screen 7052) is correct,
OTHERWISE, SELECT CANCEL in the popup window
AND REPEAT Steps F.5.3
AND F.5.4. _____

F.5.6. **SELECT** CONFIRM on popup window (screen 7052)
AND VERIFY the following:

- TARGET is correct. ☐
- PRESS SET is changing to the TARGET value. ☐
- PRESSURE is stabilized. ☐ _____

F.5.7. **REPEAT** Steps F.5.3 through F.5.6
UNTIL the desired reactor
OR throttle pressure is achieved. _____

Initials

G. SHUTDOWN

G.1. Power Reduction from 100% to 45% **[EOP]**

> **NOTE**
>
> Performance of this procedure may be required by the EOPs. Changes to this section of the procedure (including renumbering) are required to be reviewed by the EOP Coordinator.

G.1.1. **PERFORM** the following preparatory steps,
AND Steps **not** required may be marked N/A:

> **NOTE**
>
> Steps G.1.1.a thru G.1.1.l may be performed in any order.

a. **IF PRIOR TO** a planned shutdown **OR** significant load drop,
THEN INSTRUCT Plant Performance Group to install vibration data gathering instrumentation at 2CEC-PNL841 per N2-TTP-VIB-V001,
UNLESS IF waived by System Engineer. ☑

> **NOTE**
>
> Per OP-AA-102-101, the following expectations/requirements of timeliness for notifications to Generation Dispatch:
>
> - 2-hour notice is desirable
> - 30-minute notice is required
> - Immediately before is desirable

b. **VERIFY** the following have been notified of planned power changes:

- Generation Dispatch..... ☑
- Power Control..... ☑
- NDO ..... ☑

NOTE

An unplanned load reduction is one that was **not** previously scheduled.

N/A c. **IF** conducting an unplanned load reduction of 15% **OR** greater, **THEN NOTIFY** Public Service Commission Notification per the following:

N/A 1) OP-AA-106-101, Significant Event Reporting ..... N/A

N/A 2) LS-CE-1255, Northeast Site Specific - CE (New York) ..... N/A

d. **IF** containment will be de-inerted, **THEN VERIFY** GSN Electric Vaporizers are available ..... ☑

e. **IF** Condensate Demineralizers will be removed, **THEN VERIFY** removal order has been developed..... ☑

f. **IF** Clean Steam Reboilers will be swapped to the Aux Boilers, **THEN PREPARE** Aux Boilers to supply Clean Steam Reboilers..... ☑

g. **IF** Intermediate Heat Exchangers will be removed from service, **VERIFY** Auxiliary Heat Exchangers are in service. ..... ☑

h. **IF ENTERING** Mode 3, **THEN VERIFY** all Mode 3 required surveillances are completed **OR** ready for completion ..... ☑

i. **IF** desired, **THEN VERIFY** N2-OP-25 H.6.0, Preventing Automatic Swapover of Reboiler Steam Supply is complete..... ☑

j. **IF PLANNING** to reject WCS to the Main Condenser**,** **THEN VERIFY** 2NHS-MCC2008-2E, 2WCS-MOV107, RWCU DISCH TO MAIN COND is ON..... ☑

N/A k. **IF** Cooling Tower will be changed to full bypass, **THEN VERIFY** Basin Level is adjusted for Cooling Tower Mode Change (Bypass) ..... N/A

N/A l. **IF** performing Scram time testing per N2-OSP-RMC-@001, **THEN PRIOR TO** commencing power reduction, **VERIFY** communication equipment is staged **AND** prerequisites are performed. ..... N/A

G.1.2 **MONITOR** APRMs **WHEN** Core Thermal Power (CTP) is ≥ 23%. OPE

N/A G.1.3 **IF** an APRM is outside plus or minus 2% of CTP, **THEN** that APRM is INOPERABLE UNTIL a gain adjustment is performed, **OTHERWISE MARK** this step N/A. N/A

G.1.4 **PRIOR TO** planned load drop,
EVALUATE, ASSIGN AND RECORD below any system critical parameter(s) that may be in an abnormal control lineup which may be adversely affected by the load drop (check AIU book; for example, Gland Sealing Steam):

System Parameter	Operator Responsible (i.e., OATC)
None	N/A

OPE

G.1.5 **DURING** load drop, some of the following systems will require periodic adjustments depending on the significance of the drop.
DIRECT the OATC or other available RO(s) to
PERIODICALLY MONITOR the following system parameters
AND NOTIFY the US for adjustment as necessary:

- a. Reactor Water Cleanup Filter Flows..... ☑
- b. Cooling Tower Basin Level/Temperature ..... ☑
- c. Main Generator Reactive Load (MVARs) ..... ☑
- d. Turbine Vibrations..... ☑
- e. Main Condenser Vacuum..... ☑
- f. Gland Sealing Steam Pressure ..... ☑
- g. Gland Seal Exhaust Pressure ..... ☑
- h. Clean Steam Reboiler Level Control ..... ☑
- i. Condensate Demineralizer System/Demin Flow ..... ☑
- j. Feedwater Level Control Response ..... ☑
- k. MSR Drain Tank Oscillations ..... ☑
- l. Containment Parameters ..... ☑

OPE

> NOTE
>
> Maintain power and flow in the operating domain of the applicable power-flow operating maps

G.1.6 **PERFORM** the following:

- a. **DURING** Reactor power reductions
  ENSURE Reactor Rodline is periodically checked using OD-3. ..... ☑
- b. **PRIOR TO** making reactivity changes,
  DETERMINE expected operating location..... ☑
- c. **DURING** Reactivity Manipulations,
  POST signage to restrict Control Room access. ..... ☑

OPE

Initials

G.1.7. **NOTIFY** the following of power reduction:

a. Generation Dispatch. ..... ☑

b. NDO if required per OP-AA-102-101. ..... ☑ OPE

G.1.8 **OBTAIN** authorization from Shift Manager (SM) to reduce power. OPE

G.1.9 **REDUCE** power per approved ReMA. OPE

G.1.10. **IF** lowering power below 30%,
THEN WHEN power is below 75%,
PERFORM N2-OP-9 Section H.5.0, Main Condenser Vacuum Control for Potential Rub Mitigation. OPE

NOTE

- If low outdoor temperatures are expected during the load drop, then preparations for Cooling Tower mode changes (such as lowering basin level) should be aggressively pursued.
- Precaution and Limitation D.1.16 shall be observed while lowering power.

G.1.11. **DURING** the load drop, at 2CEC*PNL851,
PERIODICALLY MONITOR Offgas System Flow Rate using 2OFG-FI3A/3B. OPE

G.1.12. **IF** automatic make up air to Offgas is **not** available,
THEN WHEN Offgas flow is less than or equal to 6 SCFM,
PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **REMOVE** pipe cap from 2ARC-V47, ARC SUCTION HDR. VENT..... ☐ N/A

b. **THROTTLE OPEN** 2ARC-V47 to establish approximately 7 to 10 SCFM on each indicator (2OFG-FI3A/B)..... ☐ N/A

c. **MONITOR** Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☐ N/A

N/A

G.1.13. **WHEN** lowering power to maintain power
AND flow in operating domain of applicable power-flow maps,
THEN VERIFY the EXIT REGION of power-flow map is **not** intentionally entered.

OPE

G.1.14. **IF** removing a single Heater Drain Pump from service,
THEN PERFORM the following
OTHERWISE MARK this step N/A:

N/A

a. **REDUCE** Reactor power to approximately 85% as specified by Reactor Engineering. ..... ☐ N/A

b. **SECURE** Heater Drain Pump per N2-OP-8, Subsection G.1.0..... ☐ N/A

N/A

G.1.15. **PERFORM** the following to supply steam to HVH System:

a. **VERIFY** 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, open..... ☑

b. **VERIFY** 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, closed... ☑

c. **VERIFY** 2ASS-PV106, BUILDING HEATING INT HX STEAM SUPPLY PCV, is controlling pressure at 110 psig as indicated on 2ASS-PI115 located on TB 277 East, 2CES-RAK22..... ☑

OPE

G.1.16. **CONFIRM** RECIRC RUNBACK INHIBITED light on 2CEC*PNL602 lights
WHEN Steam flow lowers below 11.5 Mlbm/hr (approximately 72% power).

OPE

G.1.17. **DURING** power reduction below 65%
AND UNTIL MSRs are removed from service,
PERIODICALLY MONITOR reheat steam pressures
AND temperatures to ensure LP Turbine nozzle side-to-side inlet temperature differential is maintained less than or equal to 50 °F. **[C8]**

OPE

NOTE

The Heightened Awareness Region is highlighted in blue on the power-flow map. The region entry line is labeled I. DSS-CD Heightened Awareness Line.

N/A G1.18. **IF** Heightened Awareness Region as indicated on EM-950A/B is entered
AND MORE THAN ONE OPRM is inoperable,
THEN PERFORM the following:
OTHERWISE MARK these steps N/A:

N/A a. **IF** Rodline is greater than 61%,
THEN CONTACT Reactor Engineering to perform N2-RESP-12. ..... N/A ☐

N/A b. **SHIFT** an APRM recorder in each division of RPS to fast speed. ..... N/A ☐

N/A c. **MONITOR** APRMs **AND** LPRMs for positive indication of thermal hydraulic oscillations as indicated by:

- N/A Any APRM peak-to-peak value exceeds 10%. ..... N/A ☐
- N/A **OR** a change in character of noise from random to regular with a period of about two seconds. ..... N/A ☐
- N/A **OR** any LPRM peak-to-peak value exceeds three times the baseline value established for that LPRM. ..... N/A ☐

NOTE N/A

Periodic is defined as having a regular frequency of occurrence of approximately two seconds.

- N/A **OR** Periodic LPRM upscale **OR** downscale alarms. ..... N/A ☐
- N/A **OR** APRM/LPRM noise levels demonstrate a worsening trend toward above limits. ..... N/A ☐

N/A d. **IF** positive indication of any neutron flux oscillation occurs,
THEN IMMEDIATELY SCRAM reactor per N2-SOP-101C, Reactor Scram**,**
OTHERWISE MARK this step N/A. ..... N/A ☐

N/A e. **PERIODICALLY MONITOR** Rodline. ..... N/A ☐

N/A f. **WHEN** Core Thermal Power (CTP) is greater than or equal to 23%,
THEN MONITOR APRMs. ..... N/A ☐

N/A g. **IF** an APRM is outside plus or minus 2% of CTP,
THEN APRM is INOPERABLE
UNTIL a gain adjustment is performed,
OTHERWISE MARK this step N/A. ..... N/A ☐

N/A

NOTE

Before removing the Heater Drain pumps from service, sufficient condensate demineralizers should be in service to handle the increase in condensate system flow which will result from shutdown of the Heater Drain pumps. Operation with an extra demineralizer until all the Heater Drain pumps are removed from service may be required.

G.1.19. **REMOVE** Condensate Demineralizers from service at the recommendations of Chemistry per N2-OP-5, Condensate Demineralizer System. OPE

G.1.20. **DURING** Control Rod insertion,
IF any problems operating a specific Control Rod occur,
THEN REFER to N2-OP-30, Subsection F.12.0,
AND DOCUMENT any problems accordingly,
OTHERWISE MARK this step N/A. **[C3]** _____

NOTE

If Heater Drain pumps will be in recirculation mode for more than 4 hours, then the pumps should be shut down per N2-OP-8, Subsection G.1.0.

G.1.21. **WHEN** Reactor power is approximately 65%,
THEN PERFORM the following: **[C4]**

a. **PLACE** Heater Drain pumps in recirculation per N2-OP-8, Subsection G.1.0. ..... ☐

b. **EVALUATE** Turbine Vibration
AND IF required (see Precautions and Limitations),
THEN MAINTAIN steady load for one hour,
OTHERWISE MARK this step N/A..... ☐ _____

NOTE

The Feedwater minimum flow valves, 2FWR-FV2ABC, are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:

- Single Pump Operation - 23% to 27% Power
- Two Pump Operation with balanced flows - 46% to 53% Power

G.1.22. At approximately 55% power **SECURE ONE** of two operating feedwater pumps per N2-OP-3, Step G.1.0. _____

Initials

G.1.23. **EVALUATE** Turbine Vibration
AND IF required (see Precautions and Limitations),
THEN MAINTAIN steady load for one hour,
OTHERWISE MARK this step N/A. _____

G.1.24. **INSERT** control rods per Reactor Engineering instructions
UNTIL Reactor power is approximately 45%. _____

G.1.25. **SECURE** Heater Drain Pumps per N2-OP-8 Subsection G.1.0. _____

G.1.26. **RESTORE** Preheat Steam to 2OFG-E1A & B per N2-OP-42, F.7.0. _____

G.1.27. **IF** Main Transformer winding temperature is less than 50 °C,
THEN CONFIRM Main Transformer does **not** have **ALL** cooling in service,
OTHERWISE MARK this step N/A. _____

G.1.28. **IF** all Main Transformer cooling is in service
AND winding temperature is less than 50 °C,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **GENERATE** an IR..... ☐

b. **TAKE MANUAL CONTROL** of Main Transformer cooling per N2-OP-68, Section H, Manual Operation of Main Transformer Fans.. ☐ _____

G.1.29. **IF** continued power reduction is necessary,
THEN use N2-OP-101C, Plant Shutdown to continue power reduction.
OTHERWISE MARK this step N/A. _____

G.1.30. **ADJUST** movement restrictions for 2GMC-TV101 via Side-Mounted Handwheel per N2-OP-26. _____

H. OFF-NORMAL OPERATIONS

None



Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-101C

PLANT SHUTDOWN

Revision 04200

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director – Operations

SUMMARY OF ALTERATIONS

PCR-22-01248

Description: Removed obsolete P&L D.39 regarding 2FWR-2FV2A(B,C) seat erosion at certain reactor power levels.

Reason: Digital Feedwater Level Control Snap-Action Function within the 2FWR-FV2A(B,C) control logic eliminates need to avoid certain reactor power levels. The change was driven by ECP-17-000688. Therefore, P&L D. 39 is no longer applicable.

- Deleted P&L D.39 and steps automatically renumbered.

  D.39, The Feedwater minimum flow valves, 2FWR-FV2A (B,C) are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pumps, at these reactor power levels, should be minimized:

  - Single Pump Operation: 23% to 27% Power
  - Two Pump Operation with balanced flows: 46% to 53% Power

PCR-22-03304

Description: Added step G.3.30.k.1) to verify RPS signal is bypassed prior to closing MSIVs.

Reason: While JAF was in Mode 2 during control rod insertion for plant shutdown with Main Steam Line Pressure below the Group 1 isolation on Steam Line Pressure Lo trip set point, an Auto Reactor Scram occurred directly following an invalid Group 1 isolation on Steam Line Pressure Lo and Invalid MSIV Closure RPS Trip. The Mode Switch had failed to enact the logic required to bypass the Group 1 isolation on Steam Line Pressure Lo and the MSIV Closure RPS Trip when taken to the Start & Hot STBY position. Due to this, a change was required per Fleet IR 04527718, NEB-JAF-2022-53 Rx Scram from Mode 2 During Plant Shutdown.

TABLE OF CONTENTS

SECTION **PAGE**

A. REFERENCES AND COMMITMENTS 4
A.1. Developmental References 4
A.2. Performance References 4
A.3. Commitments 5
B. SYSTEM DESCRIPTION 7
C. OPERATING REQUIREMENTS 8
D. PRECAUTIONS/LIMITATIONS 9
E. STARTUP PROCEDURE 15
F. NORMAL OPERATION 16
G. SHUTDOWN PROCEDURE 17
G.1. Power Reduction 17
G.2. Reactor Shutdown by Manual Pushbutton Then Mode Switch to Shutdown Scram 30
G.3. Post Reactor Shutdown/Scram Recovery [SOP] 45
G.4. Reactor Shutdown by Individual Control Rod Insertion 63
H. OFF NORMAL PROCEDURE 77
H.1. Control Blade Guide Tube Flushing via Single Rod Scram 77
Attachment 1, RPV Cooldown Flow Chart 79
Attachment 2, IRM/APRM Overlap Verification 81
Attachment 3, Primary Containment AC Circuits 82
Attachment 4, Isolating Steam Line Drains for Pressure Maintenance 83
Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU 88
TABLE 1: Shutdown Cooling Initiation Briefing Checklist 104
TABLE 2: Deleted 105
Attachment 6, Shutdown Procedure Guidance 106
Attachment 7, Pressure Adjustments Using Automatic Cooldown 108

A. REFERENCES AND COMMITMENTS (Continued)

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

A.1.1. Due to the numerous Technical Specifications that apply to plant shutdown, they are **not** listed individually in this procedure.

A.1.2. IRM/APRM overlap SR 3.3.1.1.6

A.1.3. AR 02410659, Reactivity Management Performance Gaps During Low Power Operations (ACE) and NER NC-14-022

A.1.4. CR 1996-002962, 2FWS*MOV21A/B would **not** open during startup.

A.1.5. ECP-15-000610, NMP2 PPC Upgrade

A.1.6. ECP-16-000469, NMP2 DEHC Upgrade

A.1.7. ECP-17-000688, NMP2 Feedwater Digital Controls Upgrade

A.1.8. IR 02107901, Feedwater Level Control Valve 2FWS-LV10C Torque Switch Failed

A.1.9. IR 04323942, Oscillations of DEHC System During Cooldown

A.1.10. IR 04340889, Progressing Pride in Outage Performance - First 72 hours

A.1.11. AR 04451632, Trend IR: Declining trend in Fleet Operations Performance

A.1.12. IR 04474106, N2-OP-101C, Plant S/D, P&L D.37 Conflicts with S/D Strategy

A.1.13. LS-CE-1255, Northeast Site Specific - CE (New York)

A.1.14. OP-AA-106-101, Significant Event Reporting

A.1.15. SOER 07-1, Reactivity Management **[C26]**

A.2. Performance References

A.2.1. N2-OP-10A, Circulating Water System

A.2.2. N2-OP-31, Residual Heat Removal System

A.2.3. N2-OP-47, Fire Detection

A.2.4. N2-OP-61A, Primary Containment Ventilation Purge and Nitrogen System

A. REFERENCES AND COMMITMENTS (Continued)

A.3. Commitments

- **[C1]** NCTS 503518-01, Securing steam loads
- **[C2]** NCTS 502568-00, 2CNM-LV137 capacity
- **[C3]** IR 02062400, Control Rod Drive problems experienced during 12/13 and 12/16/91 Reactor startup
- **[C4]** SER 90-145, EOP Rev. 4 Safety Evaluation
- **[C5]** NCTS 003772-01, Vendor specifications during low load and high back pressure (low vacuum) conditions
- **[C6]** SER 21-92 & IR 02066637, Reassigned to N2-SOP-101C, Sequence Number 1
- **[C7]** IR 02067950, NRC Bulletin 93-03 "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs"
- **[C8]** LER 92-22, Reactor Scram caused by Low Vessel Level
- **[C9]** Deleted
- **[C10]** LER 93-11, Reactor Scram and ESF Actuation caused by poor work operation and planning
- **[C11]** LER 91-17, Transformer fault causes reactor scram and UPS failure causes loss of annunciation which led to declaration of a Site Area Emergency.
- **[C12]** GE SIL 292, Inadvertent Control Rod Withdrawal Supplement 2
- **[C13]** S.E. 94-081, Use of LEFM Feedwater Flow Indication for Calorimetric
- **[C14]** LER 95-08, Reactor Manual Scram Due to High Turbine Generator Vibration
- **[C15]** IR 02076409, LOCA And HELB Analysis Do Not Consider Stroke Time of MSS*MOVs
- **[C16]** IR 02081381, Rising Reactor Water Level following Scram
- **[C17]** IR 02079427, Operability of Low Vacuum Isolation in Modes 2 and 3
- **[C18]** IR 02082241, RPV Level 8 Spike After Placing Shutdown Cooling In Service
- **[C19]** PRNM MOD SER, Verify APRM Setpoints after RX Mode Switch position change

A. REFERENCES AND COMMITMENTS (Continued)

A.3 (Continued)

- **[C20]** IR 02086140, 2MSS-AOV92B failed to close during shutdown
- **[C21]** IR 02086124, Deleted
- **[C22]** IR 02086101, Hot reheat steam temperatures exceeded 50°F
- **[C23]** IR 02094523, Potential Recirculation System Jet Pump Cavitation
- **[C24]** Deleted
- **[C25]** NCTS 504636-00, T.S. Amendment #120
- **[C26]** IR 02033014, INPO SOER 07-1, Reactivity Management
- **[C27]** IR 02034667, Loss of Shutdown Cooling (2RHS*P1A)
- **[C28]** IR 02045652, NMP Unit 1 Failure of SDC IV-38-01 to Open Electrically on Demand
- **[C29]** BP-2009-000228-048, Evaluate industry best practices for Reactor Cooldown and vessel floodup
- **[C30]** USAR, Appendix C, Section C.1.7, BWR Feedwater Nozzle Program AR 04257377-06, License Renewal BWR Feedwater Nozzle Program

B. SYSTEM DESCRIPTION

Many plant systems are affected by a plant shutdown, therefore individual system descriptions will **not** be discussed in this procedure.

C. OPERATING REQUIREMENTS

C.1. Approved Shutdown Control Rod Movement Instructions have been obtained from the Reactor Engineering Department.

C.2. Rod Worth Minimizer is operable, or a qualified person is available to verify rod movement when thermal power is less than or equal to 10% of rated thermal power.

C.3. All plant systems and their auxiliaries are required to be in operation, or in standby condition, to support a normal plant shutdown.

C.4. **Discussion**

It is the intent of this procedure to outline the steps required to achieve a safe reactor shutdown. The sequence suggested may require deviation by the CRO and SM/US to allow for various plant conditions.

D. PRECAUTIONS/LIMITATIONS

D.1. Do **not** exceed a cooldown rate of 100°F in any one-hour period

D.2. Ensure rod motion is per approved Shutdown Control Rod Movement Instructions.

D.3. The Main Condenser is the preferred heat sink; main steam isolation valves shall remain open during plant shutdown period unless a specific task necessitates their closure.

D.4. If fuel failure(s) are known or suspected, before opening vacuum breakers ensure the following:

- RP should assess radiological conditions in the turbine, condenser and hotwell areas.
- Chemistry management should provide guidance regarding condensate demineralizers to be used to preclude creating hotwell airborne contamination.

D.5. During Reactor shutdown the Reactor Water Cleanup System Flow Rate should be maximized by the use of four Reactor Water Cleanup (RWCU) Filter Demins. The system should remain in service as long as possible to minimize long term radiation buildup in Recirculation piping by removing the insoluble impurities (crud bursts) before plate-out occurs.

D.6. Main Turbine should **not** be operated continuously greater than or equal to 55% load (753 MWe) with condenser vacuum less than the low vacuum alarm point (23.8" Hg). Turbine load should be reduced under deteriorated vacuum conditions in an attempt to restore vacuum. The Turbine should be manually tripped if turbine load is less than 30% (410 MWe) and vacuum less than 24.6" Hg. **[C5]**

D.7. The Mechanical Vacuum pump shall **not** be operated if there is any indication of fuel failure (except as directed by EOPs). Suppressed leaking fuel bundles do **not** constitute fuel failure.

D.8. Reactivity changes shall be closely monitored during plant shutdowns. Operators should be aware of the effects on reactivity such as steam load changes, and reductions in coolant temperature. The effects of such activities should be anticipated, and appropriate steps taken to maintain positive control at all times.

D.9. During depressurization of the RPV, either rapidly or normal cooldown, it is possible that the evolution of dissolved gases in the reference legs of level instrumentation may cause notching or mismatch to occur. Attachment 1 of this procedure shall be followed during RPV cooldown, ONLY if RDS Backfill Injection has been secured to one or more level reference legs. **[C7]**

D.10. Initiation of Shutdown Cooling in Mode 3 is the point during Reactor Cooldown at which RPV level reference legs are highly susceptible to notching. Therefore, a brief of the evolution using Table 1 should be performed before performing ANY Shutdown Cooling manipulations. (ONLY required if RDS Backfill Injection has been secured to one or more level reference legs). **[C7]**

D. PRECAUTIONS/LIMITATIONS (Continued)

D.11. If the Narrow Range level instrumentation experiences notching greater than 23", the instrumentation effected may **not** perform any automatic actions as designed. The difference between the L3 (159.3") setpoint and the variable leg tap for the Narrow Range instruments is approximately 23" (i.e. if the instrument is reading 23" higher than actual level, then it may **not** fall below the L3 actuation when actual RPV level is below the variable leg for the instrument). Whereas a notch of anything less than 23" automatic functions would be expected at the magnitude of the notch below the setpoint (i.e. a notch of 6" means the L3 automatic action would occur at an instrument reading of 159.3", but actual RPV level would be 6" lower, or 153.3").

D.12. If both Recirc Pumps trip following a scram, stratification can occur in the lower head area and cause excessive cooldown rates of the lower head to occur

D.13. Core Thermal Power calculation performed by the Process Computer will be inaccurate during WCS Reject mode of operation. Reactor Engineering should be contacted to determine true thermal power. Reactor Power as indicated by the Process Computer may be lower than actual.

D.14. Fire zones with Safe Shutdown Train components are identified in N2-OP-47, Attachment 6, to assist in determining which trains may be lost for plant shutdown.

D.15. If turbine bearing vibration rises significantly (e.g. to a value of greater than or equal to 5 mils or greater than or equal to 2 mil rise) as a result of turbine load changes, load should be held steady at the nearest practicable level after the vibration rise (at or above 160 MWe). **[C14]**

Once the vibration lowers to an acceptable stable value, turbine load should be held steady for an additional hour before continuing with load changes. This will allow the turbine rotor(s) to straighten from any rub induced bowing.

D.16. Low power or shutdown operation in sub-freezing temperatures can result in excessive cooling of Circulating Water. N2-OP-10A requires maintaining cooling tower basin temperature above 45°F.

D.17. Failure to closely monitor Reactor Water level following Scram can lead to overfilling of the Reactor. **[C16]**

D.18. For Operational Condition 2 and 3, Operators must manually ensure that the Vacuum-Low Isolation Actuation Logic is fully operational by placing all four bypass switches to the Normal position before opening any one or all TSVs. **[C17]**

D.19. In Modes 4 or 5 if a Reactor Recirculation pump is **not** operating or trips and can't be restarted, maintain RPV level 227 to 243 inches (or higher if Main Steam Line flooding is **not** a concern) unless otherwise directed by the EOPs. This will provide for natural circulation and enhanced temperature monitoring if Shutdown Cooling (SDC) is lost. This does **not** apply if recirculation pumps are secured for short periods of time (to facilitate rotation of SDC loops or other short duration shutdowns).

D.20. Due to component degradation 2MSS-AOV92A(B) may **not** fully close which could allow some degree of steam bypass leakage. High temperature differentials can be avoided by verifying 2MSS-PV28A(B) and 2MSS-PV29A(B) full closed when removing MSRs from service with the turbine still on line. **[C20]**

D. PRECAUTIONS/LIMITATIONS (Continued)

D.21. When Reactor Thermal Power is greater than 70% with charcoal absorbers in service, and Offgas System flow is greater than 25 SCFM the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B sample consistent with the effluent path and maintain proper margin to the trip setpoints.

D.22. Maintain power and flow in the operating domain of the applicable single and two loop power-flow operating maps. Determine expected operating location before making reactivity changes. **[C23]**

D.23. Primary Containment Oxygen shall be controlled per the requirements of Technical Specification 3.6.3.2.

D.24. When reactor power level reduces below the Point of Adding Heat (POAH), the Turbine Bypass Valves may close, depending on recent power history and steam system valve leakage. This will allow RPV pressure to lower below the EHC pressure setpoint and will add positive reactivity to the core due to negative temperature coefficient effects.

D.25. While performing Section G.4, Reactor Shutdown by Individual Control Rod Insertion, minimize Reactor cooldown until Control Rods are fully inserted.

D.26. As Reactor cooldown occurs while performing Section G.4, Reactor Shutdown by Individual Control Rod Insertion, positive reactivity added may be greater than the negative reactivity addition that is being achieved from control rod insertions. If this occurs, it may be necessary to stop rod movements and up-range the IRMs to follow the power increase. Reactor power will stop increasing when the Point of Adding Heat is reached (typically range 7 or 8 on the IRMs). Inward rod movement could then be recommenced.

D.27. While performing Section G.4, Reactor Shutdown by Individual Control Rod Insertion, avoid the movement of control rods for the purpose of controlling reactor pressure and reactor temperature. If control rods are used to control reactor pressure and temperature, all control rod maneuvers should be of a single notch. Sufficient time between rod maneuvers may be needed to allow for reactor pressure and power response.

D.28. While performing Section G.4, Reactor Shutdown by Individual Control Rod Insertion, closely monitor Turbine Bypass Valve (TBV) position indication. When all TBVs are shut, Reactor pressure control will be dependent on power history and the integrity of the steam system. Actions may be necessary to limit Reactor cooldown to prevent a restart event.

D.29. While performing Section G.4, Reactor Shutdown by Individual Control Rod Insertion, delays in inserting Control Rods should be avoided during controlled shutdowns when power is below that necessary to supply existing steam loads. Control room activities that could interfere with rod insertion (turn-overs, briefs, surveillance testing, etc) should be avoided or deferred whenever possible. Control Room Supervisors shall ensure that operators have a clear understanding of plant conditions that could lead to a return to criticality during a shutdown.

D.30. A CRD Deficiency Log entry should be made at any time during the performance of this procedure when a control rod drive mechanism does **not** perform as expected.

D. PRECAUTIONS/LIMITATIONS (Continued)

D.31. Seat leakage problems have been experienced with the 2FWS-LV10s in the past. After valve repair, the leakage is eradicated or minimized but will degrade as the valve wears. If leakage past the LV10s has been observed to be degrading, caution needs to be observed when operating 2FWS-MOV47A,(B,C) valves. Leakage past the LV10s can cause level and associated power excursions, particularly at lower reactor pressures.

Inadequate thrust in the valve is a contributor to seat leakage per discussion in IR 02107901. 2FWS-LV10A,B,C air actuators are designed with positive seating function that occurs in the last 0.5% of valve travel in the closed direction that maximizes pressure over the piston area and vents the bottom piston to 0 psig.

D.32. The Master Controller output signal is based on 0-100% rated feedwater flow. A one percent output to 2CNM-LV137 and 2FWS-LV55A(B) will produce approximately 40% valve position on initial 1% demand signal.

D.33. Improved BPWS Control Rod insertion process changed the Rod Pattern Control (T/S 3.1.6) and Control Rod Block Instrumentation (T/S 3.3.2.1) technical specifications to allow use of the optional BPWS control rod sequence. **[C25]**

While performing a shutdown of the plant, an optional BPWS control rod sequence may be used provided that all fully withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a Control Rod Drop Accident is eliminated by the confirmation that withdrawn control rods are coupled. When using this control rod insertion sequence for shutdown, the Rod Worth Minimizer may be reprogrammed to enforce the requirements of the improved control rod insertion process or bypassed per the allowance in the technical specifications.

In order to use the improved BPWS shutdown process the control rods must be confirmed to be coupled by ensuring the control rod monthly exercising procedure is current for rods that are fully withdrawn or the control rod is fully inserted before going below the low power setpoint (LPSP) (approximately 10% power).

If shutdown is being performed using the improved BPWS instead of the standard BPWS and RWM, **no** control rod may be withdrawn until the control rod pattern is in compliance with standard BPWS requirements.

D.34. When Core Thermal Power (CTP) is greater than or equal to 23% and any APRM indicates outside ± 2% of CTP, an APRM gain adjustment shall be performed or the APRM will be inoperable.

D.35. When making core reactivity changes, operators shall monitor redundant nuclear instrumentation as well as other plant indications. Instrumentation to be used but **not** limited to includes SRM count meters, SRM period meters, IRM recorders, APRM recorders/monitors, LPRM meters/monitors, Instantaneous and "Averaged" Power computer points, Generator Output, Recirc Flow indications, Turbine Bypass and Control Valve positions and Reactor Pressure. **[C26]**

D.36. Performance of controlled shutdowns (instead of soft scrams) minimizes radioactive crud bursts and Control Rod Drive stresses.

D. PRECAUTIONS/LIMITATIONS (Continued)

D.37. If Reactor Shutdown by Mode Switch to Shutdown Scram is to be performed, it should be done with the Turbine online. Performance of Plant shutdown via Mode Switch to Shutdown Scram after removing the Turbine from service will result in feedwater piping temperature stratification. This stratification is likely to exceed design limits for which a limited number of cycles are allowed.

When the Turbine is tripped, feed water heating is removed, and at reduced power levels, stratification within the feed water flow begins. During the level swell that follows a reactor scram, this stratification is magnified as the feed water flow is minimal during the period that reactor level is above setpoint.

The fatigue is monitored by the Fatigue Management program and is expected to remain within an acceptable level until the end of plant life. It is expected that annunciator 602302, Feedwater Thermal Stratification High, may be received in this condition, which directs prompting an engineering evaluation of Thermal Stratification per its respective ARP. (Reference: IR 04474106, N2-OP-101C, Plant S/D, P&L D.37 Conflicts with S/D Strategy)

D.38. It is preferable to control the plant in a manner that maintains the condenser available for decay heat removal until after Shutdown Cooling has been placed in service. **[C28]**

D.39. As reactor cooldown occurs, positive reactivity is added if the Moderator Temperature Coefficient (MTC) is negative (reference Cycle Management Report).

D.40. Extended periods of operation at low power (such as during operation in hot standby), should be discouraged. If sufficient rods are **not** inserted or rod insertion is interrupted, then a power increase from the cooldown can occur. Minimize the duration of hold points below the point of adding heat. In theory, rod motion should be continuous from insertion of IRMs or SRMs until All Rods In.

D.41. Low Power operations have the potential to result in resonant instabilities in EHC Pressure Control. This condition can occur with the Reactor sub-critical, decay heat less than 1% and a single bypass valve controlling reactor pressure less than 30% open. This condition will be manifested by oscillations in TBV position and small oscillations in reactor pressure. This condition has caused indicated oscillations in reactor water level, core plate differential pressure, and intermittent HPCS line break indication a frequency of approximately 4 to 5 Hz. Operation in this condition should **not** be allowed to continue and is mitigated by taking manual control of Rx pressure with the Bypass Jack which will result in a slightly lower reactor pressure to arrest the resonant oscillations (IR 04127454).

D.42. During operation of the DEHC Automatic Cooldown feature, when the pressure regulator output is less than 1%, a limit cycle may occur. This is due to current setpoints used in the bypass valve opening bias. This is a known condition and can be corrected by establishing pressure control using the Bypass Valve Opening Jack (BOJM). After establishing control on the BOJM, cooldown can either completed manually, or reactor pressure can be lowered by about 10 psig and Automatic Cooldown can be reestablished, at the discretion of the US/SM.

D.43. Prior to allowing individuals under the vessel, the RPS Scram Functions must be defeated per N2-OP-97. Bumping SRM and IRM cables may cause a half or full scram.

D. PRECAUTIONS/LIMITATIONS (Continued)

D.44. 2MSS*MOV111, 112, and 208 are isolation valves in the Main Steam Line Drains between or upstream of the Main Steam Isolation Valves (MSIVs). Because the closure times of these valves are longer than the MSIV closure times, the duration that 2MSS*MOV111, 112, or 208 are open shall be minimized. Minimizing the time these valves are open will ensure we remain within the bounds of the current isolation time analysis. The total allowable time that valves may be open during a given year is less than 135 hours, when Primary Containment Isolation function is required (reference Calculation Dispositions No. ES-177-05E and ES-109-01B). **[C15]**

D.45. When the Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig, the valves are subjected to a differential pressure greater than the engineering defined limit. Therefore, the valves must be declared INOPERABLE and Technical Specification 3.6.1.3, Condition B entered. Once entered, the associated action is to isolate the penetration with at least one valve within one hour (reference Calculation Disposition No. A10.2-O-026).

D.46. Definitions:

CRITICAL:

- Increasing Count Rates, Sustained Positive Period with **no** rod motion.

SUBCRITICAL:

- Sustained negative period,
- **AND** continuously decreasing count rate
- **WITH no** Control Rod motion.

LIMIT CYCLE:

- An oscillation that develops a constant amplitude and frequency.

E. STARTUP PROCEDURE

Refer to N2-OP-101A and N2-OP-101D.

F. NORMAL OPERATION

Refer to N2-OP-101D.

G. SHUTDOWN PROCEDURE

> **NOTE**
>
> This section allows for various starting points, parallel paths, and equivalent indications. Any given shutdown/cooldown would **not** use all of these possibilities. Thus, the steps which are **not** required to support a particular shutdown/cooldown may be made N/A (**not** Applicable) at the discretion of the SM.

G.1. Power Reduction

> **NOTE**
>
> Changes in steam load may cause changes in reactivity which could result in reactor criticality/sub-criticality or undesirable reactivity change. Personnel performing Balance of Plant operations must communicate steam load changes to Operators performing reactivity changes.

G.1.1 Using HU-AA-1211, Pre-Job Briefings,
PERFORM a Pre-Job Brief including Attachment 6, Shutdown Procedure Guidance. OPE /Today

G.1.2 **DISCUSS** the roles **AND** responsibilities of the following personnel:

- Rod Mover ..... ☑
- Peer Checker ..... ☑
- RM SRO ..... ☑
- BOP RO (Steam load impacts) ..... ☑

IF present,
THEN the QNE,
OTHERWISE MARK this step N/A ..... ☐ N/A OPE /Today

G.1.3 **VERIFY** ALL available Auxiliary Boilers are in Hot Standby **OR** in service. OPE /Today

G.1.4 **IF** improved BPWS will be used,
THEN VERIFY RWM is bypassed per N2-OP-95A,
OTHERWISE MARK this step N/A. **[C25]** OPE /Today

G.1.5 **VERIFY** the following controllers are in Manual:

- 2MSS-PIK22A, REHEATER 1A STM PRESS REGULATOR ..... ☑
- 2MSS-PIK22B, REHEATER 1B STM PRESS REGULATOR ..... ☑ OPE /Today

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.6 **DISENGAGE** travel stops for 2MSS-PV28A as follows.

a. **REMOVE** taper pin. OPE /Today

b. **CLOSE** 2MSS-PV28A-ACT-BYPASS. OPE /Today

G.1.7 **DISENGAGE** travel stops for 2MSS-PV28B as follows.

a. **REMOVE** taper pin. OPE /Today

b. **CLOSE** 2MSS-PV28B-ACT-BYPASS. OPE /Today

G.1.8 **VERIFY** the following controllers are in Auto.

- 2MSS-PIK22A, REHEATER 1A STM PRESS REGULATOR..... ☑
- 2MSS-PIK22B, REHEATER 1B STM PRESS REGULATOR..... ☑ OPE /Today

G.1.9 **REDUCE** power from 100% to 45% per N2-OP-101D. _____/_____

G.1.10 **NOTIFY** Generator Dispatch of Reactor shutdown schedule. _____/_____

G.1.11 **IF** an unplanned load reduction (not previously scheduled) of 15% or greater,
THEN WITHIN 1 hour,
PERFORM a Public Service Commission Notification per OP-AA-106-101 **AND** LS-CE-1255. _____/_____

G.1.12 **REVIEW** Precaution/Limitation D.15. **[C14]** _____/_____

G.1.13 **IF** Exit Region as indicated on EM-950A/B is entered,
THEN enter N2-SOP-29. _____/_____

G.1.14 As required,
WITHIN 24 hours **BEFORE** reducing THERMAL POWER to less than 15% of rated,
DE-INERT primary containment per N2-OP-61A, Subsection E.3.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.15. **PERFORM** the following:

a. **DURING** power reduction **UNTIL** MSRs are removed from service, **PERIODICALLY MONITOR** reheat steam pressures **AND** temperatures to ensure L.P. Turbine nozzle side-to-side inlet temperature differential is maintained less than or equal to 50°F **[C22]**. _____/_____

b. At SM discretion, **BETWEEN** 45% Reactor Power and 100 MWe Generator Output, **TRANSFER** house loads to reserve power per N2-OP-71A, Subsection F.3.0. _____/_____

c. **WHEN** reactor power is below 30% **THEN THROTTLE** 2TME-MOV22A (22B, 23A, 23B) **BETWEEN** 10 and 11.2 inches vacuum to mitigate possible packing rub in main turbine. _____/_____

d. **WHEN** Core Thermal Power (CTP) is greater than or equal to 23%, **THEN MONITOR** APRMs to ensure they indicate within ±2% of CTP. _____/_____

G.1.16. **WHEN** required, **THEN REMOVE** one of three operating Condensate and Condensate Booster pumps from service per N2-OP-3, Subsection G.1.0. _____/_____

G.1.17. **PERFORM** the following:

a. **LOWER** Reactor power to approximately 40% by reducing recirculation flow and control rod insertions as specified by Reactor Engineering. _____/_____

b. **DIRECT** CRO to make a CRD Deficiency Log entry per N2-OP-30, Subsection F.12.0, at any time during performance of this procedure **WHEN** a Control Rod Drive Mechanism does **not** perform as expected. _____/_____

NOTE

40% rated core thermal power (RCTP) is the setpoint for the Rod Worth Minimizer (RWM) Low Power Alarm Point (LPAP).

G.1.18. **IF** the improved BPWS will be **not** be used, **THEN DIRECT** Reactor Engineering to perform the following, **OTHERWISE MARK** these steps N/A: **[C25]**

- **VERIFY** correct RWM sequence is loaded..... ☐
- **VERIFY** current rod pattern is in agreement with RWM..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.19. **IF** Standard BPWS is **not** being used,
THEN DIRECT Reactor Engineering to
VERIFY ALL control rods will **not** be fully inserted **BEFORE** going below 10% power have been verified coupled to comply with improved BPWS control rod insertion sequence. **[C25]** _____/_____

G.1.20. **REVIEW** Precaution/Limitation D.15
AND evaluate need to hold power steady for one hour
BEFORE downshifting Recirc. pumps. **[C14]** _____/_____

G.1.21. **IF** Main Condenser Vacuum Control to prevent Main Turbine rubs is desired,
THEN PERFORM N2-OP-9, Attachment 5, as directed from N2-OP-9, Section H.5.0, to make 2ARC-MOV15A(B) throttleable. _____/_____

G.1.22. **WHEN** Reactor power has been lowered to approximately 40%
AND below 61% rodline,
THEN transfer Reactor Recirculation pumps to LFMG set per N2-OP-29, Subsection G.1.0. _____/_____

G.1.23. **WHEN** Reactor power has been lowered to less than 40%,
THEN NOTIFY Reactor Engineering to evaluate Feedwater Flow Correction Factor per N2-REP-32. **[C13]** _____/_____

NOTE

The Feedwater minimum flow valves, 2FWR-FV2A (B,C), are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pumps, at these reactor power levels, should be minimized:

- Single Pump Operation: 23% to 27% Power
- Two Pump Operation with balanced flows: 46% to 53% Power

G.1.24. **REMOVE** Condensate demineralizers from service during shutdown per N2-OP-5, Subsection F.3.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

- During Control Rod insertion, if any problems operating a specific Control Rod occur, refer to N2-OP-30, Subsection F.12.0, and document the problem accordingly.
- Bypassing the RWM while greater than 10% power will facilitate expeditious control rod insertions to minimize the time that the main turbine is operated at low load.

G.1.25. **IF** RWM **not** previously bypassed
OR required for expeditious control rod insertions,
WHILE CONTINUING control rod insertion to reduce power,
at the direction of SM **AND** recommendation of Reactor Engineer,
THEN BYPASS RWM per N2-OP-95A, Subsection H.1.0,
OTHERWISE MARK this step N/A. **[C3]** _____/_____

G.1.26. **AT OR BELOW** 26% Rated Thermal Power,
THEN VERIFY the following:

- Annunciator 603112 (603412), RPS A(B) CONT & STOP V CLOSURE BYPASSED alarm comes in. ..... ☐
- Rod Block Monitor BYPASS (white) light lit on 2CEC*PNL603..... ☐
- Computer Point NMP2C120, RBM BYPASS in alarm. ..... ☐ _____/_____

G.1.27. **LINE UP** Scavenging Steam to Main Condenser by performing the following:

a. **CLOSE** 2DSR-AOV83A **AND** AOV83B, MAIN SCAV STM TO 6^{TH} PT. HTR. _____/_____

b. **VERIFY** 2DSR-AOV82A **AND** AOV82B, SCAV STM TO CONDENSER VLV open. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.28. At approximately 25% power, **PERFORM** the following:

NOTE

DFWLCS will automatically transition between control modes of Low Power Control Mode and High Power Control Mode. Low Power Control Mode will only use single element control. High Power Control Mode primarily uses three element control but can use single element control if all feedwater or all steam flow signals become unavailable.

CAUTION

Seat leakage problems have been experienced with the 2FWS-LV10s in the past. After valve repair, the leakage is eradicated or minimized but will degrade as the valve wears. If leakage past the LV10s has been observed to be degrading, caution needs to be observed when at reduced reactor power and pressures. Leakage past the LV10s can cause level and associated power excursions.

a. **WHEN** Total Feed Water Flow lowers to approximately 15%, of rated (~2.6 Mlbm/hr),
THEN VERIFY the DFWLCS MODE button from any of the DFWLCS HMI display screens changes from HIGH POWER MODE to LOW POWER MODE. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.28 (Continued)

NOTE

- The RWM Low Power Setpoint, LPSP, is about 25% power.
- RWM Surveillance N2-OSP-RMC-Q@003 may be performed at any time below the LPSP.
- RWM Surveillance N2-OSP-RMC-Q@003 is required to be completed **BEFORE** exceeding 1 hour following Reactor Power reduction to less than or equal to 10% RTP, unless RWM is bypassed. (SR 3.3.2.1.2)

b. **IF** RWM is **not** bypassed **AND** improved BPWS will **not** be used,
THEN at 2CEC*PNL603 on RWM display,
CONFIRM RWM below LPSP is indicated
AND COMPLETE N2-OSP-RMC-Q@003, Rod Worth Minimizer Operability Test,
OTHERWISE MARK this step N/A. _____/_____

NOTE

- The steam pressure referenced in the following step was obtained from the IMD Loop Calibration Report. These values are given as guidance if these valves need to be manually opened.
- If any of the following were overridden closed due to failing to operate, they will need to be opened manually:
  - 2MSS-AOV85A
  - 2MSS-AOV85B
  - 2MSS-AOV85C
  - 2MSS-AOV85D

c. On 2CEC-PNL824,
VERIFY Group III Steam Drain Valves open at about 197 to 202 psia as read on MSSPA07. _____/_____

d. **IF** 2WCS-P1B is in service with 3 filters,
THEN REMOVE one filter from service per N2-OP-37, Subsection F.2.0,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

e. **IF** desired,
THEN COMMENCE manual control of Main Condenser Vacuum per N2-OP-9, Section H.5.0, Main Condenser Vacuum Control for Potential Rub Mitigation,
OTHERWISE MARK this step N/A. _____/_____

G.1.29. **BEFORE** reaching 20% Reactor power,
PERFORM the following:

a. **IF** 2WCS-P1A is in service,
THEN PERFORM N2-OP-37, Subsection F.6.0,
OTHERWISE MARK this step N/A. _____/_____

b. **IF** only 2WCS-P1B is available,
THEN PLACE WCS in Full Reject mode per N2-OP-37, Subsection F.13.0,
OTHERWISE MARK this step N/A. _____/_____

c. **REMOVE** Zinc Injection Pacification System (ZIP) from service per N2-OP-3, Subsection G.1.8. _____/_____

G.1.30. As required, **TRANSFER** Instrument Nitrogen System to Instrument Air System per N2-OP-61A, Subsection F.18.0. _____/_____

G.1.31. At approximately 20% power, **PERFORM** the following:

a. **INITIATE** Condenser Neck Spray per N2-OP-3, Subsection F.7.0. _____/_____

b. At DEHC HMI LOAD CONTROL screen (5535),
ENTER Target Load **AND** desired Turbine Load Ramp Rate to reduce Main Generator output
UNTIL approximately ½ to 1 ½ Bypass Valves are OPEN. _____/_____

c. **VERIFY** Main Turbine Bypass valves are controlling Reactor pressure. _____/_____

d. **NOTIFY** System Power Control unit is ready to be taken offline. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.31 (Continued)

CAUTION

If Reactor Shutdown by Mode Switch to Shutdown Scram is to be performed, it should be done with the Turbine online. Performance of Plant shutdown via Mode Switch to Shutdown Scram after removing the Turbine from service will result in feedwater piping temperature stratification. This stratification is likely to exceed design limits for which a limited number of cycles are allowed.

When the Turbine is tripped, feed water heating is removed, and at reduced power levels, stratification within the feed water flow begins. During the level swell that follows a reactor scram, this stratification is magnified as the feed water flow is minimal during the period that reactor level is above setpoint.

The fatigue is monitored by the Fatigue Management program and is expected to remain within an acceptable level until the end of plant life. It is expected that annunciator 602302, Feedwater Thermal Stratification High, may be received in this condition, which directs prompting an engineering evaluation of Thermal Stratification per its respective ARP. (Reference: IR 04474106, N2-OP-101C, Plant S/D, P&L D.37 Conflicts with S/D Strategy)

e. As directed by SM, **PERFORM** Reactor Shutdown by Manual Pushbutton then Mode Switch to Shutdown Scram, Section G.2, **OTHERWISE MARK** this step N/A. _____/_____

G.1.32. **WHEN** less than 20% power,
THEN PERFORM the following:

CAUTION

Due to component degradation 2MSS-AOV92A and 2MSS-AOV92B may **not** fully close.

a. At approximately 180 MWE, at 2CEC*PNL851,
REMOVE Moisture Separator/Reheaters from service by **CLOSING** 2MSS-AOV92A **AND** AOV92B, MS RHTRS STM SPLY valves. _____/_____

b. **IF** 2MSS-AOV92A(B) does **not** fully close,
THEN at 2CEC*P851 using REHEATER 1A(B) STM PRESS REGULATOR in manual,
VERIFY closed STM LOAD CONTROL VALVE 2MSS-PV28A(B) **AND** STM LOAD CONTROL BYPASS VLV 2MSS-PV29A(B),
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.32 (Continued)

c. Continue to **INSERT** control rods per approved Shutdown Control Rod Movement Instructions, **MAINTAIN** reactor power below 20%. _____/_____

d. IF **not** already performed, at approximately 100 MWe Generator output,
THEN transfer house loads to reserve power per N2-OP-71A, Subsection F.3.0,
OTHERWISE MARK this step N/A. _____/_____

NOTE

- The steam pressure referenced in the following step was obtained from the IMD Loop Calibration Report. These Values are given as guidance if these valves need to be manually opened.
- If 2MSS-AOV201 was overridden closed due to failing to operate, 2MSS-AOV201 will need to be opened manually.

e. On 2CEC-PNL824, **VERIFY** Group II Steam Drain Valves open at about 97 to 102 psia as read on MSSPA06 OR MSSPA07. _____/_____

f. **NOTIFY** System Power Control unit is ready to be taken off line. _____/_____

g. At the DEHC HMI LOAD CONTROL (screen 5535),
REDUCE Load Set
UNTIL Generator load is approximately 75 MWe. _____/_____

h. **REMOVE** Main Generator from service
AND SHUT IT DOWN per N2-OP-68, Subsection G.1.0. _____/_____

i. **SHUTDOWN** Turbine per N2-OP-21, Subsection G.1.0. _____/_____

G.1.33. **REMOVE OR** transfer **ALL** of the major steam loads from Main Steam System to Auxiliary Boilers, with exception of Steam Jet Air Ejectors (SJAE) as follows:

a. **VERIFY** SJAE on Main Steam System. _____/_____

b. **START BOTH** Auxiliary Boilers, as necessary, per N2-OP-48, Subsection E.1.0. _____/_____

c. **IF** Building Heating is required,
THEN switch Building Heating to Auxiliary Boilers per N2-OP-49, Subsection F.10.0,
OTHERWISE MARK this step N/A. _____/_____

d. **TRANSFER** Reboiler Steam Supply from Main Steam to Auxiliary Boiler Steam per N2-OP-25, Subsection F.4.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

Intent of the following step is to supply Engineering with a trend of level compared to pressure and analyze for the "notching" effects attributable to non-condensable contamination of the reference legs. Trend information for all available level points is desirable.

G.1.34. **IF** any of RDS Backfill legs are **not** in service,
AND IF required per Shift Manager (points are available on PPC),
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. On trend recorder (2CEC-DSPL620, 2CEC-DSPL621, 2CEC-DSPL622) **SET UP** trends for Reactor pressure (FWSPA101)
AND narrow range A level (ISCLA01) on one recorder
AND narrow range B level (ISCLA02) on another recorder. _____/_____

b. **SET** pressure for 0 to 1020 psig
AND level to monitor 170 to 200 inches. _____/_____

NOTE

- Technical Specifications 3.1.6 - Rod Pattern Control and 3.3.2.1 - Control Rod Block Instrumentation, Function 2 (Rod Worth Minimizer) are applicable in Modes 1 and 2 at less than or equal to 10% RTP.
- When below 10% CTP **no** control rod shall be withdrawn unless the control rod pattern is in compliance with the standard BPWS requirements.

G.1.35. **BEFORE** reducing power below 10% CTP,
VERIFY the following:

- Control rod coupling integrity for ALL fully withdrawn control rods has been verified by Reactor Engineering. ..... ☐
- Control rods have **not** been confirmed to be coupled have been inserted to position 00 as determined by Reactor Engineering..... ☐
- **IF** RWM is **not** bypassed
AND will be used for the shutdown,
THEN VERIFY RWM is operable
OR Tech Spec 3.3.2.1 is satisfied,
OTHERWISE MARK this step N/A..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.36. As directed by SM,
CONTINUE Control Rod insertion per approved Shutdown Control Rod Movement instructions to lower Reactor power to 5 to 10% as indicated on APRM recorders,
OTHERWISE MARK this step N/A.

G.1.37. **VERIFY** ALL Mode 3 required surveillances are completed **OR** in progress. _____/_____
US

_____/_____
IV SM

G.1.38. IF Section G.4, Reactor Shutdown by Individual Control Rod Insertion, is to be used,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

NOTE

Main steam line radiation monitor operability is determined by N2-ISP-MSS-Q@009, Attachments 1 through 4.

a. **VERIFY** ALL Mode 2 required surveillances are completed **OR** in progress. _____/_____

b. **BEFORE** entry into Mode 2,
VERIFY the following equipment operable:

- 2MSS*RT46A (TS 3.3.7.2) ..... ☐
- 2MSS*RT46B (TS 3.3.7.2) ..... ☐
- 2MSS*RT46C (TS 3.3.7.2) ..... ☐
- 2MSS*RT46D (TS 3.3.7.2) ..... ☐
- At least 3 IRMs operable in RPS Division 1 ..... ☐
- At least 3 IRMs operable in RPS Division 2 ..... ☐ _____/_____

c. **VERIFY EITHER** 2FWS-P1A **OR** 2FWS-P1B in service to support shutdown. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.1.39. **WHEN** reactor power lowers below 10%
OR as directed by SM,
THEN PERFORM the following:

a. **REMOVE** Oxygen in Feedwater Injection System (OFI) from service per N2-OP-3, Subsection G.2.0. _____/_____

b. **REMOVE** the Zinc Injection Pacification System (ZIP) from service per N2-OP-3, Subsection G.1.8. _____/_____

c. As reactor power is reduced,
VERIFY 2FWS-LV55A(B) is opening
as 2FWS-LV10A(B) closes per N2-OP-3, Subsection G.1.6. _____/_____

G.1.40. **WHEN** requested by Rad Protection to support drywell entry for inspection/access,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **ENERGIZE** necessary drywell lighting, receptacles and communications circuits listed on Attachment 3. _____/_____

b. **NOTIFY** EMD to perform necessary PM testing on drywell lighting, receptacles and communications. _____/_____

NOTE

- Condenser vacuum must be maintained by SJAE during depressurization using the Bypass valves when Reactor power is greater than 5%.
- Verify IRM/APRM overlap prior to entering Mode 2 from Mode 1. (SR 3.3.1.1.6)
- It is preferable to control the plant in a manner that maintains the condenser available for decay heat removal until after Shutdown Cooling has been placed in service. **[C28]**

G.1.41. **CONTINUE** shutdown by performing **ONLY ONE** of the following sections:

- Section G.2, Reactor Shutdown by Manual Pushbutton Then Mode Switch to Shutdown Scram ..... ☐
- Section G.4, Reactor Shutdown by Individual Control Rod Insertion ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

- Condenser vacuum must be maintained by SJAE during depressurization using the Bypass valves when Reactor power is greater than 5%.
- Verify IRM/APRM overlap prior to entering Mode 2 from Mode 1. (SR 3.3.1.1.6)
- It is preferable to control the plant in a manner that maintains the condenser available for decay heat removal until after Shutdown Cooling has been placed in service. **[C28]**

G.2. **Reactor Shutdown by Manual Pushbutton Then Mode Switch to Shutdown Scram**

G.2.1. **IF** Plant Shutdown will **not** be from Mode 1 directly to Mode 3,
THEN at approximately 5 to 10% Reactor Power,
PERFORM the following to verify proper IRM/APRM overlap,
OTHERWISE MARK these steps N/A: **[T/S],**

a. **VERIFY** ALL IRM range selector switches on Range 10. _____/_____

b. **VERIFY** ALL operable IRM detectors fully inserted per N2-OP-92, Section G.3.0. _____/_____

NOTE

The Acceptance Criteria for IRM/APRM overlap in Step G.2.1.f requires that operable IRMs be on range 9 or 10 and **no** concurrent power changes in progress.

c. **BEFORE** any operable APRM reading falling less than 4% as indicated on IRM/APRM recorders C51-R603A, B, C, D,
RECORD operable APRM readings below:

APRM 1 _____%

APRM 2 _____%

APRM 3 _____%

APRM 4 _____% _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.1 (Continued)

d. **DETERMINE** average APRM value by summing operable APRM readings from above,
THEN DIVIDE that value by number of operable APRMs
AND RECORD value below:

Average APRM Value _____% _____/_____

_____/_____
IV

e. **RECORD** Operable IRM Readings **AND** Range below:

- IRM Ranges are obtained at P603.
- IRM Readings are obtained at P606 **AND** P633.
- IRMs may be Ranged down to Range 9 to obtain value.

	Reading	Range		Reading	Range	
IRM A	_____	_____	IRM B	_____	_____	
IRM C	_____	_____	IRM D	_____	_____	
IRM E	_____	_____	IRM F	_____	_____	
IRM G	_____	_____	IRM H	_____	_____	_____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.1 (Continued)

f. Using Attachment 2, IRM/APRM Overlap Verification,
COMPARE Average APRM value determined in Step G.2.1.d with operable IRM readings from Step G.2.1.e
AND DETERMINE IF IRM readings are within the acceptable overlap limits for average APRM value:

Within Required Limits

IRM A	_____ Yes	_____ No	
IRM C	_____ Yes	_____ No	
IRM E	_____ Yes	_____ No	
IRM G	_____ Yes	_____ No	
IRM B	_____ Yes	_____ No	
IRM D	_____ Yes	_____ No	
IRM F	_____ Yes	_____ No	
IRM H	_____ Yes	_____ No	_____/_____

g. **IF** any operable IRM is determined to be outside of required overlap limits,
THEN NOTIFY Reactor Engineering to assess IRM indication. (Problem with control rod pattern **OR** valid IRM problem),
OTHERWISE MARK this step N/A. _____/_____

CAUTION

Observe Control Rod withdraw limitations of Precaution and limitation D.35.0 if the following step is performed.

h. **IF** control rod manipulations are required,
THEN ADJUST control rod pattern using an approved Reactivity Maneuver Form,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.1 (Continued)

i. **IF** any operable IRM remains outside of required overlap limits,
THEN NOTIFY SM **AND** take Action required by Tech Spec 3.3.1.1,
OTHERWISE MARK this step N/A. _____/_____.

CAUTION

- Do **not** reduce power below the level that would cause the Turbine Bypass Valves to close. This provides for better control of pressure and power.
- If RDS Backfill Injection has been secured to one or more RPV level reference legs, initiation of an RPV cooldown requires the guidance of Attachment 1 to this procedure be followed (see Step G.2.2.b) in order that proper steps be taken should "notching" be observed in any RPV level instrumentation. **[C7]**

G.2.2. **PREPARATION** for placing the Mode Switch to Shutdown.

a. **INITIATE** N2-OSP-RCS-@001 to monitor cooldown rate. _____/_____

b. **IF** RDS Backfill Injection has been secured to one **OR** more RPV level reference legs,
THEN initiate Attachment 1 of this procedure,
OTHERWISE MARK this step N/A. **[C7]** _____/_____

c. **VERIFY** N2-OSP-LOG-S@ALL has been completed. _____/_____

d. **IF** Scram Time Testing is required,
THEN NOTIFY Reactor Engineering,
OTHERWISE MARK this step N/A. _____/_____

e. **IF** desired to allow using Main Steam Drains in the event cooldown rate becomes excessive **AND** MSIV closure is required,
THEN PREPARE Main Steam Line Drains as follows,
OTHERWISE MARK these steps N/A. _____/_____

1) **VERIFY** 2EHS*MCC102A-7A power supply to 2MSS*MOV112 energized. _____/_____

2) At 2EHS*MCC102A-7A,
VERIFY ALARM CIRCUIT control switch to ENABLE. _____/_____

3) **VERIFY** Mechanical Vacuum pumps available for service to maintain Main Condenser vacuum in event it becomes necessary to close MSIVs to control cooldown. **[C8]** _____/_____

f. **TRANSFER** Reboiler Steam Supply from Main Steam to Auxiliary Boiler Steam per N2-OP-25, Subsection F.4.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.2 (Continued)

g. **SHUT DOWN** HWC system by performing the following:

1) At 2CEC-PNL842, **PLACE** HWC SYSTEM SHUTDOWN switch to SHUTDOWN.

2) **DIRECT** operators to **CLOSE** the following valves to prevent leakage past injection check valves:

- 2CNM-V62A..... ☐
- 2CNM-V62B..... ☐
- 2CNM-V62C..... ☐ _____/_____

NOTE

Establishing a nominal reject flow in the following step will facilitate level control following soft scram.

h. **IF** required,
THEN ESTABLISH WCS reject to Main Condenser per N2-OP-37, Subsection F.6.8.1 through F.6.14,
OTHERWISE MARK this step N/A. _____/_____.

i. **IF <u>not</u>** already performed,
THEN VERIFY ALL Mode 3 required surveillances are completed **OR** in progress. _____/_____
US

_____/_____
SM IV

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

CAUTION

- In the following step, all equipment for the feedwater pump in operation is to be left in the normal positions.
- Seat leakage problems have been experienced with the 2FWS-LV10s in the past. After valve repair, the leakage is eradicated or minimized but will degrade as the valve wears. If leakage past the LV10s has been observed to be degrading, caution needs to be observed when at reduced reactor power and pressures. Leakage past the LV10s can cause level and associated power excursions.

G.2.3. To prevent post-scram water level from reaching Level 8, **VERIFY** the following for feed pumps **not** in operation:

- 2FWS-LV10A-CLOSED ..... ☐
- 2FWS-LV10B-CLOSED ..... ☐
- 2FWS-LV10C-CLOSED ..... ☐
- 2FWS-MOV47A-CLOSED ..... ☐
- 2FWS-MOV47B-CLOSED ..... ☐
- 2FWS-MOV47C-CLOSED ..... ☐
- 2FWS-LV55A-CLOSED ..... ☐
- 2FWS-LV55B-CLOSED ..... ☐
- 2FWS-V103A-CLOSED ..... ☐
- 2FWS-V103B-CLOSED ..... ☐ _____/_____

G.2.4. **VERIFY** one of the following valves is controlling level in automatic:

- 2FWS-LV10A..... ☐
- 2FWS-LV10B..... ☐
- 2FWS-LV10C..... ☐
- 2FWS-LV55A..... ☐
- 2FWS-LV55B..... ☐ _____/_____

G.2.5. **VERIFY** Operator responsible for feedwater level control has reviewed Precautions and Limitations for feedwater level control D.31 & D.32. _____/_____

G.2.6. **VERIFY** ALL IRM range selector switches on Range 9 OR 10. _____/_____

G.2.7. **VERIFY** inserted ALL IRM detectors. _____/_____

G.2.8. **PLACE** 8 IRM/APRM **AND** IRM/RBM recorder selector switches to IRM. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.9. **INSERT** SRMs as necessary to **MAINTAIN a** count rate between 100 **AND** 10^5 cps per N2-OP-92, Section G.4.0. _____/_____

G.2.10. **VERIFY** one IRM recorder in EACH division in fast speed. _____/_____

NOTE

- As reactor cooldown occurs, positive reactivity is added if the moderator temperature coefficient is negative (reference Cycle Management Report).
- Performance of the following step may constitute a planned evolution. As such, planned entry into the associated EOPs/SOPs may be anticipated.
- To prevent over feeding the Reactor immediately following the scram, it may be necessary to place feedwater level control in MANUAL. This will require continuous monitoring of Reactor Water level.
- If Reactor Water level is being maintained by 2CNM-LV137, monitor level closely due to valve leak-by. **[C18]**
- Minimize time between pressing RPS pushbuttons and placing the Mode Switch in Shutdown, to prevent inadvertent MSIV closure on depressurization.

CAUTION

If sufficient rods are **not** inserted or rod insertion is interrupted, then a power increase from the cooldown can occur. Minimize the duration of hold points below the point of adding heat. In theory, rod motion should be continuous from insertion of IRMs or SRMs until All Rods In.

G.2.11. **ARM AND DEPRESS** the following:

- REACTOR SCRAM A1 ..... ☐
- REACTOR SCRAM B1 ..... ☐ _____/_____

G.2.12. **PLACE** Reactor Mode Switch to SHUTDOWN position. _____/_____

G.2.13. **IF** Reactor Protection System has **not** tripped,
THEN ENTER N2-SOP-101C,
OTHERWISE MARK this step N/A. _____/_____

G.2.14. **IF** Reactor is **not** scrammed,
THEN REFER to Emergency Operating Procedures,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.15. **VERIFY** ALL rods are fully inserted by one or more of the following:

- On 2CEC*PNL603, at full core display,
  VERIFY ALL Control Rod green lights, indicating ALL Rods Full-in..... ☐
- Process Computer Control Rod Position display indicates ALL rods fully inserted..... ☐
- On 2CEC*PNL603, **VERIFY** Four Rod display indicates 00. ..... ☐
- On 2CEC*PNL603, Rod Worth Minimizer, confirm shutdown softkey.... ☐ _____/_____

G.2.16. **VERIFY** Reactor power lowering on IRMs **AND** SRMs. _____/_____

G.2.17. **MONITOR AND CONTROL** Reactor level as follows:

a. **IF** required,
THEN PLACE Feedwater Level Control Valve(s) in MANUAL
AND CONTINUOUSLY MONITOR Reactor water level
UNTIL AUTO level control is restored,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.17 (Continued)

NOTE

2FWS-LV10A(B) will control in parallel with 2FWS-LV55A(B) **UNTIL** the MASTER CONTROLLER demand "OUT" is less than ~ 1.5 %, **THEN** 2FWS-LV10A(B) should be full closed **AND** 2FWS-LV55A(B) < 60% open.

b. **VERIFY** Reactor level is being maintained between 160" and 187" using ANY combination of the following:

- Condensate and Feedwater (FWS-LV10A(B,C) initially in Automatic):

  a) **WHEN** a rising level trend is noted, **THEN ADJUST** setpoint on 2FWS-HIC1600, MASTER M/A STATION Setpoint "SP" using RAISE/LOWER (↑↓) pushbuttons to 165" **WHILE** remaining in automatic..... ☐

  b) **IF** reactor water level continues to rise above 2FWS-HIC1600, MASTER M/A STATION setpoint, **THEN PLACE** 2FWS-HIC1010A(B,C), 2FWS-LV10A(B,C) CONTROLLER in manual by **DEPRESSING** Manual (M) pushbutton on controller **AND CLOSE** 2FWS-LV10A(B,C) using CLOSE (▼) pushbutton, **OTHERWISE MARK** this step N/A..... ☐

  c) **IF** in use, **THEN VERIFY** 2FWS-LV55A(B) in Auto (A) **OR PLACE** in automatic by **DEPRESSING** Auto (A) pushbutton on 2FWS-LIK1055A(B), 2FWS-LV55A(B) CONTROLLER, **OTHERWISE MARK** this step N/A..... ☐

  d) **IF** 2FWS-P1C is **not** in operation, **THEN WHEN** a rising level trend is noted, **CLOSE** 2FWS-MOV47A(B), **OTHERWISE MARK** this step N/A..... ☐

- Reactor Core Isolation Cooling..... ☐
- High Pressure Core Spray..... ☐
- Control Rod Drive Pumps..... ☐
- WCS Reject to Main Condenser..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.17 (Continued)

c. **WHEN** cause of scram is determined,
IF conditions permit,
THEN RESET scram as soon as possible per Subsection G.3.5 through G.3.12,
OR
VERIFY ALL control rods fully inserted,
THEN CLOSE 2RDS-V28, CHARGING WATER HEADER ISOL. _____/_____

d. **WHEN** reactor water level is stable,
AND at direction of US,
IF using 2FWS-LV55A(B) for level control,
THEN SLOWLY RAISE level to normal 178" to 187" band by raising 2FWS-HIC1600, MASTER M/A STATION Setpoint (SP) with 2FWS-LV55A(B) in automatic,
OTHERWISE MARK this step N/A. _____/_____

e. **VERIFY** Reactor level is being maintained between 178" and 187" using ANY combination of the following:

- Condensate and Feedwater ..... ☐
- Reactor Core Isolation Cooling ..... ☐
- High pressure Core Spray ..... ☐
- Control Rod Drive Pumps ..... ☐
- WCS Reject to Main Condenser ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.17 (Continued)

CAUTION

Differential pressure in excess of 325 psid will preclude re-opening of 2FWS*MOV21A,B if valves are closed to control RPV level. (Ref: CR-1996-002962)

f. **IF** RPV level is rising **AND** can **not** be maintained below 187",
THEN VERIFY one or more of the following to reduce feedwater flow **UNTIL** level is stabilized,
OTHERWISE MARK these steps N/A:

- 2FWS-LV10A CLOSED ..... ☐
  2FWS-LV10B CLOSED ..... ☐
  2FWS-LV10C CLOSED ..... ☐
- 2FWS-MOV47A CLOSED ..... ☐
  2FWS-MOV47B CLOSED ..... ☐
  2FWS-MOV47C CLOSED ..... ☐
- 2FWS-LV55A CLOSED ..... ☐
  2FWS-LV55B CLOSED ..... ☐
- 2FWS-V103A CLOSED ..... ☐
  2FWS-V103B CLOSED ..... ☐
- 2FWS*MOV21A CLOSED ..... ☐
  2FWS*MOV21B CLOSED ..... ☐
- WCS Reject to Main Condenser per N2-OP-37 ..... ☐
- **WHEN** cause of the scram is determined,
  IF conditions permit,
  THEN reset scram as soon as possible per Subsection G.3.5 through G.3.12. ..... ☐
- **VERIFY** ALL control rods fully inserted,
  THEN CLOSE 2RDS-V28. ..... ☐ _____/_____

g. **IF** RPV level continues to rise,
THEN PERFORM the following **UNTIL** level is stabilized,
OTHERWISE MARK these steps N/A:

1) **VERIFY** Feedwater Pump tripped. _____/_____

2) **VERIFY** CSH is **not** injecting to Reactor. _____/_____

3) **VERIFY** ICS is **not** injecting to Reactor. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.17 (Continued)

CAUTION

To minimize thermal stress to the Feedwater nozzle, hot shutdown condition without Condensate Flows feeding the Reactor Vessel should be avoided.

h. **IF** RPV level can **not** be maintained below Level 8 (<202.3")
AND RPV pressure drops below discharge pressure of Condensate Booster Pumps,
THEN trip Condensate Booster Pumps,
OTHERWISE MARK this step N/A. _____/_____

CAUTION

If water enters the Main Steam lines and Turbine Bypass Valves remain open, valve damage may result.

i. **IF** RPV level can **not** be maintained below 250" level,
THEN CLOSE ALL MSIVs,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.17 (Continued)

CAUTION

- 2MSS*MOV111 and 112, are isolation valves in the steam line drains upstream of the Main Steam Isolation Valves (MSIVs). Because the closure times of these valves is longer than MSIVs closure time, the duration that 2MSS*MOV111 and 112 are open shall be minimized. Minimizing the time that these valves are open will ensure we remain within the bounds of current isolation time analysis. The total allowable time that valves may be open during a given year is less than 135 hours, when Primary Containment Isolation function is required (reference Calculation Dispositions No. ES-177-05E and ES-109-01B). **[C15]**
- When the Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig, the valves are subjected to a differential pressure greater than the engineering defined limit. Therefore, the valves must be declared INOPERABLE and Technical Specification 3.6.1.3, Condition B entered. Once entered, the associated action is to isolate the penetration with at least one valve within one hour (Reference Calculation Disposition No. A10.2-O-026).

j. **IF** RPV level reaches 255",
THEN use MSL drains to aid in lowering level by opening the following valves,
OTHERWISE MARK these steps N/A:

1) 2MSS*MOV111 _____/_____

2) 2MSS*MOV112 _____/_____

3) **THROTTLE OPEN** 2MSS*MOV207 for 2 to 6 seconds. _____/_____

4) **AFTER** 2 minutes,
THROTTLE OPEN 2MSS-MOV207 as needed for level control. _____/_____

k. On 2CEC*PNL603,
VERIFY Scram Discharge Volume Vent and Drain valves closed. _____/_____

l. **WHEN** RPV level is stabilized **AND** control established,
THEN DOCUMENT valves left out of normal position in Step G.3.45 of this procedure. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.18. **MONITOR AND CONTROL** reactor pressure as follows:

a. **IF** in use,
THEN VERIFY Reactor pressure is being maintained by Main Turb Bypass Valves (TBV),
OTHERWISE MARK these steps N/A.

1) **VERIFY** Condenser Neck Spray in operation per N2-OP-3, Subsection F.7.0. _____/_____

2) **USE** EHC Pressure Set **OR** Bypass Opening Jack to control reactor pressure with TBVs. _____/_____

b. **IF** MSIVs are OPEN,
AND steam line drains are required for pressure control,
THEN use Attachment 4 to control reactor pressure,
OTHERWISE MARK this step N/A. _____/_____

c. **IF** MSIVs are closed,
THEN control reactor pressure using Safety Relief Valves,
AND PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **MONITOR** Suppression Pool temperature..... ☐

2) As required, **PLACE** Suppression Pool Cooling into service... ☐ _____/_____

d. **IF** MSIVs are CLOSED
AND it is desired to open them,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

NOTE

Allowing the EHC system to control pressure with water in the main steam lines could result in Turbine Bypass valve discharge pipe damage.

1) **IF** MSIVs were closed due to high reactor water level,
THEN DRAIN main steam lines
BEFORE opening MSIVs,
OTHERWISE MARK this step N/A. _____/_____

2) **ATTEMPT** to equalize pressure
AND OPEN MSIVs per N2-OP-1, Subsection F.2.0,
to regain pressure control with Turbine Bypass Valves. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.2.19. **REMOVE** Oxygen in Feedwater Injection System (OFI) from service per N2-OP-3, Subsection G.2.0. _____/_____

G.2.20. **VERIFY** HWC tripped to shutdown condition
AND PERFORM N2-OP-112, Subsection H.1.0. _____/_____

NOTE

- The Reactor Mode Switch shall remain in the Shutdown Position until the SRMs are verified operable and the Neutron Monitoring Surveillances are current.
- Following the Reactor Scram, if any SRV has lifted, surveillance N2-OSP-ISC-M@002 must be completed within 12 hours (SR 3.6.1.7.2). **[C11]**
- It is preferable to control the plant in a manner that maintains the condenser available for decay heat removal until after Shutdown Cooling has been placed in service. **[C28]**

CAUTION

- Operation of the Cooling Tower in the wrong mode during sub-freezing temperatures can result in excessive cooling of Condensate / Feedwater.
- Failure to closely monitor Reactor Water level following a Scram can lead to overfilling the Reactor. **[C16]**

G.3. **Post Reactor Shutdown/Scram Recovery [SOP]**

G.3.1. **LOG** RPV pressure and coolant temperature per N2-OSP-RCS-@001 every 30 minutes **UNTIL** cold shutdown is reached. _____/_____

G.3.2. During the first hour following Reactor scram, **MAINTAIN** pressure greater than or equal to 410 psig to assure excessive cooldown is **not** occurring. _____/_____

G.3.3. **UPDATE** SPDS per N2-OP-91B, Subsection F.2.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.4. At 2CEC*PNL608, **VERIFY** the following on APRM 1-4: **[C19]**

a. SETDOWN is displayed:

- APRM 1..... ☐
- APRM 2..... ☐
- APRM 3..... ☐
- APRM 4..... ☐ _____/_____

b. The double upward pointing triangles on FLUX bar graph are aligned to approximately 15%:

- APRM 1..... ☐
- APRM 2..... ☐
- APRM 3..... ☐
- APRM 4..... ☐ _____/_____

c. The double upward pointing triangles on STP bar graph are aligned to approximately 0.55 Wd +57.5%, with a maximum of 113.5%. (Single Loop Operation 0.52 (Wd) + 46.8%):

- APRM 1..... ☐
- APRM 2..... ☐
- APRM 3..... ☐
- APRM 4..... ☐ _____/_____

NOTE

- The Scram Discharge Volume drains to the 2DER-TK2A when the scram is reset. The volume of water drained may be sufficient to overflow 2DER-TK2A, depending on initial tank level and pump availability.
- Scram Discharge Volume High Level Bypass Switches must be placed in BYPASS prior to resetting any scram or an additional scram may occur.

G.3.5. **NOTIFY** Radwaste to operate available pump(s) for 2DER-TK2A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.6. **PLACE** Scram Discharge Volume High Level Bypass switches on 2CEC*PNL603 to BYPASS:

- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL A1 ..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL A2 ..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL B1 ..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL B2 ..... ☐ _____/_____

G.3.7. **VERIFY** 2RDS-V28, CHARGING WATER HEADER ISOL open. _____/_____

G.3.8. **RESTORE** 2GMC-TV101 handwheel to neutral position per N2-OP-26. _____/_____

G.3.9. **BEFORE** resetting scram,
IF ARI has been initiated by RRCS,
THEN RESET ARI per N2-OP-36B Subsection H.3.0,
OTHERWISE MARK this step N/A. _____/_____

G.3.10. At 2CEC*PNL603, **RESET** scram using Scram Reset Switches
AND VERIFY eight RPS scram pilot valve solenoids white lights are illuminated. _____/_____

G.3.11. **VERIFY** ALL Control Rods have settled from overtravel position to position 00. **[C12]** _____/_____

a. **IF** any Control Rod has **not** settled to position 00,
THEN reseat to position 00 by applying a rod insert signal as necessary to achieve position 00,
OTHERWISE MARK this step N/A. _____/_____

b. **IF** any Control Rod can **not** be reseated to position 00,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **NOTIFY** SM of control rods with this condition. _____/_____

2) **DISARM** control rod(s) per N2-OP-30, Subsection F.13.0. _____/_____

3) **NOTIFY** System Engineer for assistance. _____/_____

G.3.12. At 2CEC*PNL603, **VERIFY** Scram Discharge Volume Vent and Drain Valves indicate open by the following indications:

- Annunciator 603433 Clear ..... ☐
- Annunciator 603434 Clear ..... ☐
- Annunciator 603435 Clear ..... ☐
- Annunciator 603436 Clear ..... ☐
- Open (Red Lights ON) at P603 ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.13. **IF** scheduled,
THEN PERFORM N2-OSP-RMC-R001 REACTOR MODE SWITCH SHUTDOWN POSITION ROD BLOCK CHANNEL FUNCTIONAL TEST,
OTHERWISE MARK this step N/A. _____/_____

CAUTION

In Operational Mode 2 and 3, the Condenser Low Vacuum bypass switches must all be placed in NORMAL before any or all TSVs are open. **[C17]**

G.3.14. At H13-P609, **PLACE** the following keylock switches to BYPASS position:

- B22H-S24A, CNSR LOW VAC BYPASS ..... ☐
- B22H-S24C, CNSR LOW VAC BYPASS ..... ☐ _____/_____

G.3.15. At 2CEC*PNL603, **VERIFY** annunciator window 603129, DIVISION I CONDENSER VACUUM LOW BYPASSED, is illuminated. _____/_____

G.3.16. At H13-P611, **PLACE** the following keylock switches to BYPASS:

- B22H-S24B, CNSR LOW VAC BYPASS ..... ☐
- B22H-S24D, CNSR LOW VAC BYPASS ..... ☐ _____/_____

G.3.17. At 2CEC*PNL603, **VERIFY** annunciator window 603429, DIVISION II CONDENSER VACUUM LO-LO BYPASSED, is illuminated. _____/_____

G.3.18. At Panel 2CEC*PNL603, **VERIFY** the following alarms clear:

- RPS A DISCH VOLUME HIGH LEVEL TRIP window 603109..... ☐
- RPS B DISCH VOLUME HIGH LEVEL TRIP window 603409..... ☐
- SDV LEVEL HIGH window 603130..... ☐ _____/_____

G.3.19. On 2CEC*PNL603, **PLACE** Scram Discharge Volume High Level Bypass switches to NORMAL:

- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL A1..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL A2..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL B1..... ☐
- DISCH VOLUME HI WTR LEVEL BYPASS CHANNEL B2..... ☐ _____/_____

G.3.20. **CHECK** pressure trend from 2ISC*PR1623A/B (PAM recorders on P601) **AND** SRV tailpiece temperatures from Auto Depress/Safety Valve Temperature recorder at 2CEC*PNL614,
to determine if any SRV opened during scram. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.21. **IF** an SRV did lift
OR IF there is any question proving it did **not** lift,
THEN PERFORM N2-OSP-ISC-M@002, Drywell Vacuum Breaker Operability Test, within 12 hours of the lift,
OTHERWISE MARK this step N/A. **[TS]** _____/_____

> **NOTE**
>
> If during the following sequence of steps it is necessary to close the MSIVs and the shutdown is in preparation for a refueling outage when MSIV leak rate testing will be performed, then fast closure of the MSIVs will be required.

G.3.22. **IF** excessive cooldown occurs,
THEN PERFORM the following steps, which are listed in order of lower to higher impact,
WHILE being conscious that the order of performance will be based on plant conditions **AND** rate of cooldown,
OTHERWISE MARK these steps N/A: **[C1], [C2]**

a. At 2CEC-PNL824, **VERIFY** the following Steam Line Drains closed:

- 2MSS-AOV87A ..... ☐
- 2MSS-AOV87B ..... ☐
- 2MSS-AOV87C ..... ☐
- 2MSS-AOV87D ..... ☐
- 2MSS-AOV88A ..... ☐
- 2MSS-AOV88B ..... ☐ _____/_____

b. **IF** excessive cooldown continues to occur,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **BEFORE** startup of Mechanical Vacuum pump,
VERIFY **no** indication of fuel damage by:

- MSL rad monitors downscale ..... ☐
- 2OFG-RE13A **AND** B rad levels are below the ALERT (yellow) alarm if in service..... ☐ _____/_____

2) **CLOSE** 2ASS-MOV152, Main Steam Supply to Steam Jet Air Ejector. _____/_____

3) **START** Mechanical Vacuum pump per N2-OP-9, Subsection G.1.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.22 (Continued)

c. **IF** excessive cooldown continues,
THEN CLOSE 2ASS-MOV148,
OTHERWISE MARK this step N/A. _____/_____

NOTE

It is preferred to slow close the MSIVs. Fast closure may be necessary to control cooldown rate or support MSIV leak rate testing during the shutdown period. Refer to N2-OP-1, subsection G.2.0, for slow closure.

d. **IF** required (cooldown rate remains excessive),
THEN CLOSE MSIVs (preferably the outboards),
OTHERWISE MARK this step N/A. **[C4]** _____/_____

G.3.23. **ROTATE** the following switch collars to DISARM:

- REACTOR SCRAM A1 ..... ☐
- REACTOR SCRAM B1 ..... ☐
- REACTOR SCRAM A2 ..... ☐
- REACTOR SCRAM B2 ..... ☐ _____/_____

G.3.24. **IF** an RRCS initiation has occurred,
WAIT UNTIL "RRCS READY TO RESET" indicator illuminates,
THEN reset RRCS per N2-OP-36B, Subsection H.3.0,
OTHERWISE MARK this step N/A. _____/_____

G.3.25. **NOTIFY** Reactor Engineering to perform OP-AA-108-114, Post Transient Review, **AND** to obtain reactor startup information from Fuels group for subsequent reactor startup if required,
OTHERWISE MARK this step N/A. _____/_____

NOTE

N2-ISP-NMS-M@008 is the SRM Channel Functional Test. SRMs are required to be operable in Modes 3, 4 and 5.

G.3.26. **IF** required **AND <u>not</u>** performed,
THEN PERFORM N2-ISP-NMS-M@008
WITHIN 12 hours **AFTER** IRMs are on Range 2 **OR** below,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27. Based on reason for shutdown **AND** instruction from Shift Manager, **PERFORM** applicable steps below:

NOTE

- Setpoint Setdown will automatically reset if water level remains at 175 inches +5"/-5" for 5 minutes.
- The transition from HIGH/LOW power modes is automatic at ~ 15% rated feed flow.

CAUTION

To minimize thermal stress to the feedwater nozzle, avoid hot shutdown condition without Condensate Booster pump feeding to the Reactor. **[C30]**

a. **RESET** Setpoint Setdown **AND VERIFY** Feedwater level control is in LOW POWER MODE as necessary to control Reactor water level per N2 OP 3, Subsections H.1.0 **AND** G.1.5. _____/_____

b. **START** Mechanical Vacuum Pump per N2-OP-9, Subsection G.1.0, to maintain Condenser vacuum. _____/_____

c. **SHUT DOWN** the SJAEs per N2-OP-9, Sub-section G.1.0. _____/_____

d. **IF** MSIVs are open
AND cooldown is desired
AND Main Condenser is available,
THEN BEGIN Reactor cooldown (refer to Steps G.3, G.3.2 of this procedure) using Auto Rate per Attachment 7,
OTHERWISE MARK this step N/A. _____/_____

e. **IF** cooldown is **not** to be performed at this time
AND reactor pressure **AND** temperature is to be maintained,
THEN PERFORM ATTACHMENT 4,
OTHERWISE MARK this step N/A. _____/_____

f. **IF** MSIVs are isolated **AND** Main Condenser is available,
AND MSIVs can be opened,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

1) **EQUALIZE** the Main Steam Line pressure around MSIVs per N2-OP-1, Subsection F.2.0. _____/_____

2) **OPEN** MSIVs. _____/_____

3) **BEGIN COOLDOWN** using Auto Rate per Attachment 7. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

g. **IF** MSIVs are to remain isolated
AND Main Condenser is available,
THEN PERFORM the following to control Reactor cooldown,
OTHERWISE MARK these steps N/A.

CAUTION

- 2MSS*MOV111, 112 and 208 are isolation valves in the Steam Line drains between or upstream of the Main Steam Isolation Valves (MSIVs). Because the closure times of these valves is longer than MSIV closure times, the duration that 2MSS*MOV111, 112 or 208 is open shall be minimized. Minimizing the time that these valves are open will ensure we remain within the bounds of current isolation time analysis. The total allowable time that valves may be open during a given year is less than 135 hours, when Primary Containment Isolation function is required (reference Calculation Dispositions No. ES-177-05E and ES-109-01B). **[C15]**
- When the Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig, the valves are subjected to a differential pressure greater than the engineering defined limit. Therefore, the valves must be declared INOPERABLE and Technical Specification 3.6.1.3, Condition B entered. Once entered, the associated action is to isolate the penetration with at least one valve within one hour (reference Calculation Disposition No. A10.2-O-026).

1) **OPEN** 2MSS*MOV111. _____/_____

2) **OPEN** 2MSS*MOV112. _____/_____

3) **THROTTLE OPEN** 2MSS*MOV207 by
HOLDING control switch OPEN for 2 to 6 seconds. _____/_____

4) **AFTER** 2 minutes,
THROTTLE OPEN 2MSS*MOV207 as required to control cooldown rate. _____/_____

5) **IF** 2MSS*MOV207 is full open
AND Reactor pressure continues to rise,
THEN THROTTLE OPEN 2MSS-MOV187. _____/_____

6) **OPEN** 2MSS-AOV87A, B, C, D as required to control cooldown rate. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

h. **IF** Main Condenser is **not** available **BEFORE** or during cooldown, **THEN** control Reactor level and pressure and cool down using one or more of the following (Refer to Steps G.3 AND G.3.2), **OTHERWISE MARK** these steps N/A:

1) Reactor pressure control using SRV actuation per N2-OP-34, Subsection H.2.0 _____/_____

WITH

2) Injection to reactor vessel using:

- RCIC per N2-OP-35 Subsection F.3.0 OR F.4.0 ..... ☐
- Condensate and Feedwater per N2-OP-3 ..... ☐
- HPCS per N2-OP-33, Subsection H.9.0 ..... ☐ _____/_____

3) As required, **PLACE** one RHR loop in Suppression Pool Cooling mode per N2-OP-31, Subsection F.4.0. _____/_____

i. **IF** ambient air temperature is less than 50°F, **THEN TRANSFER** Cooling Tower mode of operation per N2-OP-10A, **OTHERWISE MARK** this step N/A. _____/_____

j. **IF** required, **THEN** secure Service Water makeup to CWS, **OTHERWISE MARK** this step N/A. _____/_____

k. **CONTINUE COOLDOWN** by steam removal using one **OR** more of the following:

- EHC Auto Rate per Attachment 7 ..... ☐
- EHC Pressure Setpoint controller ..... ☐
- Bypass Valve Opening Jack ..... ☐
- RCIC ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

CAUTION

If RCS Blocking Valve seal-in has **not** been defeated per S-EMP-VLV-001, movement of RCS Blocking Valves off their backseat could result in trip of Reactor Recirculation Pump.

l. **IF** 2RCS*MOV18A was backseated,
THEN take valve off backseat as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** applicable Attachments of S-EMP-VLV-001 have been performed **BEFORE** taking RCS Blocking Valves off their backseat. _____/_____

2) At P602, **HOLD** 2RCS*MOV18A control switch to CLOSE **UNTIL** dual indication is received, **THEN** release. _____/_____

3) At P602, **PLACE** 2RCS*MOV18A control switch to OPEN, **THEN** release. _____/_____

4) **VERIFY** 2RCS*MOV18A indicates open. _____/_____

m. **IF** 2RCS*MOV18B was backseated,
THEN take valve off backseat as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** applicable Attachments of S-EMP-VLV-001 have been performed **BEFORE** taking RCS Blocking Valves off their backseat. _____/_____

2) At P602, **HOLD** 2RCS*MOV18B control switch to CLOSE **UNTIL** dual indication is received, **THEN** release. _____/_____

3) At P602, **PLACE** 2RCS*MOV18B control switch to OPEN, **THEN** release. _____/_____

4) **VERIFY** 2RCS*MOV18B indicates open. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

n. **IF** 2RCS*MOV10A was backseated,
THEN take valve off backseat as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** applicable Attachments of S-EMP-VLV-001 have been performed **BEFORE** taking RCS Blocking Valves off their backseat. _____/_____

2) At P602, **HOLD** 2RCS*MOV10A control switch to CLOSE **UNTIL** dual indication is received, **THEN** release _____/_____

3) At P602, **PLACE** 2RCS*MOV10A control switch to OPEN, **THEN** release. _____/_____

4) **VERIFY** 2RCS*MOV10A indicates open. _____/_____

o. **IF** 2RCS*MOV10B was backseated,
THEN take valve off backseat as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** applicable Attachments of S-EMP-VLV-001 have been performed **BEFORE** taking RCS Blocking Valves off their backseat. _____/_____

2) At P602, **HOLD** 2RCS*MOV10B control switch to CLOSE **UNTIL** dual indication is received, **THEN** release. _____/_____

3) At P602, **PLACE** 2RCS*MOV10B control switch to OPEN, **THEN** release. _____/_____

4) **VERIFY** 2RCS*MOV10B indicates open. _____/_____

p. At approximately 500 psig, **SECURE** remaining Feedwater Pump per N2-OP-3, subsection G.1.11. _____/_____

NOTE

Multiple methods of monitoring cooldown rates should be used to prevent exceeding cooldown limits.

q. **REMOVE** from service
ONE of two operating Condensate **AND** Condensate Booster pumps per N2-OP-3, Subsection G.1.12 **AND** G.1.13. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

NOTE

- Raising Reactor water level to the band of 227 to 243" will support natural circulation if Reactor Recirculation Pumps are secured or will be secured.
- In addition, the level and temperature transient when Shutdown Cooling is initiated will be minimized.
- Industry best practices for vessel flood up have employed a slow fill method for the reactor vessel at the beginning of refuel outages which is effective in reducing Cobalt-60 peak crud burst by approximately 35 percent.
- In the following steps, the RPV water level will be raised by adjusting Reject flow, RDS and Condensate/Feedwater.
- RPV level should rise approximately 1 inch every two to three minutes to equate to approximately 80 gpm.

r. **IF** Reactor water level is **not** to remain in normal band of 178 to 187", and as directed by SM,
THEN BEGIN TO RAISE reactor water Level to target band of 227 to 243" on Shutdown Range Level Indicator,
OTHERWISE MARK this step N/A. _____/_____

s. **IF** Condensate Booster pump will **not** remain in service,
THEN at approximately 128 psig, **SECURE** remaining Condensate Booster pump per N2-OP-3, Subsection G.1.16,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.27 (Continued)

NOTE

- Placing Shutdown Cooling in service at the highest temperature and pressure possible, minimizes the release of insoluble isotopes from the fuel surfaces.
- Within 2 hours, and every 12 hours thereafter, when Reactor Steam Dome pressure is less than the RHR cut-in permissive pressure, verification of one RHR Shutdown Cooling Subsystem or Recirculation Pump operating is required. (SR 3.4.9.1)

CAUTION

While performing RHS piping warmup and flush for Shut Down Cooling, Reactor cooldown should be minimized to prevent RHS piping from being heated more than Reactor inventory, which will cause water flashing and voiding in the RHS system, and possible Reactor level loss. **[C18]**

t. **PLACE** Shutdown Cooling Loop per N2-OP-31, Residual Heat Removal System. _____/_____

u. **WITHIN** 2 hours **AFTER** Reactor Steam Dome Pressure is less than RHR cut-in permissive pressure,
VERIFY one RHR Shutdown Cooling Subsystem
OR Recirculation Pump is operating. _____/_____

v. **IF** return to Reactor Power Operation (Operation Condition 1) is **not** imminent,
THEN REMOVE RRCS, DIV I **AND** DIV II from service per N2-OP-36B, Section G, SHUTDOWN,
OTHERWISE MARK this step N/A. **[C10]** _____/_____

G.3.28. **WHEN** RCIC is **no** longer needed,
THEN BEFORE reaching 60 psig Rx pressure,
CLOSE 2ICS*MOV150, Trip Throttle Valve, to prevent damage. _____/_____

G.3.29. At approximately 60 psig, **VERIFY** RCIC system isolates. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.30. At approximately 5 psig Rx pressure,
PERFORM the following Supplemental Position Indication Test for 2MSS*MOV118 and 2MSS*MOV119:

a. **INITIATE** procedure N2-OSP-MSS-R001. _____/_____

b. **OPEN** 2MSS*MOV118, Rx Head Vent to Suppression Pool. _____/_____

c. **VERIFY** Alarm 601333 RCIC PIPE CHASE/ REACTOR VENT TEMP HIGH does **not** alarm,
proving closure of 2MSS*MOV119. _____/_____

d. **CLOSE** 2MSS*MOV118, Rx Head Vent to Suppression Pool. _____/_____

e. **OPEN** 2MSS*MOV119, Rx Head Vent to Suppression Pool. _____/_____

f. **VERIFY** Alarm 601333 RCIC PIPE CHASE/ REACTOR VENT TEMP HIGH does **not** alarm,
proving closure of 2MSS*MOV118. _____/_____

g. **OPEN** 2MSS*MOV118, Rx Head Vent to Suppression Pool. _____/_____

h. **VERIFY** Alarm 601333 RCIC PIPE CHASE/ REACTOR VENT TEMP HIGH alarms,
proving 2MSS*MOV118 **AND** 2MSS*MOV119 are open. _____/_____

i. **DOCUMENT** results of Supplemental Position Indication Test for 2MSS*MOV118 and 2MSS*MOV119 in N2-OSP-MSS-R001 remarks. _____/_____

j. **CLOSE** 2MSS*MOV108, Rx Head Vent to Main Steam Line A. _____/_____

k. **IF** open,
THEN CLOSE MSIVs as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** the following Computer Points are in alarm to prevent an RPS trip:

a) RPSBC05 RPS A1 MSIV CLSR TR BYP ..... ☐

b) RPSBC06 RPS A2 MSIV CLSR TR BYP ..... ☐

c) RPSBC07 RPS B1 MSIV CLSR TR BYP ..... ☐

d) RPSBC08 RPS B2 MSIV CLSR TR BYP ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.30.k (Continued)

2) **IF** MSIVs are **not** to be slow closed
OR are **not** already closed
OR MSIV leak rate testing will be performed,
THEN FAST CLOSE MSIVs by placing control switches to CLOSE position,
OTHERWISE MARK this step N/A. _____/_____

3) **IF** MSIVs **not** to be fast closed
OR are **not** already closed,
THEN SLOW CLOSE MSIVs per N2-OP-1, G.2.0,
OTHERWISE MARK this step N/A. _____/_____

G.3.31. **IF** N2-EOPs were used to control plant parameters following the scram,
THEN RESTORE defeated interlocks as applicable using N2-EOP-6 Modification Tracking Sheet **AND** applicable EOP-6 support procedures,
OTHERWISE MARK this step N/A. _____/_____

G.3.32. **IF** scram was from high power,
THEN REFER to N2-OP-3, H.2.0, for restoration of 2CNM-AOV101 and 2CNM-AOV109,
OTHERWISE MARK this step N/A. _____/_____

G.3.33. **RESTORE** ECCS, DG, Containment Isolations, DW Cooling and SRVs as applicable using appropriate operating procedures. _____/_____

G.3.34. **IF** desired **AND** permitted by Technical Specification Table 3.3.6.1-1,
THEN BYPASS Division I **AND/OR** II WCS Delta Flow Isolation Logic per N2-OP-37, Section H.6.0,
OTHERWISE MARK this step N/A. _____/_____

G.3.35. **WHEN** coolant temperature is less than 325°F,
THEN transfer WCS return to feedwater per N2-OP-37. _____/_____

G.3.36. **IF** initiated,
THEN STOP trends initiated in Step G.1.28.b
AND send traces to IMD and Engineering,
OTHERWISE MARK this step N/A. _____/_____

G.3.37. At 2CES-IPNL202, **AFTER** entering Mode 4, **PERFORM** the following:

- **PLACE** 2HVT-UC210A in Normal-After-STOP. ..... ☐
- **PLACE** 2HVT-UC210B in Normal-After-STOP. ..... ☐
- **SHUTDOWN** 2HVT-FN11 **OR** 12..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

CAUTION

Failure to defeat isolation signals in the following Steps may cause spurious isolations to Shutdown Cooling mode of operations, see OE #28140.

G.3.38. **AFTER** entering Mode 4,
IF desired to defeat WCS **OR** RHR **AND** RCIC HIGH AREA TEMPERATURE isolations,
THEN PERFORM N2-OP-83, Attachment 4,
OTHERWISE MARK this step N/A. **[C27]** _____/_____

G.3.39. **AFTER** entering Mode 4,
IF desired to defeat RHR Shutdown Cooling HIGH REACTOR PRESSURE isolations (Group 5),
THEN PERFORM N2-OP-83, Attachment 5,
OTHERWISE MARK this step N/A. **[C27]** _____/_____

G.3.40. **AFTER** entering Mode 4,
IF desired to de-energize RHR Shutdown Cooling Suction Isolation Valves (Group 5),
THEN PERFORM N2-OP-83, Attachment 6,
OTHERWISE MARK this step N/A. _____/_____

CAUTION

Prior to allowing individuals under the vessel, the RPS Scram Functions must be defeated per N2-OP-97. Bumping SRM and IRM cables may cause a half or full scram.

G.3.41. **AFTER** entering Mode 4,
IF desired to defeat RPS Scram Functions,
THEN PERFORM N2-OP-97, Section H.5.0, Defeating RPS Scram Functions,
OTHERWISE MARK this step N/A. _____/_____

G.3.42. **UNLESS** otherwise directed,
MAINTAIN Reactor temperature less than 100°F WHILE shut down. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

The following step is included to prevent unnecessary wear on IAS Booster Skids while the plant is shutdown.

G.3.43. **IF** shutdown is in preparation for a Refueling Outage or extended outage,
THEN shutdown the following Instrument Air Booster Skids,
OTHERWISE MARK this step N/A:

- IAS-SKD1, N2-OP-19, Section G.3.0 ..... ☐
- IAS-SKD2, N2-OP-19, Section G.3.0 ..... ☐
- IAS-SKD3, N2-OP-19, Section G.3.0 ..... ☐
- IAS-SKD4, N2-OP-19, Section G.3.0 ..... ☐
- IAS-SKD5, N2-OP-19, Section G.3.0 ..... ☐
- IAS-SKD6A, N2-OP-19, Section G.5.0 ..... ☐
- IAS-SKD6B, N2-OP-19, Section G.5.0 ..... ☐ _____/_____

G.3.44. **IF** shutting down for Refueling Outage
AND stroking 2ASS-AOV147 is scheduled,
THEN PERFORM N2-OP-9, Attachment 3, for stroking 2ASS-AOV147
AND attach copy to associated work order,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.3.45. **EXPLAIN** all abnormal conditions **OR** lineups relating to shutdown below. **ATTACH** additional pages as needed.

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

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_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

_____ _____/_____

NOTE

- Condenser vacuum must be maintained by SJAE when Reactor power is greater than 5%.
- **VERIFY** IRM/APRM overlap prior to entering Mode 2 from Mode 1. (SR 3.3.1.1.6)
- It is preferable to control the plant in a manner that maintains the condenser available for decay heat removal until after Shutdown Cooling has been placed in service. **[C28]**

G.4. **Reactor Shutdown by Individual Control Rod Insertion**

G.4.1. At approximately 5 to 10% Reactor Power,
PERFORM the following to **VERIFY** proper IRM/APRM overlap: **[T/S]**

a. **VERIFY** ALL IRM range selector switches on Range 10. _____/_____

b. **VERIFY** ALL operable IRM detectors fully inserted per N2-OP-92 Section G.3.0. _____/_____

NOTE

The Acceptance Criteria for IRM/APRM overlap in Step G.4.1.f requires that operable IRMs be on range 9 or 10 and **no** concurrent power changes in progress.

c. **BEFORE** any operable APRM reading falling less than 4% as indicated on IRM/APRM recorders C51-R603A,B,C,D **RECORD** operable APRM readings below:

APRM 1 _____%

APRM 2 _____%

APRM 3 _____%

APRM 4 _____% _____/_____

d. **DETERMINE** average APRM value
by summing operable APRM readings from above,
THEN divide value by number of operable APRMs
AND RECORD value below:

Average APRM Value _____% _____/_____

_____/_____
IV

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.1 (Continued)

NOTE

- IRM Ranges are obtained at P603
- IRM Readings are obtained at P606 AND P633
- IRMs may be Ranged down to Range 9 to obtain value

e. **RECORD** Operable IRM Readings AND Range below:

	Reading	Range		Reading	Range
IRM A	_____	_____	IRM B	_____	_____
IRM C	_____	_____	IRM D	_____	_____
IRM E	_____	_____	IRM F	_____	_____
IRM G	_____	_____	IRM H	_____	_____

_____/_____

f. Using Attachment 2, IRM/APRM Overlap Verification, compare Average APRM value determined in Step G.4.1.d with operable IRM readings from Step G.4.1.e AND determine IF IRM readings are within acceptable overlap limits for average APRM value:

	Within Required Limits	
IRM A	_____ Yes	_____ No
IRM C	_____ Yes	_____ No
IRM E	_____ Yes	_____ No
IRM G	_____ Yes	_____ No
IRM B	_____ Yes	_____ No
IRM D	_____ Yes	_____ No
IRM F	_____ Yes	_____ No
IRM H	_____ Yes	_____ No

_____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.1 (Continued)

g. **IF** any operable IRM in Step G.4.1.f above is determined to be outside of required overlap limits,
THEN NOTIFY Reactor Engineering to assess IRM indication. (Problem with control rod pattern or valid IRM problem),
OTHERWISE MARK this step N/A. _____/_____.

CAUTION

Observe Control Rod withdraw limitations of Precaution and limitation D.34 if the following step is performed.

h. **IF** control rod manipulations are required,
THEN adjust control rod pattern using an approved Reactivity Maneuver Form,
OTHERWISE MARK this step N/A. _____/_____

i. **IF** any operable IRM in Step G.4.1.f above remains outside of required overlap limits,
THEN NOTIFY SM and take Action required by Tech Spec 3.3.1.1,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.2. **PERFORM** the following **BEFORE** entry into Reactor MODE 2:

CAUTION

- When reactor power level reduces below the Point of Adding Heat (POAH), the Turbine Bypass Valves may close, depending on recent power history and steam system valve leakage. This will allow RPV pressure to lower below the EHC pressure setpoint, and will add positive reactivity to the core due to negative temperature coefficient effects.
- If RDS Backfill Injection has been secured to one or more RPV level reference legs, initiation of a RPV cooldown requires the guidance of Attachment 1 to this procedure be followed (see Step G.4.2.a) in order that proper steps be taken should "notching" be observed in any RPV level instrumentation. **[C7]**

a. **IF** RDS Backfill Injection has been secured to one or more RPV level reference legs,
THEN initiate Attachment 1 of this procedure,
OTHERWISE MARK this step N/A. _____/_____

b. **VERIFY** N2-OSP-LOG-S@ALL has been completed. _____/_____

c. **INITIATE** N2-OSP-RCS-@001 to monitor the cooldown rate every 30 minutes **UNTIL** cold shutdown is reached. _____/_____

d. To allow using Main Steam Drains in event cooldown rate becomes excessive **AND** MSIV closure is required,
THEN PREPARE Main Steam Line Drains as follows,
OTHERWISE MARK these steps N/A:

1) **VERIFY** 2EHS*MCC102A-7A power supply to 2MSS*MOV112 energized. _____/_____

2) At 2EHS*MCC102A-7A, **VERIFY** ALARM CIRCUIT control switch to ENABLE. _____/_____

3) **VERIFY** Mechanical Vacuum pumps available for service to maintain Main Condenser vacuum **BEFORE** securing SJAE AND Offgas. **[C8]** _____/_____

NOTE

Establishing a nominal reject flow in the following step will facilitate level control while shutting down the unit.

G.4.3. **IF** WCS is **not** already in Full Flow Reject,
THEN ESTABLISH WCS reject to Main Condenser per N2-OP-37, Subsection F.6.9 through F.6.14,
OTHERWISE MARK this step N/A. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.4. **VERIFY** Operator responsible for Feedwater Level Control has reviewed Precautions and Limitations for feedwater level control D.31 and D.32. _____/_____

CAUTION

- In the following step, ALL equipment for the feedwater pump in operation is to be left in the normal position.
- 2FWS-LV10s have the potential to leak by, causing Reactor level and pressure excursions. Closely monitor plant parameters while operating at reduced reactor power and pressures.

G.4.5. To prevent water level from reaching Level 8 during plant shutdown, **VERIFY** the following for feed pumps **not** in operation:

- 2FWS-LV10A-CLOSED ..... ☐
- 2FWS-LV10B-CLOSED ..... ☐
- 2FWS-LV10C-CLOSED ..... ☐
- 2FWS-MOV47A-CLOSED ..... ☐
- 2FWS-MOV47B-CLOSED ..... ☐
- 2FWS-MOV47C-CLOSED ..... ☐
- 2FWS-LV55A-CLOSED ..... ☐
- 2FWS-LV55B-CLOSED ..... ☐
- 2FWS-V103A-CLOSED ..... ☐
- 2FWS-V103B-CLOSED ..... ☐ _____/_____

NOTE

2FWS-LV10A(B) will control in parallel with 2FWS-LV55A(B) until the MASTER CONTROLLER demand “OUT” is less than ~ 1.5 %, then 2FWS-LV10A(B) should be full closed and 2FWS-LV55A(B) < 60% open.

G.4.6. **WHEN** 2FWS-LV10A(B) is full closed **AND** 2FWS-LV55A(B) is controlling at less than 50% open
THEN place 2FWS-HIC1010A(B), 2FWS-LV10A(B) CONTROLLER in MANUAL (M). _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.7. **VERIFY** one of the following valves is controlling level in automatic in normal 178" to 187" band:

- 2FWS-LV10A..... ☐
- 2FWS-LV10B..... ☐
- 2FWS-LV55A..... ☐
- 2FWS-LV55B..... ☐ _____/_____

NOTE

Leakage past the LV10 can mask whether existing steam loads are truly within the capacity of the LV55. It is recommended that the MOV47 be slowly throttled closed while monitoring LV55 position.

G.4.8. **IF ALL** of the following conditions are met,
THEN CLOSE 2FWS-MOV47A(B) for OPERATING feed pump:

- RPV level is continuing to rise above Setpoint “SP” of 2CNM-LIK1137, 2CNM-LV137 CONTROLLER..... ☐
- 2FWS-LV10A(B) is fully shut..... ☐
- FWS-LV55A(B) is controlling level in automatic ..... ☐ _____/_____

G.4.9. Align nuclear instrumentation for mode change to MODE 2 as follows:

a. **VERIFY** fully inserted ALL IRM detectors **not** bypassed per N2-OP-92, Section G.3.0. _____/_____

b. **POSITION** ALL IRM range selector switches as necessary to maintain a reading between 25 **AND** 75 on 0 to 125% scale. _____/_____

c. **VERIFY** ALL APRM readings are less than or equal to 10%. _____/_____

d. **PLACE** 8 IRM/APRM **AND** IRM/RBM recorder selector switches to IRM. _____/_____

e. **VERIFY** one IRM recorder in EACH division in fast speed. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

- IRM upscale scram is in effect in Reactor Mode 2.
- 603116 and 603416 are expected alarms.

G.4.10. **OBTAIN** permission to enter Reactor MODE 2. _____/_____

_____/_____
SM

CAUTION

In the following step, movement of the Reactor Mode Switch BEYOND the START/HOT STBY position will result in a Reactor Scram.

G.4.11. **PLACE** Reactor Mode Switch in START/HOT STBY position. _____/_____

a. **UPDATE** SPDS per N2-OP-91B. _____/_____

b. At 2CEC*PNL608, **VERIFY** the following on APRM 1 through 4:

- SETDOWN is displayed:

  APRM 1..... ☐

  APRM 2..... ☐

  APRM 3..... ☐

  APRM 4..... ☐ _____/_____

- The double upward pointing triangles on FLUX bargraph are aligned to approximately 15%:

  APRM 1..... ☐

  APRM 2..... ☐

  APRM 3..... ☐

  APRM 4..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.11.b (Continued)

- The double upward pointing triangles on STP bargraph are aligned to approximately 0.55 Wd + 57.5%, with a maximum of less than or equal to 113.5%.
(Single Loop Operation 0.52 Wd+46.8%):

  APRM 1 ..... ☐

  APRM 2 ..... ☐

  APRM 3 ..... ☐

  APRM 4 ..... ☐ _____/_____

NOTE

- IRM surveillances must be completed within 12 hours of the time Reactor Mode 2 is entered if surveillance is **not** current to meet the completion requirement of SR 3.4.1.1.4.
- There is **no** requirement to delay the shutdown until the IRM Channel Functional Test is completed. This test may be performed after all control rods have been inserted.

G.4.12. **PERFORM** N2-ISP-NMS-W@009, IRM CHANNEL FUNCTIONAL TEST. _____/_____

G.4.13. **SHUTDOWN** HWC system by performing the following:

a. At 2CEC-PNL842,
PLACE HWC SYSTEM SHUTDOWN switch to SHUTDOWN. ..... ☐

b. **DIRECT** operators to **CLOSE** the following valves to prevent leakage past injection check valves:

- 2CNM-V62A ..... ☐
- 2CNM-V62B ..... ☐
- 2CNM-V62C ..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.14. **WHEN** reactor power is less than 5%,
THEN REMOVE Steam Jet Air Ejectors (SJAE) from service as follows:

a. At panel H13-P609,
PLACE the following keylock switches to BYPASS:

- B22H-S24A, CNSR LOW VAC BYPASS ..... ☐
- B22H-S24C, CNSR LOW VAC BYPASS ..... ☐ _____/_____

b. At panel 2CEC*PNL603, **VERIFY** annunciator window 603129, DIVISION I CONDENSER VACUUM LOW BYPASSED, is illuminated. _____/_____

c. At panel H13-P611, **PLACE** the following keylock switches to BYPASS:

- B22H-S24B, CNSR LOW VAC BYPASS ..... ☐
- B22H-S24D, CNSR LOW VAC BYPASS ..... ☐ _____/_____

d. At panel 2CEC*PNL603,
VERIFY annunciator window 603429, DIVISION II CONDENSER VACUUM LO-LO BYPASSED, is illuminated. _____/_____

NOTE

In the following steps, the time that the offgas vacuum pump runs with 2ARC-AOV104 closed should be minimized.

e. **START** Mechanical Vacuum pump
AND SHUT DOWN SJAEs per N2-OP-9, Section G.1.0, Securing the ARC System, to maintain Condenser vacuum. _____/_____

f. **SHUT DOWN** Offgas per N2-OP-42. _____/_____

g. **VERIFY** HWC tripped to shutdown condition
AND PERFORM N2-OP-112, Subsection H.1.0. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

- Refrain from using Control Rods to maintain Reactor pressure. The purpose of the following steps is to facilitate the insertion of Control Rods into the core.
- Reactor depressurization/cooldown may be minimized by removing SJAEs and starting a Air Removal Pump.
- Xenon buildup will be adding negative reactivity which will lower Reactor power.
- During certain periods of core life, the Reactor temperature coefficient of reactivity (αT) may be positive. When αT is positive, the cooldown experienced while shutting the unit down will assist in maintaining the unit subcritical.

CAUTION

- Minimize Reactor cooldown until Control Rods are fully inserted.
- As Reactor cooldown occurs, positive reactivity added may be greater than the negative reactivity addition that is being achieved from control rod insertions. If this occurs, it may be necessary to stop rod movements and uprange the IRMs to follow the power increase. Reactor power will stop increasing when the Point of Adding Heat is reached (typically range 7-8 on the IRMs). Inward rod movement could then be recommenced.
- Avoid movement of control rods for controlling reactor pressure and temperature. If control rods are used to control reactor pressure and temperature, all control rod maneuvers should be of a single notch. Sufficient time between rod maneuvers may be needed to allow for reactor pressure and power response.
- Closely monitor Turbine Bypass Valve (TBV) position indication. When all TBVs are shut, Reactor pressure control will be dependent on power history and the integrity of the steam system. Actions may be necessary to limit Reactor cooldown to prevent a restart event. **[C29]**
- Delays in inserting Control Rods should be avoided during controlled shutdowns when power is below that necessary to supply existing steam loads. Control room activities that could interfere with rod insertion (turn-over, briefs, surveillance testing, etc) should be avoided or deferred whenever possible. Control Room Supervisors shall ensure that operators have a clear understanding of plant conditions that could lead to a return to criticality during a shutdown.

G.4.15. Fully **INSERT** ALL control rods as follows:

a. **VERIFY** SRM recorder in fast speed. _____/_____

b. **PERFORM** the following:

- Range IRM Range Switches..... ☐
- **INSERT** SRMs as necessary to maintain a count rate between 100 AND 10^5 cps per N2-OP-92, Section G.4.0 ..... ☐ _____/_____

c. **RECORD** time **WHEN** first IRM is ranged to Range 2:

Time: _____ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.15 (Continued)

NOTE

N2-ISP-NMS-M@008 is the SRM Channel Functional Test. SRMs are required to be operable in Modes 3, 4 and 5.

d. **IF** required **AND not** performed,
THEN PERFORM N2-ISP-NMS-M@008
WITHIN 12 hours **AFTER** IRMs are on Range 2 or below,
OTHERWISE MARK this step N/A. _____/_____

NOTE

- Low Power operations at high reactor pressure has the potential to result in resonant instabilities in EHC Pressure Control. This condition can occur with the Reactor sub-critical, decay heat less than 1% and a single bypass valve controlling reactor pressure less than 30% open. This condition will be manifested by oscillations in TBV position and small oscillations in reactor pressure. This condition has caused indicated oscillations in reactor water level, core plate differential pressure, and intermittent HPCS line break indication a frequency of approximately 4 to 5 Hz. Operation in this condition should **not** be allowed to continue and is mitigated by taking manual control of Rx pressure with the Bypass Jack which will result in a slightly lower reactor pressure to preclude the resonant oscillations (IR 04127454).

- Refrain from using the Turbine Bypass Valve Opening Jack DECREASE pushbutton until all Control Rods are inserted. IRMs will be very sensitive at this power range and may cause a half or full scram, if pressure is increased.

G.4.16. **MONITOR** for the following conditions:

a. **IF** Reactor power rises such that multiple APRM downscales clear due to Reactor Cooldown,
THEN EXIT this procedure
AND ENTER N2-SOP-101C, Reactor Scram,
OTHERWISE MARK this step N/A. _____/_____

b. **IF** EHC/Bypass valve oscillations occur,
OR to prevent oscillations,
THEN TAKE manual control of EHC using Turbine Bypass Valve Opening Jack to stabilize Reactor pressure,
OTHERWISE MARK this step N/A. _____/_____

G.4.17. **PERFORM** Attachment 4 as required to control cooldown rate during control rod insertion. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

NOTE

These actions in the following step are listed in order of greater impact. Perform only those steps necessary in order to limit Reactor cooldown.

G.4.18. **IF** reactor cooldown rates become excessive during control rod insertion,
THEN VERIFY the following valves closed,
OTHERWISE MARK this step N/A.

a. Turbine Bypass Valves. _____/_____

b. Main steam line low point drain valves:

- 2MSS-AOV87A, MSL A LOW POINT DRAIN VALVE ..... ☐
- 2MSS-AOV87B, MSL B LOW POINT DRAIN VALVE ..... ☐
- 2MSS-AOV87C, MSL C LOW POINT DRAIN VAL ..... ☐
- 2MSS-AOV87D, MSL D LOW POINT DRAIN VAL ..... ☐ _____/_____

c. Main Steam Line Drain Header Isolation Valves:

- 2MSS-AOV88A, MSL DRAIN HEADER ISOL VAL ..... ☐
- 2MSS-AOV88B, MSL DRAIN HEADER ISOL VAL ..... ☐ _____/_____

d. 2ASS-MOV148, MAIN STM TO AUX STM ISOL VLV. _____/_____

e. Outboard MSIVs per N2-OP-1, Section G.2.0, MSIV Slow Closure. _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

CAUTION

- 2MSS*MOV111 and 112 are isolation valves in the steam line drains upstream of the Main Steam Isolation Valves (MSIVs). Because the closure times of these valves is longer than MSIV closure time, the duration that 2MSS*MOV111 and 112 are open shall be minimized. Minimizing the time that these valves are open will ensure we remain within the bounds of the current isolation time analysis. The total allowable time that valves may be open during a given year is less than 135 hours, when Primary Containment Isolation function is required (reference Calculation Dispositions No. ES-177-05E and ES-109-01B). **[C15]**
- When the Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig, the valves are subjected to a differential pressure greater than the engineering defined limit. Therefore, the valves must be declared INOPERABLE and Technical Specification 3.6.1.3, Condition B entered. Once entered, the associated action is to isolate the penetration with at least one valve within one hour (reference Calculation Disposition No. A10.2-O-026).

G.4.19. **BEFORE** ALL Control Rods In,
IF MSIVs are closed
AND Reactor Pressure is rising,
THEN CONTROL pressure using MSL drains by opening the following,
OTHERWISE MARK this step N/A:

a. 2MSS*MOV111..... ☐

b. 2MSS*MOV112..... ☐

c. **THROTTLE OPEN** 2MSS*MOV207 for 2 to 6 seconds..... ☐

d. **AFTER** 2 minutes,
THROTTLE OPEN 2MSS-MOV207 as needed for pressure control. . ☐ _____/_____

G.4.20. **IF** SRV opening is required to control rising Reactor Pressure,
THEN exit this procedure
AND enter N2-SOP-101C, Reactor Scram,
OTHERWISE MARK this step N/A. _____/_____.

G.4.21. **DURING** control rod insertion,
CONTROL Reactor Level in normal 178" to 187" control band
USING one or more of the following:

- 2FWS-LV55A(B) in automatic..... ☐
- 2FWS-LV55A(B) in manual..... ☐
- RDS-P1A(B) normal operation..... ☐
- WCS Reject to Main Condenser per N2-OP-37 ..... ☐
- Throttling 2FWS-V103A(B) ..... ☐
- Throttling 2FWS*MOV21A(B)..... ☐ _____/_____

G. SHUTDOWN PROCEDURE (Continued) **Initials/Date**

G.4.22. **VERIFY** ALL Control Rods inserted by one or more of the following:

- At 2CEC*PNL603 full core display, ALL Control Rod Full-In green lights ..... ☐
- At 2CEC*PNL603, Rod Worth Minimizer, confirm shutdown softkey ..... ☐
- Process Computer Control Rod Position display ..... ☐
- At 2CEC*PNL603, Four Rod Display indicates 00 ..... ☐ _____/_____

G.4.23. **RECEIVE** SM permission to enter Reactor MODE 3. _____/_____

NOTE

The following step will defeat Manual Scrams and the Shutdown Scram function of the Reactor Mode Switch temporarily to achieve Mode 3 without placing a hydraulic scram transient on the control rods. This will make control rod exercising during the shutdown unnecessary, assuming **no** other control rod maintenance is performed.

G.4.24. **PERFORM** N2-OP-97, Section H.3.0, Defeating the Mode Switch to Shutdown Scram. _____/_____

G.4.25. **VERIFY** Reactor Mode Switch is in SHUTDOWN. _____/_____

CAUTION

The remaining steps in Section G.4 are to be performed ONLY after the Reactor Mode Switch has been placed in SHUTDOWN.

G.4.26. **IF** MSIVs are closed
AND it is desired to open them,
THEN attempt to equalize pressure
AND OPEN MSIVs per N2-OP-1, Subsection F.2.0,
in order to regain pressure control with Turbine Bypass Valves,
OTHERWISE MARK this step N/A. _____/_____.

G.4.27. **GO TO** Section G.3, Post Reactor Shutdown/Scram Recovery, to complete plant shutdown, per the following:

a. **MARK** Steps G.3 through G.3.27.c, inclusive, as N/A. _____/_____

b. **COMPLETE** Section G.3 starting at Step G.3.27.d. _____/_____

H. OFF NORMAL PROCEDURE

H.1. Control Blade Guide Tube Flushing via Single Rod Scram

H.1.1. **USING** HU-AA-1211, Pre-Job Briefings,
PERFORM a Pre-Job Brief including Attachment 6, Shutdown Procedure Guidance. _____/_____

NOTE

- The purpose of this section is to remove crud from the bottom of a guide tube and load it into a CRDM scheduled for replacement during an outage. This minimizes the new CRDM from becoming immediately contaminated when placed into service thereby increasing service life.
- XX-YY is the number of the HCU being manipulated.

H.1.2. For control rod(s) to be flushed, as steps are completed,
LIST identification
AND INITIAL appropriate box on Attachment 5 using information provided by Reactor Engineering. _____/_____

H.1.3. **VERIFY** Control Rod is at position 00. _____/_____

H.1.4. **CLOSE** 2RDS-V113 (XX-YY), Charging Water Isolation Valve. _____/_____

H.1.5. **REMOVE** plug from 2RDS-V107 (XX-YY), Accumulator Drain. _____/_____

H.1.6. **USING** thread lubricant (Loctite 58031-Maximo stock number N9574180),
ATTACH drain line to HCU at 2RDS-V107 (XX-YY) downstream of fitting
AND route to floor/equipment drain. _____/_____

NOTE

The control rod becomes inoperable in the next step. Refer to T.S.3.1.5.

CAUTION

The next step may release water under high pressure.

H.1.7. **THROTTLE OPEN** 2RDS-V107 (XX-YY), Accumulator Drain,
UNTIL pressure remains steady as indicated on PI205, Local Pressure Indicator. _____/_____

H.1.8. **CLOSE** 2RDS-V107 (XX-YY), Accumulator Drain. _____/_____

H. OFF NORMAL PROCEDURE (Continued) **Initials/Date**

H.1.9. **REMOVE** drain hose from outlet of 2RDS-V107 (XX-YY) AND install plug. _____/_____

CAUTION

- This procedure directs high temperature reactor water to reactor building drain tanks. Dirt and excessive temperatures in the tanks could set off fire alarms.
- If an alarm associated with SDV level is received, then immediately close any open scram switches by placing SRI Test switches to the NORMAL position.

H.1.10. At HCU, a licensed operator shall **SIMULTANEOUSLY PLACE BOTH** SRI TEST switches to TEST position **FOR** approximately 3 minutes. _____/_____

H.1.11. **VERIFY BOTH** SRI TEST switches in NORMAL position. _____/_____

CAUTION

The Charging Water Isolation Valve, 2RDS-V113 must be cracked open slowly until the accumulator is fully charged before opening the valve completely.

H.1.12. **OPEN** 2RDS-V113 (XX-YY), Charging Water Isolation Valve. _____/_____

Attachment 1, RPV Cooldown Flow Chart

ENTRY CONDITION

RPV cooldown commenced and RDS Ref. Leg Backfill has been secured to one OR more RPV Ref. Legs

Station a dedicated Operator to monitor RPV Level Indications for notching and mismatch (see Note 1) using an R*TIME trend display with the following computer points: ISCLA106, ISCLA107, ISCLA108, ISCLA109.

While executing the following procedure:

IF	THEN
Notching occurs in some Channels, but NOT ALL,	• Use other valid Channels. • Reduce Reactor cooldown rate as low as pratical UNTIL notching subsides.
Notching occurs in ALL Channels,	• Use lowest end of notch as most valid. • Reduce Reactor cooldown rate as low as pratical UNTIL notching subsides.
Level mismatches occur between Channels,	• Use lowest reading Instrument as most valid. • Reduce Reactor cooldown rate as low as pratical UNTIL mismatch subsides.
Magnitude of notching or mismatches is severe, such that confidence in Level Indication is lost,	• Enter EOP RPV Control AND take actions for a loss of RPV Level Instruments.
Notches OR mismatch greater than satisfactory Channel Check occurs (see Note 3),	ESF Actuation may NOT occur WHEN required... • Use lowest indication OR other valid Channels AND alternate indications to identify breaks (sump levels, area temperatures, radiation monitoring instrumentation). • Isolate manually IF appropriate. • Declare respective Condensing Chamber Instrumentation inoperable.

While executing the following procedure:

System realignment AND maintenance other than that required for normal shutdown procedures (SDC, WCS Full Reject), which could raise the potential for system breaks OR Reactor drain down, shall be minimized AND should be authorized by the Operations Manager.

While executing the following procedure:

IF	THEN
Rapid cooldown is required (TS, emergency),	Continue at Ⓐ

IF more than one Channel is demonstrating notching, maintain RPV pressure. Contact I&C AND backfill Reference Legs (see Note 2).

BEFORE INITIATING SHUTDOWN COOLING

− Brief the operating crew on the evolution per Table 1.
− Walk down the SDC piping (see Note 4)

Ⓐ

Continue Reactor cooldown

IF more than one Channel is demonstrating notching, contact IMD AND backfill Reference Legs (see Note 2).

Wait UNTIL Reactor cooldown is complete AND the Reactor is in Mode 4.

Document observed notching on an IR.

Exit this procedure.

Attachment 1, RPV Cooldown Flow Chart (Continued)

NOTE

- Notching is the evolution of gases from the water in the cold reference legs of RPV level instrumentation which is observed as a step change rise in level. Normally followed in a short time by a step change down in RPV indicated level. It may occur at Reactor pressures below 450 psig, however, monitoring for notching shall be done at all reactor pressures. A Reactor Operator should be utilized to monitor for notching. In those cases when personnel availability is limited, another person (i.e. an Auxiliary Operator) may be utilized if briefed regarding his/her responsibilities. An R*TIME trend display may be utilized to monitor for notching using the following points for trending: ISCLA106, ISCLA107, ISCLA108, ISCLA109.

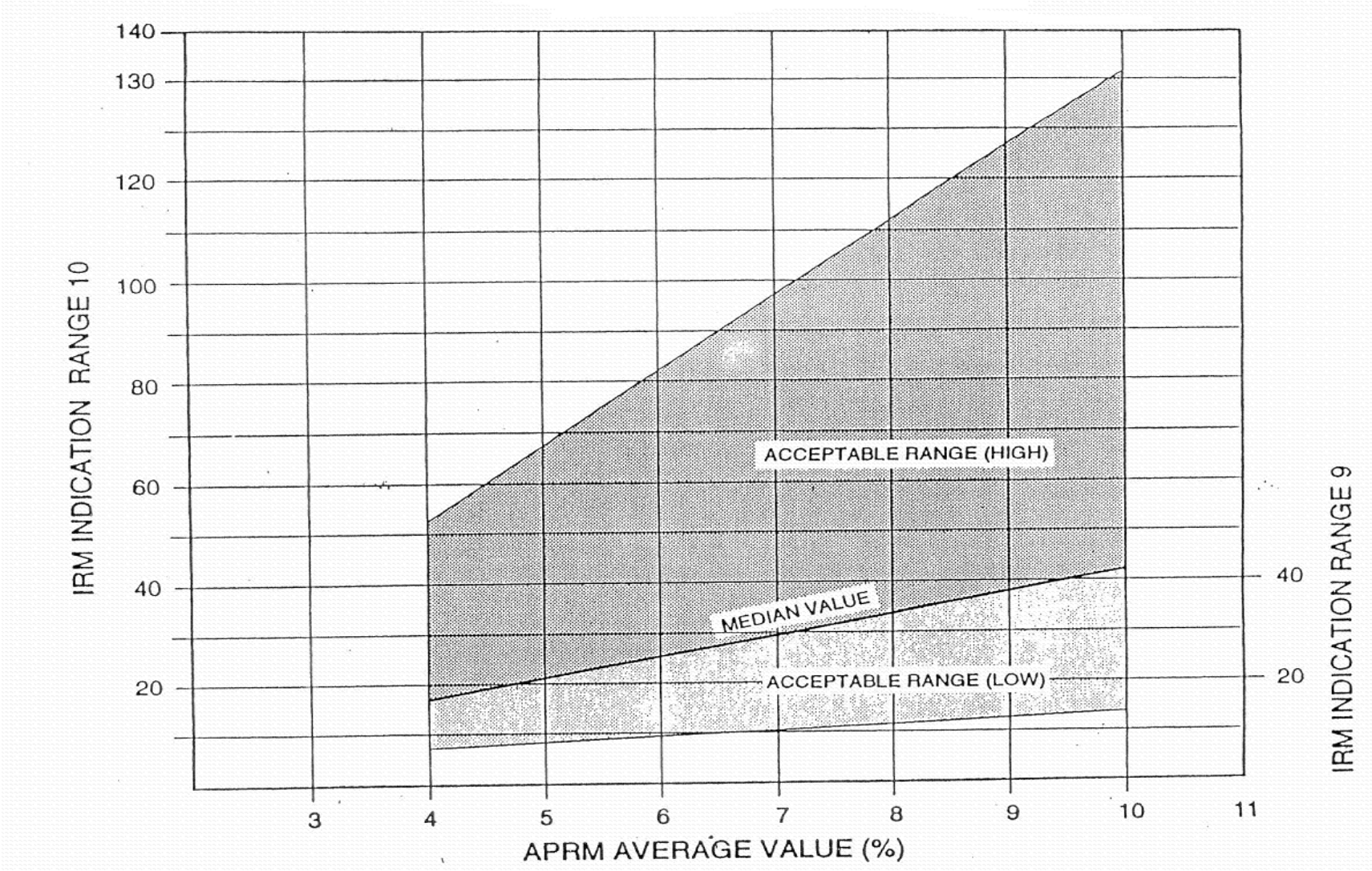
- Reference leg backfill allowable flow rates are small due to reference leg nozzle thermal stress concerns, and the evolution of backfilling an entire reference leg will take several hours. Proper documentation (IR) is warranted for Engineering follow-up evaluation. Reference leg backfill is accomplished using IMD approved procedures.

- Inoperability of an instrument due to notching shall be based on the CHANNEL CHECK differential value for that instrument. If the initial notch disappears after a short duration, the channel may be declared Operable. If more than one notch occurs on a channel, that channel shall be declared inoperable until reference leg backfill is performed and CHANNEL CHECK criteria is satisfied. Once the Reactor is in Mode 4 and the level instrumentation satisfies CHANNEL CHECK criteria, it may be declared operable before the reference leg backfill is performed.

- Walkdown of the SDC System for the purposes of this procedure is to verify piping integrity and that there are **no** obvious leakage paths. It is **not** intended to be a hand over hand inspection of the system nor is it intended that entrance into normally inaccessible areas be performed (i.e. High Radiation, high temperature, etc.).

- This Attachment and its requirements are required ONLY when the RDS Backfill Injection System has been secured to one or more RPV level instrument reference legs.

Attachment 2, IRM/APRM Overlap Verification



Attachment 3, Primary Containment AC Circuits

Breaker Number	Equipment Powered	Required Position	Actual Position	Initials
Reactor Building, El. 261', By South HCUs				
2LAR-PNLN05-14	Switch for Normal Lighting Contactor Coil Circuit	OFF		
2LAR-PNLU02-12	Switch for Essential Lighting Contactor Coil Circuit	OFF		
2LAR-PNLU02-14	Essential Lighting Circuits El. 240'	OFF		
2LAR-PNLU02-15	Essential Lighting Circuits El. 261'	OFF		
2LAR-PNLU02-16	Essential Lighting Circuits El. 289'	OFF		
Reactor Building, El. 261', By RDS FCV Station				
2LAR-PNLN12-1	Normal Receptacle Circuits El. 240'	OFF		
2LAR-PNLN12-2	Normal Receptacle Circuits El. 261'	OFF		
2LAR-PNLN12-3	Normal Receptacle Circuits El. 289'	OFF		
2LAR-PNLN12-11	Normal Lighting Circuits El. 261'	OFF		
2LAR-PNLN12-12	Normal Lighting Circuits El. 261'	OFF		
2LAR-PNLN12-13	Normal Lighting Circuits El. 261'	OFF		
2LAR-PNLN12-14	Normal Lighting Circuits El. 289'	OFF		
2LAR-PNLN12-15	Normal Lighting Circuits El. 289'	OFF		
2LAR-PNLN12-16	Normal Lighting Circuits El. 289'	OFF		
2LAR-PNLN12-17	Normal Lighting Circuits El. 240'	OFF		
2LAR-PNLN12-18	Normal Lighting Circuits El. 240'	OFF		
2LAR-PNLN12-19	Normal Lighting Circuits El. 240'	OFF		
Reactor Building, El. 289'				
2WPS-PNL200-20	2WPS-RCPT51A & 2WPS-RCPT51B Welding Receptacles Elev 261'	OFF		
2WPS-PNL200-26	2WPS-RCPT52A & 2WPS-RCPT52B Welding Receptacles Elev 261'	OFF		
Reactor Building, El. 306'				
2LAR-PNLU03-12	Communications Paging System for Drywell	OFF		

Attachment 4, Isolating Steam Line Drains for Pressure Maintenance

NOTE

- Low Power operations at high reactor pressure has the potential to result in resonant instabilities in EHC Pressure Control. This condition can occur with the Reactor sub-critical, decay heat less than 1% and a single bypass valve controlling reactor pressure less than 30% open. This condition will be manifested by oscillations in TBV position and small oscillations in reactor pressure. This condition has caused indicated oscillations in reactor water level, core plate differential pressure, and intermittent HPCS line break indication a frequency of approximately 4 to 5 Hz. Operation in this condition should **not** be allowed to continue and is mitigated by taking manual control of Rx pressure with the Bypass Jack which will result in a slightly lower reactor pressure to preclude the resonant oscillations (IR 04127454).

- Refrain from using the Turbine Bypass Valve Opening Jack DECREASE pushbutton until all Control Rods are inserted. IRMs will be very sensitive at this power range and may cause a half or full scram, if pressure is increased.

1.0 **IF not** using Bypass Valves to control pressure,
THEN at 2CEC*PNL851, **VERIFY** the following Bypass Valves closed,
OTHERWISE MARK this step N/A:

- 2MSS*PSV89A, MAIN TURBINE BYPASS VALVE ..... ☐
- 2MSS*PSV89B, MAIN TURBINE BYPASS VALVE ..... ☐
- 2MSS*PSV89C, MAIN TURBINE BYPASS VALVE ..... ☐
- 2MSS*PSV89D, MAIN TURBINE BYPASS VALVE ..... ☐
- 2MSS*PSV89E, MAIN TURBINE BYPASS VALVE ..... ☐ _____/_____

2.0 **PLACE** COMMON MASTER SWITCH GROUP I, II & III DRAIN VALVES in OVERRD. _____/_____

3.0 **IF not** using Bypass Valves to control pressure,
THEN VERIFY EHC Pressure Regulator setpoint as read on Digital EHC is higher than Reactor pressure (Computer Point FWSPA101),
OTHERWISE MARK this step N/A. _____/_____

Attachment 4, Isolating Steam Line Drains for Pressure Maintenance(Continued)

4.0 At SM discretion,
VERIFY the following valves CLOSED:

- MSS-AOV92A, STEAM SUPPLY VLV, at 2CEC*PNL851..... ☐
- MSS-AOV92B, STEAM SUPPLY VLV, at 2CEC*PNL851..... ☐
- ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, at 2CEC*PNL851..... ☐
- ASS-MOV152, CNDS AIR REMOVAL MAIN STM ISOL VLV, at 2CEC*PNL851..... ☐
- TME-AOV121, GLAND SEAL STM MAIN STM SPLY VLV, at 2CEC*PNL851..... ☐

NOTE

The following two valves may remain open if the SM determines access to the plant is **not** permitted or man power/time constraints prevent operation of these valves.

- 2ASS-STV143, BLDG INTMD EXCH MN ST ISOL, at 2CES-IPNL203. ☐
- 2ASS-SOV142, MAIN STEAM SUPPLY, at 2CES-IPNL122..... ☐ _____/_____

5.0 **VERIFY** the following valves CLOSED:

- At 2CEC*PNL602, MSS*MOV112 OR MSS*MOV111, MAIN STM LINE DRAIN ISOL VLV..... ☐
- MSS*MOV208, MSIV DRAIN VLV ..... ☐ _____/_____

6.0 At 2CEC-PNL824,
VERIFY the following CLOSED:

- MSS-AOV87A, MAIN STM LINE DRAIN VALVE..... ☐
- MSS-AOV87B, MAIN STEAM LINE DRAIN VLV..... ☐
- MSS-AOV87C, MAIN STM LINE DRAIN VLV..... ☐
- MSS-AOV87D, MAIN STM LINE DRAIN VLV..... ☐ _____/_____

7.0 **VERIFY** MSS-MOV187, MAIN STM LINE PRESS EQL/WARMING THROTTLE closed, by holding control switch in closed position at 2CEC*PNL602 _____/_____

8.0 At 2CEC*PNL603,
MONITOR Reactor Pressure on C33-R605. _____/_____

9.0 **VERIFY** N2-OP-1, Attachment 4, Section 1.0, has been completed. _____/_____

Attachment 4, Isolating Steam Line Drains for Pressure Maintenance(Continued)

10.0 As required to **CONTROL** reactor pressure,
COMMENCE CLOSING/OPENING the following valves at 2CEC*PNL851:

- 2TME-AOV121, GLAND SEAL STM MAIN STM SUPPLY VLV ..... ☐
- 2ASS-MOV148, MAIN STM TO AUX STM ISOL VLV ..... ☐

NOTE

If any of the following were overridden closed due to failing to operate, they will need to be opened manually:

- 2MSS-AOV194
- 2MSS-AOV203
- 2MSS-AOV205
- 2MSS-AOV209

At 2CEC-PNL824:

- MSS-AOV87A, MAIN STM LINE DRAIN VALVE..... ☐
- MSS-AOV87B, MAIN STEAM LINE DRAIN VLV..... ☐
- MSS-AOV87C, MAIN STM LINE DRAIN VLV ..... ☐
- MSS-AOV87D, MAIN STM LINE DRAIN VLV ..... ☐
- MSS-AOV203, MAIN STM LINE HEADER DRAIN VLV ..... ☐
- MSS-MOV21A, TURBINE STOP VLV MSV3 DRAIN VLV ..... ☐
- MSS-MOV21B, TURBINE STOP VLV MSV4 DRAIN VLV ..... ☐
- MSS-MOV21C, TURBINE STOP VLV MSV1 DRAIN VLV ..... ☐
- MSS-MOV21D, TURBINE STOP VLV MSV2 DRAIN VLV ..... ☐
- MSS-AOV88A, MAIN STM LINE HEADER DRAIN VLV..... ☐
- MSS-AOV88B, MAIN STM LINE HEADER DRAIN VLV..... ☐
- MSS-AOV209, MAIN STM LINE HEADER DRAIN VLV ..... ☐
- MSS-AOV194, MAIN STM LINE HEADER DRAIN VLV ..... ☐
- MSS-AOV205, MAIN STM LINE HEAD DRAIN VLV ..... ☐
- ASS-AOV144, AUX STM TO OFFGAS SYS LOW POINT DRAIN VLV ☐
- DTM-AOV6A, AUXILIARY STM DRAIN VLV ..... ☐
- DTM-AOV6B, AUXILIARY STM DRAIN VLV ..... ☐
- DTM-AOV108, AUXILIARY STM DRAIN VLV..... ☐
- DTM-AOV111, AUXILIARY SEAL STM HDR LOW POINT DRAIN ..... ☐
- DTM-AOV166, AUXILIARY SEAL STM HDR LOW POINT DRAIN ..... ☐ _____/_____

Attachment 4, Isolating Steam Line Drains for Pressure Maintenance(Continued)

11.0 **IF** the following valves were closed during performance of this Attachment,
THEN PLACE administrative controls on these as time permits.

- ASS-MOV148, MAIN STM TO AUX STM ISOL VLV ..... ☐
- TME-AOV121, GLAND SEAL STM MAIN STM SPLY VLV ..... ☐ _____/_____

12.0 **WHEN DIRECTED** to return main steam drains to normal,
THEN VERIFY the following valves OPEN:

At 2CEC*PNL851:

- 2TME-AOV121, GLAND SEAL STM MAIN STM SUPPLY VLV ..... ☐
- 2ASS-MOV148, MAIN STM TO AUX STM ISOL VLV ..... ☐

NOTE

If any of the following were overridden closed due to failing to operate, they will need to be opened manually:

- 2MSS-AOV194
- 2MSS-AOV203
- 2MSS-AOV205
- 2MSS-AOV209

At 2CEC-PNL824:

- MSS-AOV87A, MAIN STM LINE DRAIN VALVE..... ☐
- MSS-AOV87B, MAIN STEAM LINE DRAIN VLV..... ☐
- MSS-AOV87C, MAIN STM LINE DRAIN VLV ..... ☐
- MSS-AOV87D, MAIN STM LINE DRAIN VLV ..... ☐
- MSS-AOV203, MAIN STM LINE HEADER DRAIN VLV ..... ☐
- MSS-MOV21A, TURBINE STOP VLV MSV3 DRAIN VLV ..... ☐
- MSS-MOV21B, TURBINE STOP VLV MSV4 DRAIN VLV ..... ☐
- MSS-MOV21C, TURBINE STOP VLV MSV1 DRAIN VLV ..... ☐
- MSS-MOV21D, TURBINE STOP VLV MSV2 DRAIN VLV ..... ☐
- MSS-AOV88A, MAIN STM LINE HEADER DRAIN VLV..... ☐
- MSS-AOV88B, MAIN STM LINE HEADER DRAIN VLV..... ☐
- MSS-AOV209, MAIN STM LINE HEADER DRAIN VLV ..... ☐

Attachment 4, Isolating Steam Line Drains for Pressure Maintenance(Continued)

12.0 (Continued)

- MSS-AOV194, MAIN STM LINE HEADER DRAIN VLV ..... ☐
- MSS-AOV205, MAIN STM LINE HEAD DRAIN VLV ..... ☐
- ASS-AOV144, AUX STM TO OFFGAS SYS LOW POINT DRAIN VLV ☐
- DTM-AOV6A, AUXILIARY STM DRAIN VLV ..... ☐
- DTM-AOV6B, AUXILIARY STM DRAIN VLV ..... ☐
- DTM-AOV108, AUXILIARY STM DRAIN VLV ..... ☐
- DTM-AOV111, AUXILIARY SEAL STM HDR LOW POINT DRAIN ..... ☐
- DTM-AOV166, AUXILIARY SEAL STM HDR LOW POINT DRAIN ..... ☐ _____/_____

13.0 **PLACE** COMMON MASTER SWITCH GROUP I, II, III DRAIN VALVES to NORM. _____/_____

14.0 **VERIFY** N2-OP-1, Attachment 4, Section 2.0, has been completed. _____/_____

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
18-59										
22-59										
26-59										
30-59										
34-59										
38-59										
42-59										
14-55										
18-55										
22-55										
26-55										
30-55										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
34-55										
38-55										
42-55										
46-55										
10-51										
14-51										
18-51										
22-51										
26-51										
30-51										
34-51										
38-51										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
42-51										
46-51										
50-51										
06-47										
10-47										
14-47										
18-47										
22-47										
26-47										
30-47										
34-47										
38-47										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
42-47										
46-47										
50-47										
54-47										
02-43										
06-43										
10-43										
14-43										
18-43										
22-43										
26-43										
30-43										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
34-43										
38-43										
42-43										
46-43										
50-43										
54-43										
58-43										
02-39										
06-39										
10-39										
14-39										
18-39										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
22-39										
26-39										
30-39										
34-39										
38-39										
42-39										
46-39										
50-39										
54-39										
58-39										
02-35										
06-35										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
10-35										
14-35										
18-35										
22-35										
26-35										
30-35										
34-35										
38-35										
42-35										
46-35										
50-35										
54-35										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
58-35										
02-31										
06-31										
10-31										
14-31										
18-31										
22-31										
26-31										
30-31										
34-31										
38-31										
42-31										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
46-31										
50-31										
54-31										
58-31										
02-27										
06-27										
10-27										
14-27										
18-27										
22-27										
26-27										
30-27										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
34-27										
38-27										
42-27										
46-27										
50-27										
54-27										
58-27										
02-23										
06-23										
10-23										
14-23										
18-23										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
22-23										
26-23										
30-23										
34-23										
38-23										
42-23										
46-23										
50-23										
54-23										
58-23										
02-19										
06-19										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
10-19										
14-19										
18-19										
22-19										
26-19										
30-19										
34-19										
38-19										
42-19										
46-19										
50-19										
54-19										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
58-19										
06-15										
10-15										
14-15										
18-15										
22-15										
26-15										
30-15										
34-15										
38-15										
42-15										
46-15										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
50-15										
54-15										
10-11										
14-11										
18-11										
22-11										
26-11										
30-11										
34-11										
38-11										
42-11										
46-11										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
50-11										
14-07										
18-07										
22-07										
26-07										
30-07										
34-07										
38-07										
42-07										
46-07										
18-03										
22-03										

Attachment 5, Flushing of CRD Guide Tubes by Single Rod SCRAM Operator Signoffs at HCU (Continued)

Step Number	H.1.3	H.1.4	H.1.5	H.1.6	H.1.7	H.1.8	H.1.9	H.1.10	H.1.11	H.1.12
	VERIFY rod at "00"	**CLOSE** 2RDS-V113	**REMOVE** 2RDS-V107 plug	**INSTALL** Drain line	**THROTTLE OPEN** 2RDS-V107	**CLOSE** 2RDS-V107	**REMOVE** drain line	**PLACE BOTH** SRI switches to Test	**VERIFY BOTH** SRI switches in Normal	**OPEN** 2RDS-V113
Control Rod	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)	(✓)
26-03										
30-03										
34-03										
38-03										
42-03										

TABLE 1: Shutdown Cooling Initiation Briefing Checklist

PURPOSE

- Precautions deemed necessary AFTER the event at WNP-2 AND concern over notching preventing automatic ESFs ☐
- WNP-2 had a 32" notch for about 2 hours coincident with initiating SDC ☐
- Increased awareness to the possibilities of an RPV drain down event with notching ☐

EXPECTATIONS

- Always use good practices (Self Verification, Additional Verification, communications, etc.) ☐
- Questioning Attitude, Procedural Adherence, Attention to Detail ☐
- SAFETY is the highest priority ☐

RESPONSIBILITIES

- Dedicated Operator to monitor trends for NR level indications for notching ☐
- Operators performing SDC evolutions ☐
- Other evolutions in progress (**No** evolutions other than normal shutdown that have potential Reactor drain down effects) ☐
- Current plant status ☐
- Identify SM, Unit Supervisor (US), CRO, etc ☐

EXPECTED PLANT RESPONSE AND OPERATOR ACTIONS

- Normal to see a slight level decrease upon SDC initiation ☐
- A step rise in level could be indicative of notching (although other evolutions may look like a notch, such as securing a Recirculation Pump) ☐
- Define notching/mismatch ☐
- Criteria for instrumentation inoperability ☐
- Temporary modifications installed ☐
- Tagouts anticipated ☐
- Contingencies ☐
- See precaution D.11 (23" notch may inhibit NR ESFs) ☐

TABLE 2: Deleted

NOTE

Table 2 was deleted in Revision 04100.

Attachment 6, Shutdown Procedure Guidance

Time Critical nature of control rod insertions during Soft Shutdown

As reactor cooldown occurs, positive reactivity is added if the moderator temperature coefficient is negative (reference Cycle Management Report).

IF sufficient rods are **not** inserted OR rod insertion is interrupted,
THEN a power increase from the cooldown can occur. Minimize the duration of hold points below the point of adding heat. In theory, rod motion should be continuous from insertion of IRMs OR SRMs until ARI.

Review Definition of Critical

Increasing Count Rates, Sustained Positive Period with **no** rod motion

Review Definition of Subcritical

Specifically, the Reactor is considered subcritical when achieving a sustained negative period AND a continuously decreasing count rate with **no** Control Rod motion.

Roles and Responsibilities including Critical Parameter monitoring for Nuclear Instrumentation

Discuss the roles and responsibilities of the following personnel:

Rod Mover:

Peer Checker:

RM SRO:

BOP RO: Steam load impacts

QNE, if present:

Attachment 6, Shutdown Procedure Guidance (Continued)

Changes in steam load may cause changes in reactivity which could result in reactor criticality/sub-criticality OR undesirable reactivity change. Personnel performing Balance of Plant operations must communicate steam load changes to Operators performing reactivity changes.

Review Contingencies for Key/Critical Parameters:

IF Reactor goes RE-CRITICAL unintentionally (Rx from a SUBCRITICAL condition to a CRITICAL condition)

OR

Rod insertion cannot overcome the positive reactivity insertion due to moderator temperature changes

AND/OR Xenon changes due to low rod worth or other equipment issue,

THEN SCRAM the reactor.

Initial/Date

Attachment 7, Pressure Adjustments Using Automatic Cooldown

NOTE

- HOLD/GO/CANCEL buttons may be selected at any time during pressure adjustments to pause/resume/cancel the pressure change (i.e. to stabilize the plant, allow for control rod insertion, etc.).
- Automatic cooldown is performed by the pressure regulator controlling at least one BPV partially open, with automatic changes rate of the pressure change.
- The AUTO RATE button is selectable when the difference between PRESSURE SET and PAM PRESSURE is less than 5 psig or when PRESSURE SET is less than PAM PRESSURE.
- To exit automatic cooldown, select MANUAL RATE.
- Automatic cooldown will be automatically terminated when RPV pressure decreases to 50 psig.
- The AUTO RATE or MANUAL RATE button will be highlighted in blue indicating it is controlling and was the last method used for plant heat up or cooldown.
- During operation of the DEHC Automatic Cooldown feature, when the pressure regulator output is less than 1%, a Limit Cycle may occur. This is due to current setpoints used in the bypass valve opening bias. This is a known condition and can be corrected by establishing pressure control using the Bypass Valve Opening Jack (BOJM). After establishing control on the BOJM, cooldown can either completed manually, or reactor pressure can be lowered by about 10 psig and Automatic Cooldown can be reestablished, at the discretion of the US/SM.

1.1 **NAVIGATE** to the PRESSURE CONTROL screen (5532) on panel 2CEC*PNL851 HMI.0. _____/_____

1.2 **SELECT** MANUAL RATE (unless already highlighted blue). _____/_____

1.3 **SELECT** 1.5 PSIG/SEC. _____/_____

1.4 **ENTER** a PRESSURE SET value to match Reactor Pressure (PAM PRESSURE). _____/_____

1.5 **SELECT** ENTER. _____/_____

1.6 **SELECT** GO. _____/_____

Attachment 7, Pressure Adjustments Using Automatic Cooldown (Continued)

1.7 **WHEN** PAM PRESSURE **AND** PRESSURE SET **ARE WITHIN** 5 psig of each other **AND** AUTO RATE button is enabled (**not** gray), **THEN SELECT** AUTO RATE. _____/_____

1.8 **ENTER** desired COOLDOWN TEMPERATURE RATE. _____/_____

1.9 **SELECT** ENTER. _____/_____

1.10 **SELECT** CONFIRM **AND VERIFY** RATE is correct. _____/_____

1.11 **ENTER** a PRESSURE SETPOINT value to set the cooldown target pressure. _____/_____

1.12 **SELECT** ENTER **AND VERIFY** PRESSURE CONTROL TARGET is correct. _____/_____

1.13 **SELECT** GO. _____/_____

1.14 **PERFORM** the following to change COOLDOWN TEMPERATURE RATE:

a. **ENTER** desired COOLDOWN TEMPERATURE RATE. ..... ☐

b. **SELECT** ENTER. ..... ☐

c. **SELECT** CONFIRM, **AND VERIFY** RATE is correct..... ☐

d. **VERIFY** new rate is processing **AND** BPVs are controlling..... ☐ _____/_____

1.15 **IF** a Limit Cycle is observed in Reactor pressure **WHILE** in bi-stable bypass valve operation, **THEN PERFORM** the following, **OTHERWISE MARK** this step N/A.

a. **PLACE** Bypass Valve Opening Jack in service. ..... ☐

b. **IF** directed to control Reactor cooldown manually, **THEN CONTINUE** Reactor Cooldown Manually, **OTHERWISE LOWER** Reactor pressure 10 to 11 psig **AND PLACE** Automatic Cooldown back in service per steps 1.1 through 1.14..... ☐ _____/_____

Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-96

REACTOR MANUAL CONTROL AND ROD POSITION INDICATION SYSTEM

Revision 01100

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-21-02929:

- Description: Add steps in section H.2 to help identify and correct the loss of display for one 4-rod group on the 4-rod display.
- Reason: AR 04446364-05, Loss of Position Indication for 4 Control Rods
- H.2.1 preceding note, added that performance of H.5 Alternate Rod Position Indication Upon Loss of Indication [C2] [C5] is an acceptable method to satisfy the 24-hour surveillance requirement for Control Rod Position Indication required by TSSR 3.1.3.1 (reference IR 04446364-04).
- H.2.6 added new steps to replace a faulty rod display MUX card.

Procedure Writer Changes:

- Updated fonts to bold and lowercase underline for **not**, **cannot** and **no**, bold and uppercase for **ACTION VERBS** and **LOGIC WORDS** per AD-NM-101-1005 and AD-AA-101-1002, **no** revision bars used.
- Removed “open” or “close” from six steps that require throttling 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV, as personnel are trained for operation of this pressure control valve, and “throttle” is the correct action verb to describe it’s intended use. Often, specifying only “open” or “close” is insufficient direction as the throttle valve requires manipulation in both directions to accomplish the desired and specified differential pressure changes, **no** revision bars used. Changes made in the following steps:
  - F.2.1.e and F.2.1.f
  - F.3.1.f and F.3.1.g
  - F.3.2.g and F.3.2.h
- G SHUTDOWN reformatted previous revision paragraph into a note and step.
- For the upcoming transition to Exelon Smart Procedures (ESP), which does **not** display traditional MS Word tables well, and also in which the display of graphics are limited (graphics are also available by simply looking at the panel, are in training material, and are in the RMCS Vendor Manual, and may need to be deleted in the next revision for final conversion to ESP), moved the following previous revision attachments and figures:
  - Previous revision ATTACHMENT 1: ROD BLOCKS WITHDRAWAL BLOCKS, is now B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks, **no** revision bars used.
  - Previous revision FIGURE 1: REACTOR MANUAL CONTROL SYSTEM, is now B.7.1 Reactor Manual Control System, **no** revision bars used.
  - Previous revision FIGURE 2: ANALYZER PAGE INDICATORS, is now B.7.2 Analyzer Page Indicators, **no** revision bars used.
  - Previous revision FIGURE 3: RPIS DISPLAY CARD INDICATIONS, is now B.7.3 RPIS Display Card Indications, **no** revision bars used.
  - Previous revision Attachment 2, ROD POSITION VERIFICATION UPON LOSS OF POWER table information incorporated into written steps of section H.5 Alternate Rod Position Indication Upon Loss of Indication [C2] [C5], **no** revision bars used.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 4
B. SYSTEM DESCRIPTION 6
C. OPERATING REQUIREMENTS 23
D. PRECAUTIONS AND LIMITATIONS 24
E. STARTUP 25
F. NORMAL OPERATIONS 26
F.1. Control Rod Selection 26
F.2. Control Rod Withdrawal 27
F.3. Control Rod Insertion 32
G. SHUTDOWN 35
H. OFF-NORMAL PROCEDURE 36
H.1. Mispositioned Control Rod(s) [SOP] [C1] 36
H.2. Loss of Rod Position Indication or Control Rod Display 38
H.3. Rod Drift Alarm Test 41
H.4. Bypassing a Rod in Rod Drive Control 42
H.5. Alternate Rod Position Indication Upon Loss of Indication [C2] [C5] 45
H.6. Inoperable RDCS [C4] 62
H.7. Preventing Control Rod Withdraw Double Notching 63
H.8. Preventing Control Rod Insert Double Notching 66
H.9. Manually Inserting and Removing a Rod Block Monitor Rod Block 69

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

- 3.1.5, Control Rod Scram Accumulators
- 3.9.5, Control Rod OPERABILITY - Refueling
- 3.1.3, Control Rod OPERABILITY
- 3.9.4, Control Rod Position Indication
- 3.3.2.1, Control Rod Block Instrumentation
- 3.9.2, Refuel Position One-Rod Out Interlock
- 3.1.6, Rod Pattern Control

A.2. Licensee Documentation

- USAR Section 7.7.1.1, Reactor Manual Control System
- USAR Section 15.4.1.2, Continuous Rod Withdrawal During Reactor Startup

A.3. Technical Information

A.3.1. Drawings

- 12177-PID-30A, Control Rod Drive Hydraulic System
- 12177-PID-30B, Control Rod Drive Hydraulic System
- 12177-PID-30C, Control Rod Drive Hydraulic System

A.3.2. G. E. Drawings

- 791E406TY, RMCS Elementary Diagram
- 761E354, CRD Hydraulic System Functional Control Diagram
- 828E303AA, Rod and Detector Display Elementary Diagram
- 828E426AA, Rod Position Information System Schematic Diagram
- GEK 83320B, Figure 4-2, Analyzer Page Indicators

A.3.3. Instruction Manuals

- N20287 - GEK 83320B Reactor Manual Control System
- N20269 - GEK 83317 Control Rod Drive

A.3.4. General Electric (GE) Service Information Letters (SILs)

- G.E. SIL 174, Improved Rod Select Push Buttons
- G.E. SIL 407 Preventing Unauthorized Control Rod Patterns

A.3.5. Others

LER 86-10

A.4. **Commitments**

- **[C1]**, SOER 84-02, Recommendation 1, Prompt notification of Nuclear Engineering personnel and General Supervisor Operations prior to mispositioned control rod recovery.
- **[C2]**, 502882-06, Add an Alternate Method to Determine Control Rod Position.
- **[C3]**, SOER 84-02, Recommendation 6, Provide guidance for continuous withdrawal and insertion using continuous insert pushbutton or notch override switch.
- **[C4]**, IR 02062325, Reactor Manual Control System Lockup and RBM operability Tech Spec compliance.
- **[C5]**, IR 02095342, RPIS Failure - Loss of 42 Control Rod Position Indications

B. SYSTEM DESCRIPTION

B.1. Reactor Manual Control System (RMCS)

Composed of four subsystems which allow the operator to select and move control rods as needed for efficient fuel management, and to control core power and flux distribution.

The Rod Drive Control System (RDCS) accomplishes all control rod movements except for a scram. The RDCS monitors a variety of conditions pertaining to the control rods and gives individual indications and summaries of those conditions. The RDCS receives inputs from various plant systems and will prevent rod withdrawals by enforcing rod blocks if selected plant parameters exceed their trip setpoints. See B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks for a list of rod blocks.

The Rod Position Indication System (RPIS) provides information on the vertical position and temperature of the control rod drives.

The Rod Worth Minimizer (RWM) functions to limit control rod worths to minimize the effect of a control rod accident or rod withdrawal errors by enforcing adherence to specified control rod patterns through the use of control rod blocks. The operation of the RWM is covered by N2-OP-95A.

B.2. Rod Drive Control System (RDCS)

The RDCS consists of the controls indicators and circuitry necessary for normal insertion and withdrawal of control rods, one at a time. This system includes interlocks to prevent normal rod motion when rod block signals are present. The RDCS will **not** cause or prevent a scram, which is controlled by the Reactor Protection System N2-OP-97.

The RDCS operates in three modes: 1) The operator control mode, 2) The scan mode, and 3) The Self-Test mode. The Operator Control mode is used to move control rods. The Scan mode is used to gather status data for a rod. The Self-Test mode is used to test the functions of the transponder cards and the related command circuitry. The system spends approximately 100 microseconds in a mode, and alternates between the Operator Control mode and the Scan mode until the status data is gathered for all rods. Each time the scan mode is entered, a different rod is scanned according to a prescribed sequence. When all rods have been scanned, the Self-Test mode will replace the Scan mode for one alternation to test one transponder card. Each time the system enters the Self-Test mode, the next rod in the sequence is scanned so that the circuitry for all rods is tested.

The RDCS is comprised of the following major components:

- Rod Select Module (RSM)
- Rod Drive Control Cabinet (RDCC)
- Branch Junction Modules (BJMs)
- HCU Transponders
- Display Memory Module (DMM)
- Rod and Detector Display (R&DD)

B. SYSTEM DESCRIPTION (Continued)

B.2.1. Rod Select Module

The Rod Select Module is located on the sloping portion of the Operator's console on 2CEC*PNL603. It contains a push button array for selecting:

- The control rod to be moved
- Stabilizer valve set (A or B)
- Drift control (Test, Reset)
- Rod movement commands
- Accumulator Trouble Acknowledge

It also contains two identical transmitter cards, that independently sense which buttons have been pressed by the operator. The transmitter cards then generate duplicate binary REQUEST words which are transmitted to the activity control cards in the Rod Drive Control Cabinet. The RSM also receives inputs from the Rod Sequence Control System and the Rod Worth Minimizer.

B.2.2. Rod Drive Control Cabinet

The Rod Drive Control Cabinet is the heart of the system and is located in panel P616. It contains four major sections. At the top of the cabinet are two identical Activity Control sections. Below this is an Analyzer section, and the Power Supply Section is in the bottom of the cabinet.

The two Activity Control sections are identical and independent. Each section consists of an Input Isolator card, an Activity Control card, and a Rod Motion Timer Card. Each Activity Control section receives a REQUEST word from the Rod Select Module. The REQUEST word, inputs from the rest of the system, plus the rod block signals from other systems enter the Activity Control section through the Input Isolator card, which electrically isolates the signals coming from outside the cabinet. The main function of the Activity Control Card is to monitor the status of the plant to determine if rod motion requested by the operator is permissible. If the request for rod movement is permitted (i.e.; **no** Rod Blocks), the Activity Control Cards each generate a COMMAND word and initiates operation of the respective Rod Motion Timer card. The Rod Motion Timer cards produce a 10 second timing signal to move a control rod one notch.

B. SYSTEM DESCRIPTION (Continued)

The Analyzer section contains an Analyzer card, a Fault Map card, and isolator cards. The Analyzer section is designed to detect errors in control signals that may occur due to electrical noise, equipment faults, etc.; and to prevent inadvertent rod motion due to these errors. The Analyzer section compares the COMMAND words from the Activity Control Sections for equivalency. If they are the same, one of the COMMAND words is transmitted to the transponders through the Branch Junction Modules, and the other is retained in the Analyzer as the REFERENCE word. The REFERENCE word is compared to the ACKNOWLEDGE word (returned to the cabinet from the Transponder of the selected rod) for errors. If a sufficient number of errors occur, the location of the fault (the BJM or the Transponder) will be indicated on the Fault Map card and system operation will be interrupted. The Analyzer section includes a number of controls and display registers which are used to verify proper system operation and to assist IMD personnel in troubleshooting the system. The Display registers hold all of the data available at the time the decision was made to stop system operation.

The Fault Map card contains an array of LEDs laid out in a mimic of the core and is used to indicate faults with the Transponders. The Fault Map card also contains counters that are used to control the Scan mode and the Self-Test mode.

The power section of the RDCC contains the main circuit breakers for the system, the Power Gate and the Contact Power Supply. The Power Gate consists of two large silicon-controlled rectifiers (SCRs) and controls the 120 VAC power to the transponders. If the Analyzer section encounters a problem in the system, it interrupts the control signal to the SCRs, turning them off.

This will disconnect all power to the HCUs and will prevent a failure in the system from causing inadvertent rod motion. The Contact Power supply furnishes power to all external contacts for the system.

B.2.3. **Branch Junction Module**

The Branch Junction Module is a enclosure mounted at the end of a cluster of HCUs. A BJM serves as a termination point for cables from the control room, adjoining BJMs, and for the Transponders on its group of HCUs. The BJM receives the COMMAND word, amplifies and shapes the signal, and transmits it to its associated cluster of transponders, and to the downstream BJM (if applicable). It also receives the ACKNOWLEDGE words from transponders within its cluster and downstream BJMs, and transmits it upstream to the next BJM or the RDCC.

B.2.4. **HCU Transponders**

The Transponders compare the rod address within the COMMAND word (transmitted from the RDCC) to their own. The Transponder that matches (i.e.; that rod has been selected for movement), will process the COMMAND sequence generated by the RDCC. The selected Transponder immediately returns an ACKNOWLEDGE word back to the Analyzer Card in the RDCC. The ACKNOWLEDGE word includes information on the present status of: the solenoid valves, scram valve position, accumulator pressure and liquid level, and the scram test switches. The selected Transponder continues to process the COMMAND work to open and close the Directional control Valves in the specified sequence to move the control rod. The Transponders also provide amplification to signals being passed through it to and from other Transponders.

B. SYSTEM DESCRIPTION (Continued)

B.2.5. Display Memory Module (DMM)

The Display Memory Module receives information from the RPIS and RDCS, stores the information from the RPIS and the RDCS, stores the information in its memory, and prepares the information for display on the Rod and Detector Display (R&DD) and the Four Rod display.

B.2.6. Rod and Detector Display (R&DD)

The Rod and Detector Display, or "full core display", is mounted in the vertical section of 2CEC*PNL603. It is an array of back-lit indicator elements arranged in a mimic of the core. Six parameters are displayed for every control rod:

- FULL IN: Lit when the rod is fully inserted.
- FULL OUT: Lit when the rod is fully withdrawn.
- DRIFT: Lit if the rod is moving and should **not** be.
- SELECTED: Specific rod's push button is lit when the rod is selected on the RSM.
- ACCUM: Lit if the accumulator pressure or water level is abnormal in the HCU.
- SCRAM: Lit if both scram valves on the HCU have opened.

The display also provides indication of Upscale and Downscale alarms for the LPRMs in each location (from the Neutron Monitoring System).

In addition to information given by the full core display, the 4-rod display presents the vertical position of a selected rod and the other rods in it's group, and LPRM readings for the LPRMs located next to the group. The rod positions displayed will be in the form of a two-digit number with an illuminated background. Even notch positions are indicated by numbers and Odd notch positions are indicated by dashes. If RPIS is receiving abnormal data (e.g.; a reed switch stuck open), it will be indicated by a double X in the display. If the selected group has less than four rods, the display window(s) will be blank for the locations that do **not** have rods.

B.2.7. Operator Control Mode

In the operator control mode, the operator initiates movement of a control rod by momentarily depressing the rod select push button on the RSM to select the rod to be moved. Then the operator pushes the INSERT or WITHDRAW push button as desired. If there are **no** rod blocks present, the RDCC initiates the sequence to move the rod in the desired direction.

To insert the selected rod, the RDCC sends a signal to the selected rod's HCU to open the insert supply and exhaust valves to drive the rod inward. After about 3 seconds, during which the rod moves in slightly more than one notch (6 inches), the insert valves close, and the settle valve opens for about 5 seconds to allow the rod to settle back down to the next notch position. Depressing the INSERT push button continuously causes a continuous insertion of the selected control rod. A control rod can also be continuously inserted by depressing and holding the continuous insert push button. This bypasses the rod motion timer cycle and the rod settle feature does **not** operate in this mode.

B. SYSTEM DESCRIPTION (Continued)

To withdraw the selected rod, the RDCC sends a signal to the selected rod's HCU to open the insert supply and exhaust valves for 0.6 seconds to insert the rod far enough to free the collet fingers from the detent and hold them open. After the insert valves shut, the withdraw valves open to drive the rod out. The withdraw supply valve closes after 1.5 seconds and the withdraw exhaust (settle) valve remains open for an additional 6 seconds to allow the rod to "settle" down to the next notch. Continuous withdrawal of a control rod is achieved by simultaneously depressing the CONTINUOUS WITHDRAW and the WITHDRAW push buttons. The dual push button requirement prevents an inadvertent continuous withdrawal of control rods.

B.2.8. **Scan Mode**

In the scan mode, the Analyzer card initiates commands to gather status data from a single HCU. Accumulator pressure and water level, scram valve position, and scram test switch position are checked. The Fault Map card generates a rod I.D. and sends it via the Analyzer card to each of the Activity Control cards.

Each Activity Control card forms a COMMAND word (although in this case, **no** rod motion is called for) which is transmitted in the same manner as in the Operator Control mode. The selected Transponder transmits the HCU status data back to the RDCC. The system does **not** verify correctness of the reply because the ability of the Transponders to reply correctly is checked in the Self-Test mode and, besides there is **no** reference available for the status data. At the next alternation with the Operator Control mode, a counter advances and causes the Fault Map card to generate the next rod I.D., which is processed as above.

B.2.9. **Self-Test Mode**

This mode is used for diagnosis and evaluation. As in the Scan Mode, the Fault Map card generates the rod I.D., but this time it also generates a specific valve control signal determined by a built-in program in the Analyzer. It sends this information to each Activity Control, where a COMMAND word is formed and processed exactly as in the other modes. The Analyzer program briefly exercises the valve driver circuit on the HCU; **not** long enough for mechanical valve motion, but long enough to verify electrical operation by means of a drive monitoring circuit located on the Transponder. The ACKNOWLEDGE word returned to the Analyzer is checked for correct response; both that the correct Transponder replies as indicated by the I.D. and that the valve driver circuit response is correct. In the Self-Test evaluation, a comparison failure of more than a prescribed number of words results in an automatic shutdown of the RDCS and 603303, ROD DRIVE CONTROL SYSTEM INOPERABLE annunciator.

B. SYSTEM DESCRIPTION (Continued)

B.3. Interlocks

The Reactor Manual Control System (RMCS) inhibits movement or selection of control rods under various conditions. The rod blocks caused by inputs from other systems are listed in section B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks.

The existing NUMAC APRMs have an adjustable rate of 0.52 to 0.78 for the slope of the flow biased curves. The single loop flow biased rod block slope **cannot** be set to the desired slope of .50 until a firmware update is completed for MELLLA+ at a later date. The single loop operation EPU APRM rod block setpoint formula is modified by ECP-10-000818 to use the slope of 0.52 and maintains a conservative rod block setpoint.

Rod select blocks are initiated by the RDCC to prevent the selection of a control rod under the following conditions:

- A control rod has already been selected for movement and the movement cycle is **not** complete.
- The rod selected key-lock switch on the Rod Select Module is **not** rotated fully clockwise.
- One rod is selected and withdrawn while in the Refuel mode.

RMCS system failures (i.e.; disagreement between the Activity Control sections, or an error detected by the Analyzer section) and performance of manual testing will impose rod blocks.

B.4. Rod Withdraw Blocks

B.4.1. SRM Downscale Alarm Rod Withdraw Block

a. Setpoint: 3 cps

b. Bypassed:

- IRMs Range 3 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed

B.4.2. SRM Upscale Rod Withdraw Block

a. Setpoint: 1×10^5 cps

b. Bypassed:

- Associated IRM on Range 8 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.3. SRM INOP Rod Withdraw Block

a. Setpoints:

- Module unplugged
- High Voltage Low
- SRM Select Switch **not** in Operate*

b. Bypassed:

- Associated IRM on Range 8 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed
- * INOP Inhibit toggle switch actuated

B.4.4. SRM Detector **not** Full In Rod Withdraw Block

a. Setpoint: < 100 cps **AND** Detectors **not** fully inserted

b. Bypassed:

- IRMs Range 3 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM Bypassed

B.4.5. IRM Downscale Alarm Rod Withdraw Block

a. Setpoint: 5/125 of scale

b. Bypassed:

- IRM on Range 1
- Mode switch in RUN
- IRM bypassed

B.4.6. IRM Upscale Alarm Rod Withdraw Block

a. Setpoint: 108/125 of scale

b. Bypassed:

- Mode switch in RUN
- IRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.7. IRM Detector **not** Full In Rod Withdraw Block

a. Setpoint: IRM Detector **not** fully inserted

b. Bypassed:

- Mode switch in RUN
- IRM bypassed

B.4.8. IRM INOP Rod Withdraw Block

a. Setpoints:

- Module unplugged
- High Voltage Low
- IRM Drawer Selector Switch **not** in Operate*

b. Bypassed:

- Mode switch in RUN
- IRM bypassed
- *INOP inhibit toggle switch actuated

B.4.9. APRM Downscale Alarm Rod Withdraw Block

a. Setpoint: $\geq$ 4% Average Neutron Flux

b. Bypassed”

- Reactor Mode Switch **not** in RUN
- APRM bypassed

B.4.10. APRM Upscale Alarm Rod Withdraw Block

a. Setpoints:

- $\leq$ 12% Simulated Thermal Power
- $\leq$ 0.55 Wd + 48.5% (two loop operation)
- $\leq$ 0.52 (W-ΔW) + 41.1% (single loop operation) (clamp at 107.5%)

b. Bypassed:

- Reactor Mode Switch **not** in RUN
- APRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.11. APRM INOP Rod Withdraw Block

a. Setpoints:

- Any Self-Test fault detected
- Total LPRM inputs from any one axial level < 3

b. Bypassed:

- APRM bypassed

B.4.12. APRM Flow Upscale Alarm Rod Withdraw Block

a. Setpoint: ≤ 108% Flow

b. Bypassed:

- Associated APRM bypassed

B.4.13. RBM Downscale Alarm Rod Withdraw Block

a. Setpoint: Average RBM Flux for Rod selected LPRMs is less than 75.8% power

b. Bypassed:

- Reactor Mode Switch **not** in RUN
- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B.4.14. RBM Upscale Alarm Rod Withdraw Block

a. Setpoints:

1) Low: (≥ 26% & < 61% RTP) *124.2%

2) Intermediate: (≥ 61% & < 81% RTP) *< 119.2%

3) High: (≥ 81% RTP) *< 114.2%

* Local LPRM Power is Normalized to 100% upon rod selection

b. Bypassed:

- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B. SYSTEM DESCRIPTION (Continued)

B.4.15. RBM INOP Alarm Rod Withdraw Block

a. Setpoints:

- Critical Self-Test Fault detected
- RBM keylock switch in INOP
- < 1/2 required LPRM inputs
- > 1 rod selected
- Firmware/Software Watchdog Timer timed out
- Loss of input power

b. Bypassed:

- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B.5. **Rod Worth Minimizer Withdraw and Insert Rod Blocks**

The following is a brief summary of the rod blocks imposed by the Rod Worth Minimizer. For more information, refer to N2-OP-95A.

The Rod Worth Minimizer (RWM) applies rod block and permissive signals to the RDCS which generate the rod block or permissive output signals. Rod blocks are applied when the RWM permissive requirements are **not** met and power is below the LPSP. A control rod in a group is in the permissive state when there are two or less insert errors and **no** withdraw errors.

Another RWM system interlock involves the Rod Test sequence. The Rod Test mode may only be entered if **no** more than one control rod is withdrawn from the fully inserted position. If a rod is being tested and is in any position other than notch 00, a rod withdrawal block will be initiated if a second rod is selected on the RSM. If more than one rod is withdrawn beyond the 00 position, and the Rod Test push button is depressed, both insert and withdraw blocks are immediately applied. The Rod Test push button must be depressed a second time to return the system to normal.

Once power is raised above the LPSP, **no** rod blocks are applied, but alarm indications and insert error displays are still provided. At power levels above the Low Power Alarm Point (LPAP), the monitoring functions of the RWM are automatically bypassed. System hardware failures will initiate rod blocks at any power level. Whenever a RWM program abort occurs, the RWM is automatically taken off-line, withdraw and insert blocks are applied and the RWM must be reinitialized before the program can be executed again.

B. SYSTEM DESCRIPTION (Continued)

B.5.1. RWM Withdrawal Rod Block

a. Setpoints:

- Any Withdraw Error
- 3 Existing Insert errors **AND** an Insert Block applied causes a block for any rod selected other than one causing an Insert error
- Rod Test:
  - One rod withdrawn
    AND another rod selected
  - One rod withdrawn
    AND Rod Test Push button is depressed
- RWM program abort.

b. Bypassed:

- Reactor Power above the LPSP (as sensed by steam flow)
- RWM Bypassed

B.5.2. RWM Insert Rod Block

a. Setpoints:

- Two Insert Errors existing
  AND third Insert Error is made
- A Withdraw error existing
  AND a Withdraw Block applied causes a block for any rod selected other than one causing the Withdraw error.
- When power reaches the LPSP while lowering Rx Power
  AND:
  - A Withdraw error exists
  - **OR** the Operating Sequence is **not** latched
- One rod withdrawn
  AND the Rod Test push button is depressed
- RWM program abort.

b. Bypassed:

- Reactor Power above the a LPSP (as sensed by steam flow)
- RWM Bypassed

B. SYSTEM DESCRIPTION (Continued)

B.6. Rod Position Indication System (RPIS)

The RPIS continuously monitors the vertical position and temperature of the control rod drives. The position information is transmitted to the Rod Drive Control System (RDCC) for display on the Rod and Detector Display (R&DD) and the 4-Rod Display, and to the process computer.

A thermocouple is attached to each control rod drive. The thermocouple sends a temperature signal to a chart recorder on panel 2CES*RAK007 (Rx Bldg. elev. 261', North) and energizes an annunciator on 2CEC*PNL603 if a high temperature condition is sensed.

The Position Indication Probes are the primary source of information for the system. Each control rod drive contains a Position Indicator Probe, which contains 53 reed switches connected in an eleven wire array. A closed reed switch will connect two of the eleven wires together and provides a unique output corresponding to the position of the rod. The drive piston in the CRD contains a magnet that closes the reed switch located nearest to the magnet. Forty-nine (49) switches are used to detect the position of the control rod. They are spaced at 3 inch intervals and are numbered from notch position 00 to 48. Notch 00 indicates that the rod is fully inserted into the core and notch 48 indicates that the rod is fully withdrawn. The even numbered switches correspond to the latch positions of the drive. The odd numbered switches correspond to intermediate positions between the notches on the drive. A second switch is located at the same location as the 00 position and is used to light the green FULL-IN indication on the full core display, R&DD. Another switch is located 1 1/4 inches further in than the notch 00 switch and keeps the green FULL-IN light energized when the rod is driven in past position 00 (e.g.; during a scram). Another switch is located at the same location as the notch 48 switch and is used to light the red FULL-OUT indication on the full core display. The last switch is located two inches further out than notch 48 and is used to detect an uncoupled rod. During normal rod motion, the rod is prevented from traveling past notch 48 by the control rod backseating on the stub tube. However, if the control rod drive is uncoupled from the blade, the drive is free to move out past notch 48 and will close the "overtravel" reed switch. This switch is used to initiate an annunciator on 2CEC*PNL603.

All of the wires from the probes terminate in the Rod Position Indication Cabinet (RPIC) located in 2CEC*PNL615 in the control room. The RPIC is the heart of the RPIS. It contains three modules, a Control module and two Multiplex modules. The Control module contains all of the electronics for controlling the system.

There are **no** operator controls or displays directly associated with the RPIS. RPIS supplies the information to the Display Memory Module, which drives the Rod and Detector Display and the 4-Rod Display.

Typical signal flow through the system is as follows: A counter in the RPI Cabinet advances automatically through all of the rod addresses. A multiplexed word, the IDENT word, is distributed to all of the Probe Multiplexer (MUX) cards in the system. These cards (each of which services a 4 rod group) compare the address contained in the IDENT word with their identity as defined by their Row and Column IDENT cards.

The card whose identity matches the address will respond with the position data from the corresponding control rod probe. This data is analyzed by the Probe Data Processor III Card in the RPIC.

B. SYSTEM DESCRIPTION (Continued)

The processor first determines if the data is acceptable. That is, the data must represent a realistic state for the control rod (e.g.; the rod **cannot** be in both positions 12 and 14 at the same time). If the data is **not** realistic, then some defect exists in the system.

If the data is acceptable, the processor will translate it from a pure binary format into a binary coded decimal format that is used to drive the position displays. This code and other data is appended to the IDENT code and transmitted to the Display Memory Module as the POSITION word. The position data is also loaded into a local memory which can be accessed by the plant computer. If the data is **not** acceptable, it is discarded and a fault code is generated. The fault code is transmitted instead, as part of the POSITION word.

After transmitting a POSITION word, the counter will advance to the next address and the process will be repeated. This cycle is repeated continuously, checking the status of all the rods. It takes about 40 milliseconds to cycle through all of the rods. The system does **not** stop if a fault is detected.

Operator inputs into the system are multiplexed into REQUEST words by the Rod Select Module in the RDCS. A REQUEST word is transmitted to the RPI Cabinet. From this word, the RPIC extracts:

- The IDENT of the selected rod.
- The Drift Test command (if applicable).
- The Drift Reset request (if applicable).

One of the bits in the POSITION word is used to identify the selected rod. When, in the process of scanning the probes, the processor comes across the rod whose IDENT is the same as the REQUEST word, it will set this bit into the POSITION word prior to transmitting it.

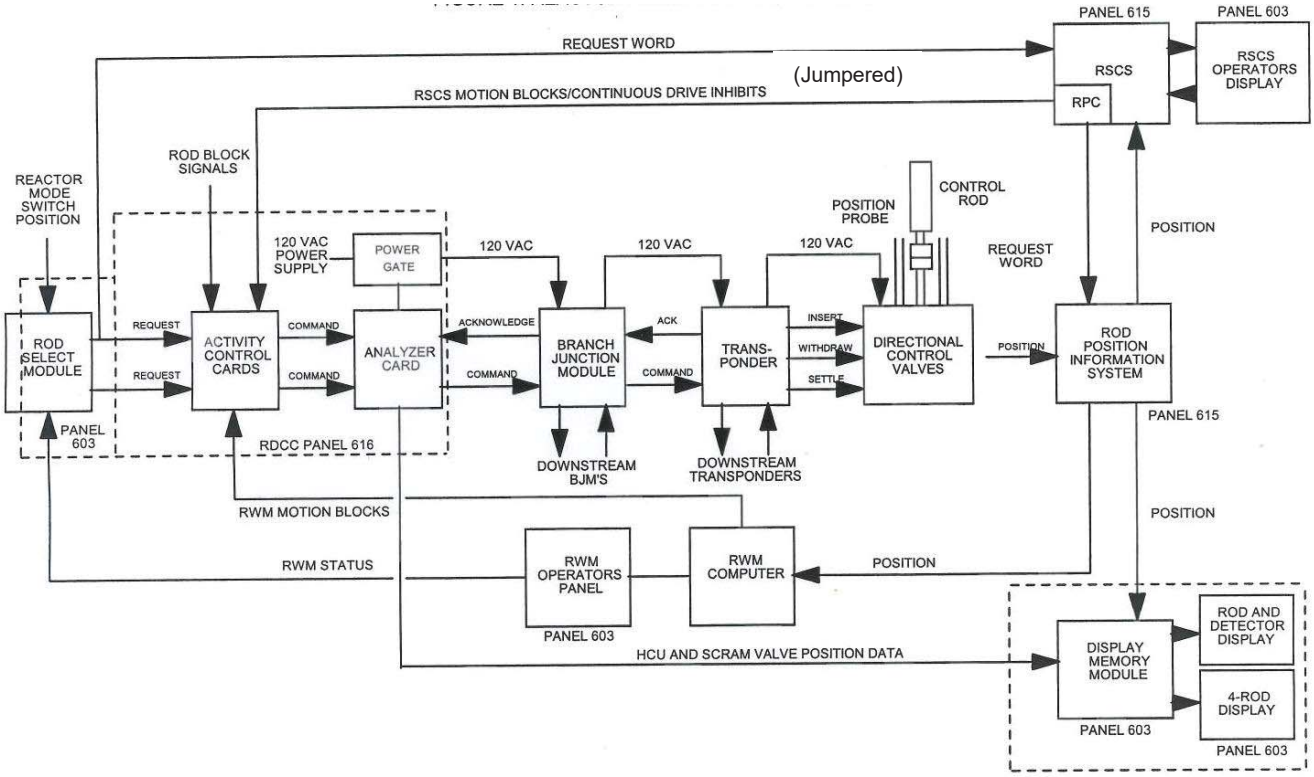
Unless a control rod is being moved, it should only be located at an even-numbered notch position. If a rod is located at an odd-numbered position, it is defined as Drifting (unless the operator is moving that rod). One bit in the POSITION word carries the drift information for that rod, and once set, remains even if the rod stops drifting (i.e.; latches into an even-numbered notch). A "Drift Reset" signal originated by the operator and carried by the REQUEST word, will cause all drift signals to reset unless the rod is still drifting. The "Drift Test" is used to check the drift detection function by temporarily overriding the "rod driving" signal (coming from RDCS) while the operator moves a rod through an odd-numbered position.

The 4-rod display presents the vertical position of a selected rod and the other rods in it's group. The rod positions are displayed in the form of a two digit number. Even notch positions are indicated by numbers and Odd notch positions are indicated by dashes. If RPIS is receiving abnormal data (e.g.; a reed switch stuck open), it will be indicated by a double X in the display. If the selected group has less than four rods, the display window(s) will be blank for the locations that do **not** have rods.

B. SYSTEM DESCRIPTION (Continued)

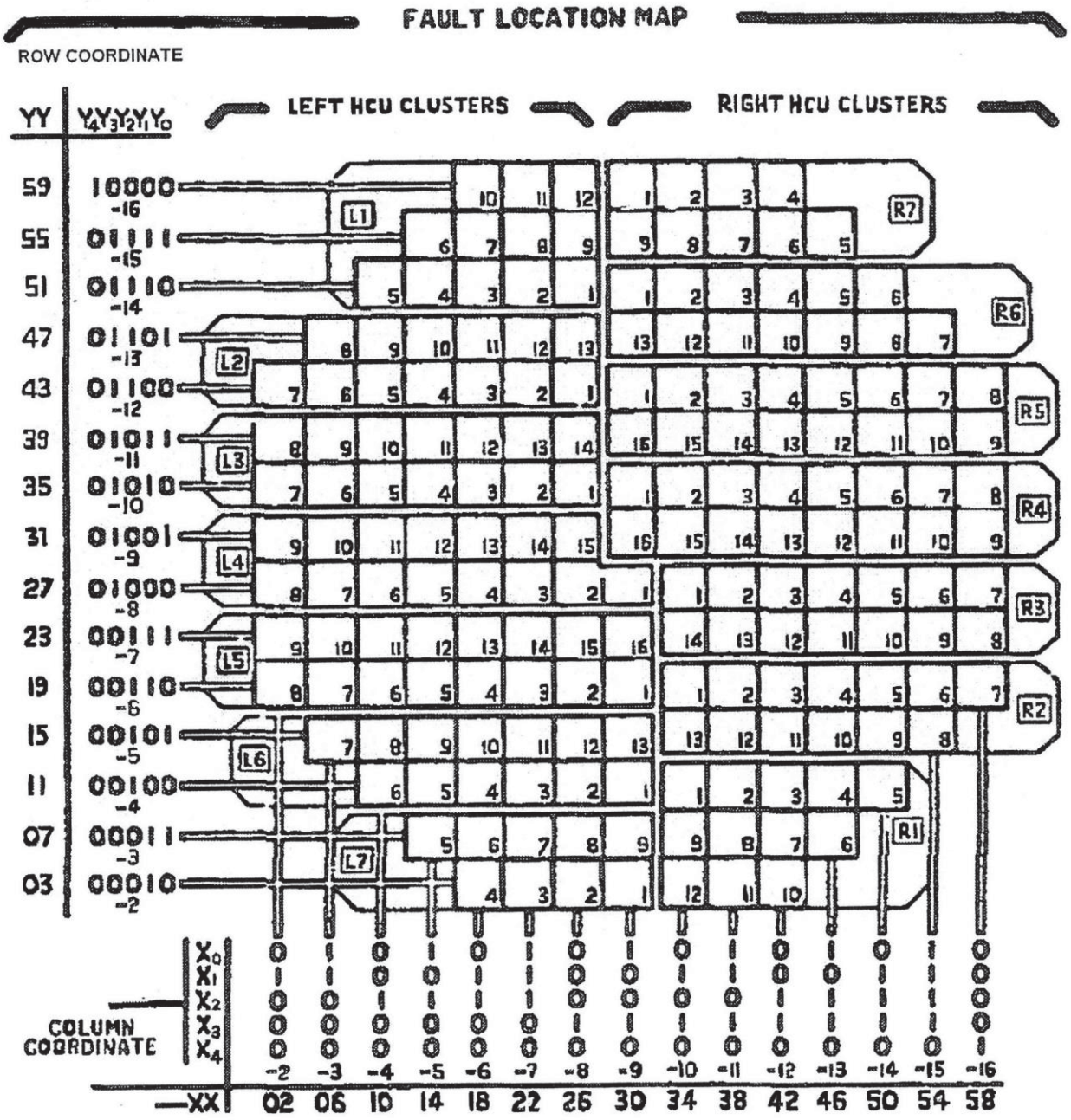
B.7. System Figures and Panel Depictions

B.7.1. Reactor Manual Control System



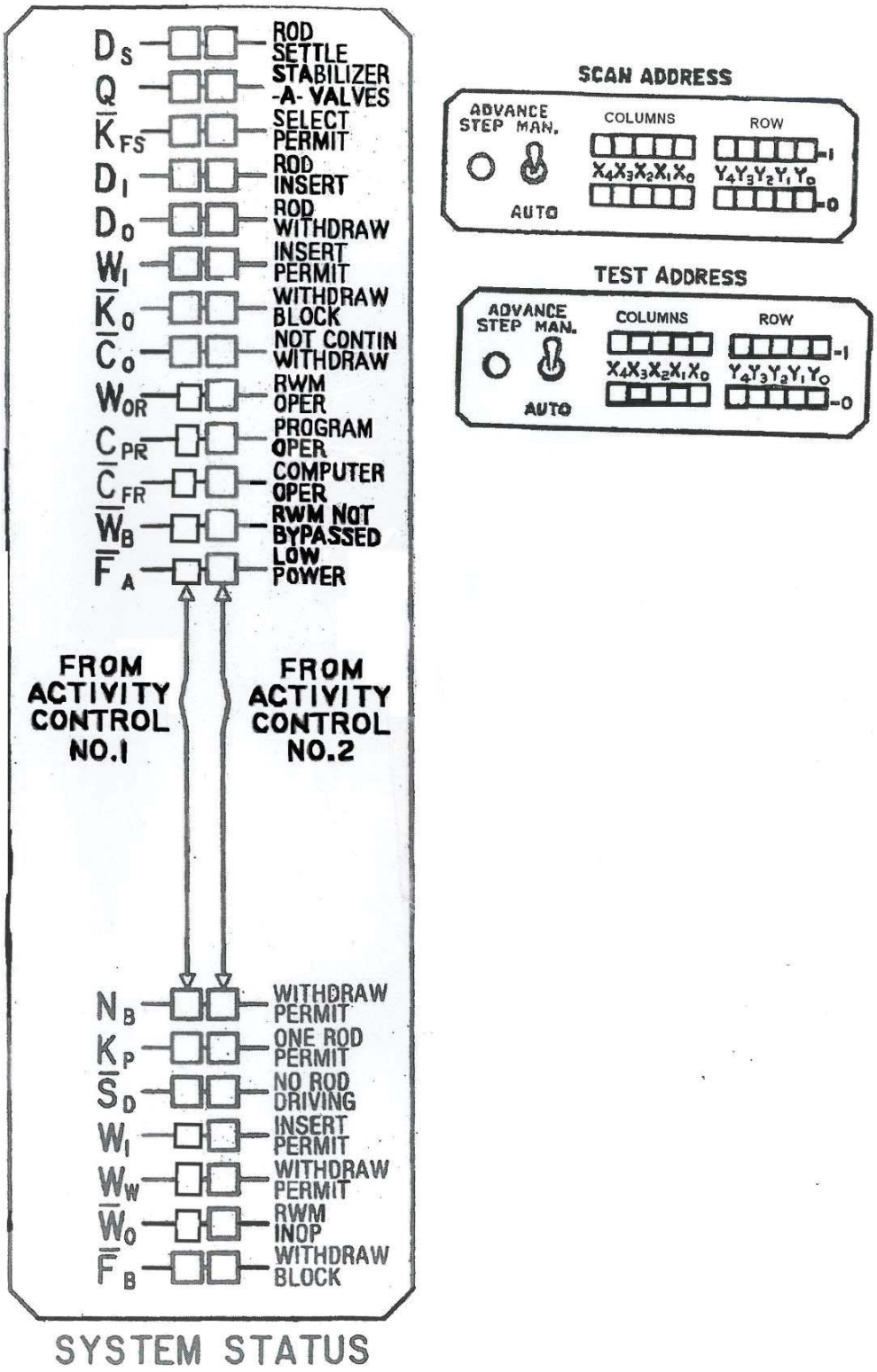
B. SYSTEM DESCRIPTION (Continued)

B.7.2. Analyzer Page Indicators



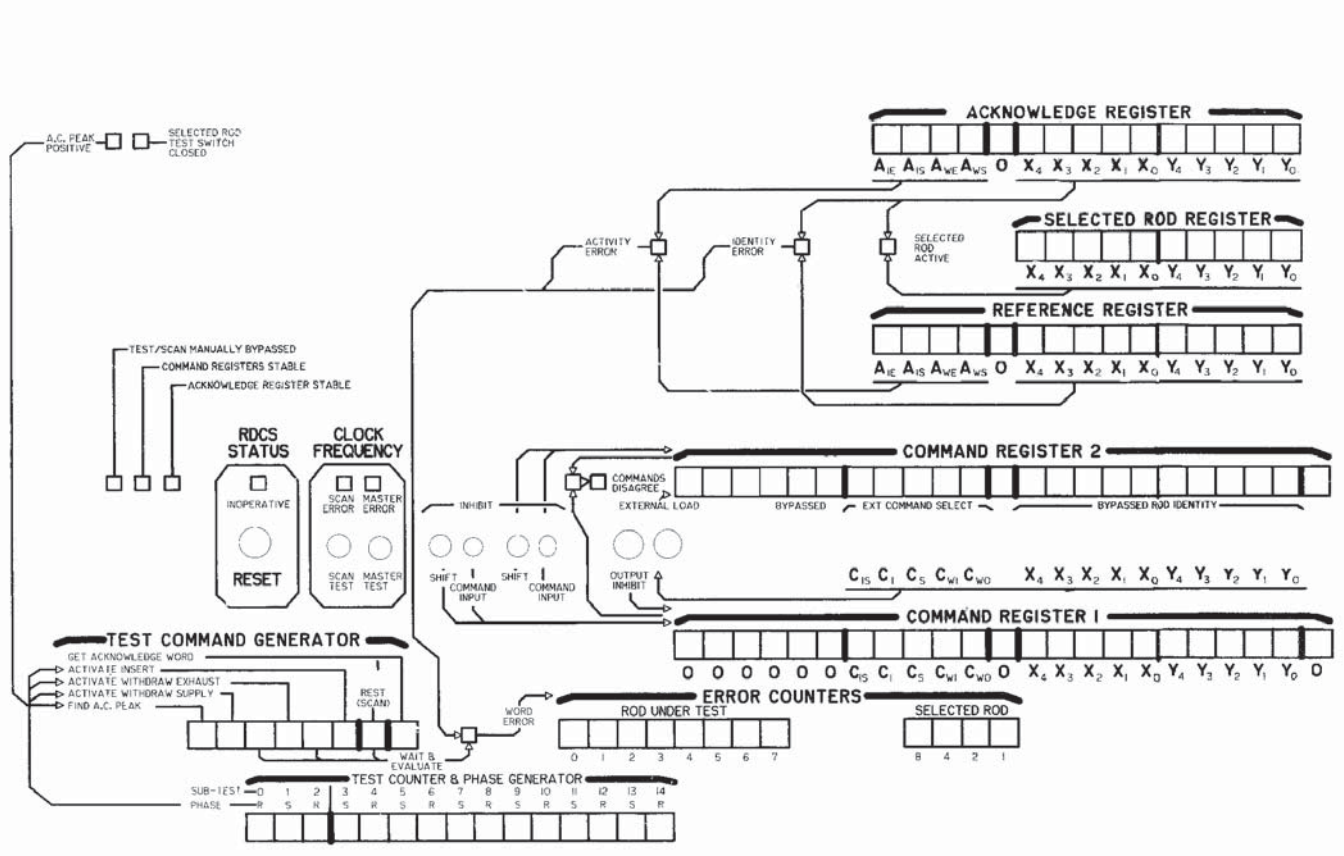
B. SYSTEM DESCRIPTION (Continued)

B.7.2 (Continued)

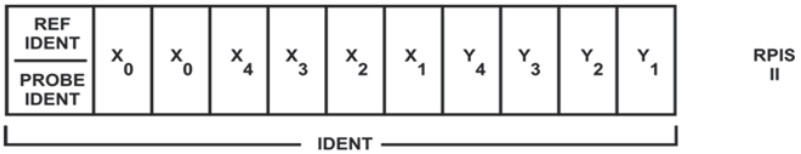


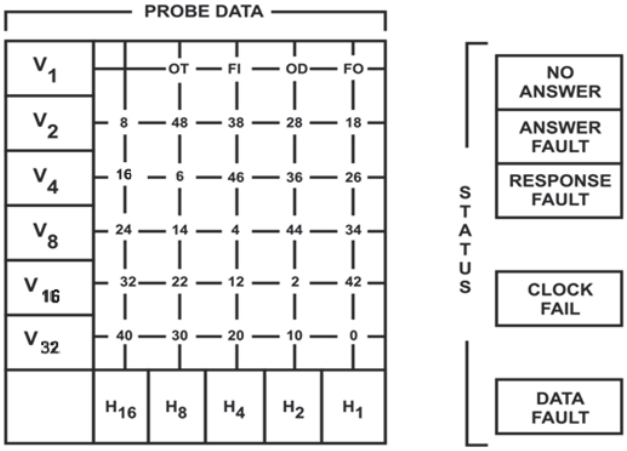
B. SYSTEM DESCRIPTION (Continued)

B.7.2 (Continued)



B.7.3. RPIS Display Card Indications





B. SYSTEM DESCRIPTION (Continued)

C. OPERATING REQUIREMENTS

The following systems should be in operation:

- N2-OP-30, Control Rod Drive
- N2-OP-71C, 600V A.C. Power Distribution
- N2-OP-92, Neutron Monitoring
- N2-OP-95A, Rod Worth Minimizer System
- N2-OP-97, Reactor Protection System

D. PRECAUTIONS AND LIMITATIONS

D.1. Use caution when using Continuous Rod Withdrawal in order to avoid large reactivity transients. Continuous Rod Withdrawal should **not** be used when approaching criticality.

D.2. Monitor the appropriate nuclear instrumentation response closely during **any** rod movement.

D.3. During reactor startup and shutdown, the selected control rod sequence must be strictly adhered to, in order to prevent local fuel damage. If the sequence **cannot** be adhered to, the Reactor Engineer shall be contacted for instructions. Refer to Technical Specification 3.1.6, Rod Pattern Control.

D.4. Whenever a rod is withdrawn to the full-out position, a coupling check must be performed.

D.5. Use CAUTION when using the Continuous Insert push button for rod movement, because the rod motion timer is bypassed and there is **no** settle function. Fully inserting a rod in this manner could cause damage to the CRD seals.

D.6. Under certain conditions, operation may continue with the Rod Worth Minimizer inoperable. However, additional restrictions must be applied. Refer to section H. of N2-OP-95A (RWM).

D.7. Unless specifically directed by the Emergency Operating Procedures, using the individual scram test switches to insert control rods to rapidly reduce power is prohibited. This method of power reduction can result in high rod worths, which could result in fuel damage.

D.8. Do **not** replace a light bulb on the rod select module for a selected rod. With the lens cover removed on the select module and that rod selected, damage could occur to the reactor manual control system should an object fall onto the exposed contacts. A different rod should be selected prior to removing the lens cover for a rod on the select matrix.

D.9. When a rod is bypassed in the RDCS, this rod becomes transparent to the control system. The rod will **not** function in the operator control scan, or test modes. Associated HCU status checks and other electronic checks performed by the system will be unavailable for the bypassed rod (**no** accumulator alarms). Refer to Technical Specification 3.1.5 (Modes 1 and 2) or 3.9.5 (Mode 5) for actions. In lieu of declaring the accumulator inoperable, local monitoring of the accumulator indications is permitted.

D.10. If in Mode 5, the Mode switch must be locked in REFUEL prior to withdrawing a control rod and One-Rod-Out Interlock shall be operable. (N2-OSP-RMC-W@002) (Technical Specification 3.9.2).

D.11. If in Mode 5, whenever a control rod is withdrawn from position 00, the FULL IN light **not** lit must be verified with a valid position observed on the four rod display (Technical Specification 3.9.4).

E. STARTUP

E.1. **VERIFY** the required support systems listed in Section C are in operation. _____

E.2. **COMPLETE** N2-OP-96-LINEUPS, Attachment 1, N2-OP-96 Walkdown Electric Lineup. _____

E.3. **TURN** the power switch for the Rod Select Module to ON (fully clockwise). _____

E.4. **VERIFY** that all alarms
AND rod blocks are clear,
EXCEPT for those expected for the given plant conditions. _____

E.5. **VERIFY** all required surveillance tests have been performed. _____

F. NORMAL OPERATIONS

F.1. Control Rod Selection

F.1.1. At 2CEC*PNL603 **SELECT** the rod to be moved by **MOMENTARILY DEPRESSING** the pushbutton for that rod on the Rod Select Module. _____

CAUTION

Do **not** replace a light bulb on the rod select module for a selected rod. With the lens cover removed on the select module and that rod selected, damage could occur to the reactor manual control system should an object fall onto the exposed contacts. A different rod should be selected prior to removing the lens cover for a rod on the select matrix.

F.1.2. **OBSERVE** the following:

- The pushbutton for selected rod is lit..... ☐
- The white select light on full core display is lit for control rod. ..... ☐
- The rod's position is indicated **AND** in correct location on 4-Rod Display. ..... ☐
- **IF** available, **THEN** Rod Worth Minimizer indicates correct rod **AND** rod position **OTHERWISE MARK** this step N/A..... ☐

_____

F.2. **Control Rod Withdrawal**

F.2.1. **PERFORM** Notch Withdrawal as follows:

a. **VERIFY** Reactor mode switch is **not** in SHUTDOWN
AND there are **no** rod withdrawal blocks. _____

b. **IF** in Mode 5,
THEN VERIFY Mode Switch is locked in REFUEL
AND One-Rod-Out Interlock is operable,
OTHERWISE MARK this step N/A. _____

c. **VERIFY** correct rod has been selected per sequence being used. _____

d. **WITHDRAW** selected control rod as follows:

1) **DEPRESS** WITHDRAW push button. ..... ☐

2) **CONFIRM** Rod motion as indicated by:

- Rod position indication on 4-Rod Display ..... ☐
- **OR** Withdraw light is illuminated. ..... ☐

3) **RELEASE** WITHDRAW pushbutton. ..... ☐

4) **OBSERVE** SETTLE light illuminated. ..... ☐ _____

NOTE

Control rods are more likely to double notch at elevated drive water differential pressures.

e. **IF** control rod failed to withdraw,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.2.1.d,
OTHERWISE MARK this step N/A. _____

f. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.1 (Continued)

g. **IF** control rod fails to withdraw at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

h. **OBSERVE** selected rod moves out one notch to next even-numbered position on 4-Rod Display. _____

i. **IF** in Mode 5
AND withdrawn rod was at position 00,
THEN VERIFY FULL IN light is **not** lit
AND a valid position is observed on four rod display,
OTHERWISE MARK this step N/A. _____

j. **MONITOR** appropriate nuclear instruments for proper response. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.2. **PERFORM** Continuous Withdrawal as follows: **[C3]**

a. **VERIFY** Reactor mode switch is **not** in SHUTDOWN
AND there are **no** rod withdrawal blocks. _____

b. **IF** in Mode 5,
THEN VERIFY Mode Switch is locked in REFUEL
AND One-Rod-Out Interlock is operable,
OTHERWISE MARK this step N/A. _____

c. **VERIFY** correct rod has been selected per sequence being used
AND that continuous rod withdrawal is allowed for given conditions. _____

NOTE

If control rood being withdrawn has a final target position of 48, then coupling check per 0 may be performed concurrently with step F.2.2.d.

d. **SIMULTANEOUSLY DEPRESS** WITHDRAW
AND CONTINUOUS WITHDRAW push buttons
AND OBSERVE WITHDRAW
AND CONTINUOUS WITHDRAW lights are on. _____

NOTE

- Control rods are more likely to double notch at elevated drive water differential pressures.
- Drive Speed will be faster when at elevated drive water pressure, therefore single notch movement should be used.

e. **IF** control rod failed to withdraw,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.2.1.d to perform single Notch Withdrawal of the stuck control rod,
OTHERWISE MARK this step N/A. _____

f. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.2 (Continued)

g. **IF** control rod fails to withdraw at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

h. **OBSERVE** selected rod moves continuously outward on 4-Rod Display. _____

i. **IF** in Mode 5
AND withdrawn rod was at position 00,
THEN VERIFY FULL IN light is **not** lit
AND a valid position is observed on four rod display,
OTHERWISE MARK this step N/A. _____

j. **MONITOR** appropriate nuclear instruments closely for proper response. _____

NOTE

- For control rods with a target position other than position 48, both pushbuttons should be released to allow the control rod to settle 2 notches prior to the target position, to avoid overshooting the target.
- Exceptions to this guidance can be taken as authorized by the reactor engineer when it is undesirable to have a control rod settle in an intermediate position.

k. **RELEASE BOTH** push buttons
AND ALLOW rod to settle into desired notch
AND OBSERVE proper cycling of INSERT, WITHDRAW,
AND SETTLE lights. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.3. **WHEN** a control rod is withdrawn to position 48,
THEN PERFORM a Control Rod Coupling Check as follows:

a. **ATTEMPT** to withdraw the Control Rod past position 48 by **DEPRESSING AND HOLDING BOTH** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons
AND OBSERVE the following:

- Position 48 shows on Four Rod Display. ..... ☐
- Annunciator 603444, CONTROL ROD OVERTRAVEL, remains clear..... ☐
- Control rod FULL OUT indicating light illuminated on Full Core Display..... ☐
- Any indicated response on nuclear instrumentation while withdrawing control rod..... ☐ _____

b. **RELEASE** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons. _____

F.3. **Control Rod Insertion**

N/A F.3.1. **PERFORM** Notch Insertion as follows:

a. **VERIFY** there are **no** rod insert blocks. N/A

b. **VERIFY** correct rod has been selected per sequence being used. _____

c. **INSERT** selected control rod as follows:

1) **DEPRESS** INSERT push button. ..... ☐

2) **CONFIRM** Rod motion as indicated by:

- Rod position indication on 4-Rod Display ..... ☐
- **OR** INSERT light is illuminated..... ☐

3) **RELEASE** INSERT pushbutton. ..... ☐

4) **OBSERVE** SETTLE light illuminated. ..... ☐ _____

d. **OBSERVE** selected rod moves in one notch to next even-numbered position on 4-Rod Display. _____

e. **MONITOR** appropriate nuclear instruments for proper response. _____

f. **IF** control rod failed to insert,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.3.1.c,
OTHERWISE MARK this step N/A. _____

g. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

h. **IF** control rod fails to insert at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.2 **PERFORM** Continuous Insertion as follows: **[C3]**

CAUTION

Do **not** use the Continuous Insert push button for normal rod movement. The rod motion timer is bypassed and there is **no** settle function, fully inserting a rod in this manner could cause damage to the CRD seals.

a. **VERIFY** there are **no** rod insert blocks. _____

b. **VERIFY** correct rod has been selected per sequence being used. _____

c. **DEPRESS AND HOLD** INSERT push button
AND OBSERVE INSERT light remains on as long as push button is held down. _____

d. **OBSERVE** selected rod moves continuously inward on 4-Rod Display. _____

NOTE

- For control rods with a target position other than position 00, the pushbutton should be released to allow the control rod to settle 2 notches prior to the target position, to avoid overshooting the target.
- Exceptions to this guidance can be taken as authorized by the reactor engineer when it is undesirable to have a control rod settle in an intermediate position or when performing control rod exercising.
- For control rods with a target position of 00, the pushbutton should be released to allow the control rod to settle into position 00.

e. **RELEASE** push button
AND ALLOW rod to settle into desired notch
AND OBSERVE proper cycling of INSERT
AND SETTLE lights. _____

f. **MONITOR** appropriate nuclear instruments for proper response. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.2 (Continued)

NOTE

- Control rods are more likely to double notch at elevated drive water differential pressures.
- Drive Speed will be faster when at elevated drive water pressure, therefore single notch movement should be used.

g. **IF** control rod failed to insert,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.3.1.3 to perform single Notch Withdrawal of the stuck control rod,
OTHERWISE MARK this step N/A. _____

h. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

i. **IF** control rod fails to insert at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

G. SHUTDOWN

NOTE

The Reactor Manual Control and Rod Position Indication System will normally remain in continuous operation.

G.1. **DE-ENERGIZE** the power supplies listed in N2-OP-96-LINEUPS, Attachment 1. _____

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-3
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 100% power. 2MSS*PSV129 is inoperable due to failed 'A' and 'B' solenoids. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.		
Turnover:	Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7. Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA.		
Critical Tasks:	CT-1: Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout.		
	CT-2: Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV. Reactor water level must not be below -14" for more than 15 minutes.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Secure Raising Suppression Pool Water Level with HPCS N2-OP-33
2	N/A	R – ATC, SRO	Lower Reactor Power with Recirculation Flow N2-OP-101D
3	Overrides	C – SRO TS – SRO	SLC Squib Valve Continuity Loss ARP 601701, Technical Specifications
4	NM12	I – ATC, SRO	APRM Fails Downscale N2-OP-92
5	AD05	C – All MC – BOP TS – SRO	ADS SRV Inadvertently Opens N2-SOP-34, N2-SOP-101D, Technical Specifications
6	ED02 EG01	C – All	Loss of Offsite Power N2-SOP-03, N2-SOP-101C, N2-EOP-RPV
7	DG04	C – BOP, SRO MC – BOP	Emergency Diesel Generators Fail to Automatically Start N2-SOP-03, N2-SOP-01
8	RR20	M – All MC – All	Coolant Leak in Drywell N2-EOP-RPV, N2-EOP-PC
9	RC01 CS02 RH14	C – All MC – All	RCIC Fails to Automatically Start, HPCS Fails to Automatically Inject, Low Pressure ECCS Systems Fail to Automatically Inject N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-3**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 7 & 9**	2	
2. Abnormal events (2-4) **Events 5 & 6**	2	
3. Major transients (1-2) **Event 8**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-RPV, EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)	0	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 3** Revision: **0.0**

Title: **November 2023 NRC Scenario 3**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/11/23
	N/A / Dan Delaney	7/11/23
	N/A / Mike Carno	7/11/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-33, High Pressure Core Spray System
2. N2-OP-101D, Power Changes
3. N2-OP-92, Neutron Monitoring
4. N2-SOP-34, Stuck Open Safety Relief Valve
5. N2-SOP-101D, Rapid Power Reduction
6. N2-SOP-29, Sudden Reduction in Core Flow
7. N2-SOP-08, Unplanned Power Changes
8. N2-SOP-03, Loss of AC Power
9. N2-SOP-101C, Reactor Scram
10. N2-EOP-RPV (1-3), RPV Control - Flowchart
11. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
12. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
13. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
14. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
15. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 100% power with 2MSS*PSV129 inoperable. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.

   b. Event 1 is the normal evolution performed by the BOP operator to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0.

   c. Event 2 is a reactivity evolution. The ATC operator will lower Reactor power to 95% using Recirculation flow per the provided ReMA.

   d. Event 3 occurs when an SLC squib valve loses continuity. The SRO will evaluate Technical Specifications.

   e. Event 4 occurs while the power reduction is in progress. APRM 2 will fail downscale. The SRO will evaluate Technical Specifications. The crew will bypass APRM 2 per N2-OP-92.

   f. Event 5 occurs when 2MSS*PSV127, an ADS SRV, inadvertently opens. The crew will execute N2-SOP-34. The crew to reduce Reactor power to approximately 90% per N2-SOP-101D. The crew will be able to close the open SRV. During the transient heat will be added to the suppression pool, requiring entry into N2-EOP-PC (1-3). The crew will place Suppression Pool cooling in service. The SRO will evaluate Technical Specifications.

   g. Events 6 & 7 begin when a loss of all off-site power occurs. 115 KV line 5 will be lost first, followed shortly by 115KV line 6. A Main Generator load reject results in a Reactor scram. Div 1 and II Diesel Generators will fail to automatically start. The crew must manually start one or more Diesel Generators *(critical task)*. The crew will

execute N2-SOP-03, N2-SOP-101C, and N2-EOP-RPV (1-3) to stabilize the plant.

h. Events 8 & 9 occur when a coolant leak develops in the Drywell. RCIC, HPCS, and low pressure ECCS will fail to automatically initiate. The crew will control injection system to restore and maintain Reactor water level *(critical task)*.

2. Termination Criteria

a. All control rods inserted.

b. Div I and/or Div II EDG powering their respective bus(es).

c. Reactor water level being controlled above top of active fuel.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because loss of all AC power significantly degrades the ability to control the plant.*
Initiating Cue:	*Multiple annunciators and equipment losses will indicate the loss of all AC power. N2-SOP-03 and N2-SOP-01 provide direction to start an EDG and restore power.*
Success Path:	*Manually start Div I and/or Div II EDG.*
Performance Feedback:	*Restoration of AC voltage and multiple systems will provide performance feedback to the crew regarding the start of EDG(s).*
Measurable Performance Standard:	*Manipulation of EDG controls will provide observable action for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

CT-1.0, Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because maintaining Reactor water level above -14 inches ensures adequate core cooling through the preferred method of core submergence. This protects the integrity of the fuel cladding.*

Initiating Cue:	*Multiple Reactor water level indicators and annunciators will provide indications of lowering Reactor water level. N2-EOP-RPV (1-3) provides multiple procedure steps directing injection.*
Success Path:	*Manually injection with RCIC, HPCS, SLC, and/or low pressure systems.*
Performance Feedback:	*Multiple Reactor water level indicators and annunciators will provide performance feedback regarding the success of injection with injection systems.*
Measurable Performance Standard:	*Manipulation of pumps and/or valves in the injection system(s) will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

a. CT-2.0, Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV (1-3). Reactor water level must not be below -14" for more than 15 minutes.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-SE-01, station blackout

   b. DMS-SE-02, loss of off-site power or loss of grid

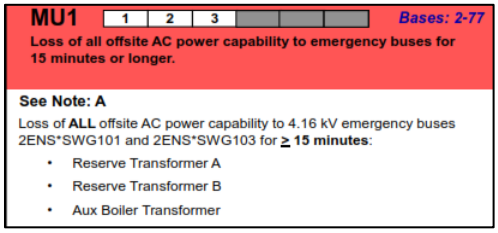
   c. DMS-RL-01, LOCA; RPV level maintained above TAF using high pressure preferred injection and/or high-pressure alternate injection systems; containment parameters controlled; no blowdown

6. Technical Specifications (Applicable actions for initial conditions only)

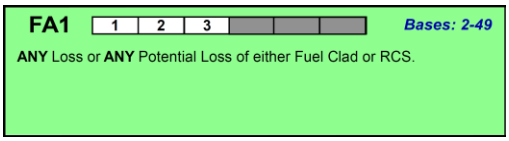
   a. Technical Specification 3.5.1 Condition B (14 days) was entered 1 hour ago.

7. EAL Classification

   a. Unusual Event, EAL MU1:

1) 

b. Alert, EAL FA1:

1) 

B. Initial Conditions

1. IC Number

   a. IC-021 or equivalent (IC-202 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

      1) **NM12B**, APRM Channel Failure – Downscale (2), Final = true — **TRG2**

      2) **ED02A**, Loss Of Off-Site 115KV Line 5, Final = true — **TRG5**

      3) **ED02B**, Loss Of Off-Site 115KV Line 6, Final = true, Delay = 30 seconds — **TRG5**

      4) **DG04A**, EDG 1 fail to UV/LOCA Auto-Start, Final = true — **Inserted**

      5) **DG04B**, EDG 3 fail to UV/LOCA Auto-Start, Final = true — **Inserted**

      6) **EG01**, Main Generator Trip, Final = true, Delay = 30 seconds — **TRG5**

      7) **RR20**, RR Loop Rupture – DBA LOCA, Final = 0.1 — **TRG6**

      8) **RC01**, Reactor Core Isolation Cooling Auto Start Failure, Final = true — **Inserted**

9)	**CS02**, High Pressure Core Spray Fails to Auto Start, Final = true	**Inserted**
10)	**RH14A**, ECCS Fails to Initiate (Div I), Final = true	**Inserted**
11)	**RH14B**, ECCS Fails to Initiate (Div II), Final = true	**Inserted**
12)	**SL02A**, 2SLS*VEX3A Fails to Fire When SLC Pump is Turned On, Final = true	**TRG1**
13)	**ann601701,** div 1 slc system inoperable, Final = crywolf	**TRG1**
b.	Remotes	
1)	**RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = online	**TRG10**
2)	**RM02-041**, SWP23B Current Radiation Level Online, Final = online	**TRG11**
3)	**RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, Final = on	**TRG10**
4)	**RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, Final = on	**TRG11**
c.	Overrides	
1)	**DO-1168**, OFF SQUIB VLV A READY (F004A) (WHITE), Final = off	**TRG1**
2)	**DI-358**, ON SLC SQUIB VALVE VEX3A INOP (AMBER), Final = on	**TRG1**
3)	**DI-1171**, OPEN ADS VALVE PSV 127, Final = on	**TRG3**
4)	**DI-2351**, OPEN ADS VALVE PSV 127, Final = on	**TRG3**
5)	**DI-3215**, INSERT FUSE, Final = off	**Inserted**
6)	**DI-3216**, INSERT 9 SOL B FUSE, Final = off	**Inserted**
7)	**DI-2095**, INSERT FUSE, Final = off	**Inserted**
8)	**DI-2096**, INSERT FUSE, Final = off	**Inserted**

d. Annunciators

1) None

e. Event Triggers

Event #	Event Action	Command
None		

f. Equipment Out of Service

1) 2MSS*PSV129 inop

g. Support Documentation

1) Provide N2-OP-33 section H.3.0 in progress up to step H.3.7.

2) Provide ReMA for lowering power to 95% with Recirc flow

3) Provide N2-OP-101D completed up to step F.3.6.b.3

h. Miscellaneous

1) Place clearance reference tags on the 2MSS*PSV129 control switches on P628 and P631

2) Place a clearance reference tag on the vertical section of P601 near the 2MSS*PSV129 solenoid indicating light

3) Protect RCIC

C. Shift Turnover Information

1. Reactor Power: 100%

2. Rodline: Above 100%

3. Technical Specification LCOs in effect:

a. Technical Specification 3.5.1 Condition B (14 days) was entered 1 hour ago

4. Significant Problems / Abnormalities / Equipment Out of Service:

   a. 2MSS*PSV129 is inoperable.

   b. Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0.

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7.

   b. Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA.

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 100%
- 2MSS*PSV129 is inoperable due to failed 'A' and 'B' solenoids
- Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.

PART III: Remarks/Planned Evolutions:

- Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7.
- Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA. N2-OP-101D is in progress up to step F.3.6.b.3 in preparation for the power reduction.

Event #1 – Secure Raising Suppression Pool Water Level with HPCS

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~100% power - Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0 • Verifiable Actions - Secure raising of Suppression Pool water level with HPCS • Final (expected) operating result - Reactor at ~100% power - HPCS in standby lineup

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0, starting at step H.3.7 • Once lineup is secured, exits Technical Specification 3.5.1 Condition B
	BOP • Acknowledges direction to secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0, starting at step H.3.7: • WHEN desired Suppression Pool Level is reached, THEN CLOSE 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL • WHEN HPCS SYSTEM FLOW is less than 634 GPM, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, is OPEN • STOP 2CSH*P1, HPCS PUMP 1 • VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV, is CLOSED • NOTIFY SM that 2CSH*MOV111 is CLOSED AND operability concern PER IR 02083078 no longer exists • RESTORE CST Level, as required PER N2-OP-4

Event Termination Criteria	• Raising of Suppression Pool water level with HPCS secured

Event #2 – Lower Reactor Power with Recirculation Flow

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~100% power • Verifiable Actions - Lower Recirculation flow • Final (expected) operating result - Reactor power at ~95%

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to lower Reactor power to 95% using Recirc flow per ReMA and N2-OP-101D • Provides oversight for reactivity maneuver
Note: T.S. 3.4.1 requires that loop flow mismatch be maintained ≤ 5% of rated core flow (108.5 mlbm/hr) when operating at an effective core flow ≥ 70% of rated core flow. In this case the loop flow mismatch allowable would be 5.425 mlbm/hr.	**ATC** • Acknowledges direction to lower Reactor power to 95% using Recirc flow. • Lowers power to 95% by lowering core flow as follows: ◦ Moves 2RCS*HYV17A&B individually in the close direction, maintaining loop flow differential at a minimal value by alternating between the two valves. • Monitors APRMs and rate of power change
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• Reactor power has been lowered sufficiently as determined by the Lead Evaluator

Event #3 – SLC Squib Valve Continuity Loss

Event Information	• Presumed or required initial plant operating conditions - Both SLC systems in a normal standby lineup • Verifiable Actions - Technical Specifications addressed • Final (expected) operating result - SLC system A declared inoperable

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td><u>Booth Operator</u>
When directed by the lead evaluator, insert the following malfunction and overrides:
TRG1 SL02A, 2SLS*VEX3A Fails to Fire When SLC Pump is Turned On, Final = true
ann601701, div 1 slc system inoperable, Final = crywolf
DO-1168, OFF SQUIB VLV A READY (F004A) (WHITE), Final = off
DI-358, ON SLC SQUIB VALVE VEX3A INOP (AMBER), Final = on</td><td></td></tr>
<tr><td>2SLS*VEX3A loses power/continuity

The following annunciator alarms:
• 601701, DIVISION I STBY LIQ CONT SYSTEM INOPERABLE</td><td><u>ATC</u>
• Recognizes / reports annunciator
• Recognizes / reports loss of power/continuity to 2SLS*VEX3A</td></tr>
<tr><td><table><tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr><tr><td>3.1.7</td><td>A</td><td>A.1</td></tr><tr><th>Action</th><th colspan="2">Description</th></tr><tr><td>A.1</td><td colspan="2">Restore SLC subsystem to operable status (7 days)</td></tr></table></td><td><u>SRO</u>
• Acknowledges failure of 2SLS*VEX3A
• Refers to TS 3.1.7 and determines Condition A applies (7 days)</td></tr>
</table>

Instructor Actions / Plant Response	Operator Actions
Role Play: If requested to check milliamp reading on ammeter C41A-Z1A in back of 2CEC*PNL629, wait 1 minute, then report it indicated 0 mA. **Role Play:** If requested to check 2EHS*MCC102 BKR 16D, wait 2 minutes, then report the breaker is closed but the 10A fuse is blown.	**ATC** • Executes ARP • Coordinates with BOP / field operator to verify: ○ Milliamp reading on ammeter C41A-Z1A (0 amps)

Event Termination Criteria	• Technical Specifications addressed

Event #4 – APRM Fails Downscale

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~95% power • Verifiable Actions - APRM 2 bypassed • Final (expected) operating result - Reactor power at ~95% with APRM 2 bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction**: **TRG2** **NM12B**, APRM Channel Failure – Downscale (2), Final = true *APRM 2 fails downscale*	
The following annunciators alarm: • *603214, APRM TRIP SYSTEM DOWNSCALE* • *603442, CONTROL ROD OUT BLOCK* *The following computer points are generated:* • *RDSBC12, CONTROL ROD OUT BLOCK* • *NMP2C107, APRM DOWNSCALE ALARM* • *NMP2BC23, APRM 2 NEUTRON FLUX DOWNSCALE*	**ATC** • Recognizes / reports annunciators • Recognizes / reports APRM 2 indicates downscale
	SRO • Acknowledges failure of APRM 2 • Evaluates Technical Specification 3.3.1.1 • Directs bypass of APRM 2 per N2-OP-92

Instructor Actions / Plant Response	Operator Actions
	ATC • Bypasses APRM 2 per N2-OP-92 section H.2.0: • Verifies no other APRM in bypass • Performs a channel check to verify no other APRM is inoperable for the division being bypassed (APRMs within 2%) • Places APRM BYPASS joystick to the 2 position • Verifies APRM 2 BYPASS light is lit on 2CEC*PNL603 • Coordinates with BOP to verify the following: ◦ BYP is displayed in inverse video in the header for APRM 2 Chassis at H13-P608 ◦ Blue BYPASSED LED is lit for APRM 2 on each 2/4 MDL at H13-P608
Role Play: As the booth operator, inform the BOP that APRM 2 indicates bypassed on all 2/4 MDL modules.	**BOP** • Monitors plant parameters to verify proper operations • Provides peer checks as needed • Coordinates with ATC to verify the following: ◦ BYP is displayed in inverse video in the header for APRM 2 Chassis at H13-P608 ◦ Blue BYPASSED LED is lit for APRM 2 on each 2/4 MDL at H13-P608

Event Termination Criteria	• APRM 2 is bypassed

Event #5 – ADS SRV Inadvertently Opens

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~95% power - 2MSS*PSV129 inoperable • Verifiable Actions - Close 2MSS*PSV127 - Technical Specification 3.5.1 evaluated • Final (expected) operating result - Reactor at ~90% power - 2MSS*PSV127 closed - Suppression Pool Cooling in service, if required

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **overrides**: **TRG3** **DI-1171**, OPEN ADS VALVE PSV 127, Final = on **DI-2351**, OPEN ADS VALVE PSV 127, Final = on *2MSS*PSV127 opens*	
The following annunciators alarm: • *602553, ADS ACC TANK 32/33/34 PRESSURE LOW (comes in then clears)* • *601537, ADS VALVES / SAFETY VALVES LEAKING* • *601548, SAFETY / RELIEF VALVE OPEN* • *601543, SAFETY RELIEF VALVE SWITCH IN OFF POSITION (When switch taken to OFF)* *The following computer points are generated:* • *SVVBC08, SAF/RLF VLV PSV127 STAT (Open)* • *ADSBC20, ADS VLVS/SAF VLVS LEAK (Alarm)*	**Crew** • Recognizes / reports annunciators • Recognizes / reports 2MSS*PSV127 is open

Instructor Actions / Plant Response	Operator Actions
The following annunciator alarms ~ 3 minutes after event initiation: • *601560, Suppression Pool Water Temp High* *Suppression Pool temperature exceeds 90°F* **Note:** Based on the timeliness of crew actions, the crew may not reach 90°F in the suppression pool and therefore will not enter N2-EOP-PC (1-3).	**SRO** • Acknowledges reports • Directs BOP to execute N2-SOP-34 • Directs ATC to lower Reactor power to 90% per N2-SOP-101D • Provides oversight of reactivity manipulation • May enter N2-EOP-PC (1-3) on high Suppression Pool temperature • May direct initiation of Suppression Pool Cooling • May direct start of 5th Service Water pump • Refers to TS 3.5.1 and determines Condition E applies due to 2 of 7 ADS SRVs inoperable (14 days) • If places one division of RHR in Suppression Pool Cooling, determines TS 3.5.1 Condition F applies (72 hours) • If places two divisions of RHR in Suppression Pool Cooling, determines TS 3.5.1 Condition H applies (enter LCO 3.0.3 immediately) • If Suppression Pool average temperature exceeds 90°F, determines TS 3.6.2.1 Condition A applies (24 hours)

Spec	Condition	Applicable Actions
3.5.1	E	E.1
Action	**Description**	
E.1	Restore ADS valve to operable status (14 days)	

Spec	Condition	Applicable Actions
3.5.1	F	F.1 or F.2
Action	**Description**	
F.1	Restore ADS valve to operable status (72 hours)	
F.2	Restore low pressure ECCS injection / spray subsystem to operable status (72 hours)	

Spec	Condition	Applicable Actions
3.5.1	H	H.1
Action	**Description**	
H.1	Enter LCO 3.0.3 (immediately)	

Spec	Condition	Applicable Actions
3.6.2.1	A	A.1 and A.2
Action	**Description**	
A.1	Verify SP average temp ≤110°F (once per hour)	
A.2	Restore SP average temp ≤90°F (24 hours)	

Instructor Actions / Plant Response	Operator Actions
	ATC • Lowers Reactor power to approximately 90% using Recirc flow per N2-SOP-101D
Note: The candidate may remove the SRV fuses in any order. The scenario is scripted in the order listed in N2-SOP-34. The SRV will close when the 'A' and 'B' solenoid fuses are pulled.	**BOP** • Executes N2-SOP-34: ◦ Identifies 2MSS*PSV127 as the open SRV ◦ Places the keylock switch for 2MSS*PSV127 in Off ◦ Determines 2MSS*PSV127 failed to close by the use of one or more of the following indications: - SPDS Computer - ERF computer points MSSZC111 and MSSZC128 - Reactor power change - Generator output change - Steam flow/Feed flow mismatch - Acoustic monitor ◦ Informs crew of need to reduce Reactor power to approximately 90% ◦ Using detail 2 of N2-SOP-34 removes fuses in any of the following order until the SRV closes: - 'C' solenoid fuses - 'A' solenoid fuses - 'B' solenoid fuses ◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'C' solenoid fuses is panel 628 strip 'A' fuse numbers F49 & F50 ◦ Removes fuses F49 & F50 ◦ Determines that 2MSS*PSV127 did not close by pulling the 'C' solenoid fuses ◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'A' solenoid fuses is panel 628 strip 'K' fuse numbers F5A & F6A ◦ Removes fuses F5A & F6A

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched as field operator to perform pre-start checks on 2SWP-P1A(B-E), wait 3 minutes then report that 2SWP-P1A(B-E) is ready for start and that all pre-start checks are completed. **Role Play:** If requested to place SWP radiation monitor RE23A in service, wait 2 minutes and **insert** the following **remote**: **TRG10** **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, FINAL=On **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Sample Pmp Power, FINAL=On Then report 2SWP*RE23A is in service. **Role Play:** If requested to place SWP radiation monitor RE23B in service, wait 2 minutes **insert** the following **remote**: **TRG11** **RM02-041**, SWP23B Current Radiation Level Online, FINAL=On **RM03-041**, SWP23B Current Radiation Level Sample Pmp Power, FINAL=On Then report 2SWP*RE23B is in service	◦ Determines, using detail 2 of N2-SOP-34 that the location of the 'B' solenoid fuses is panel 631 strip 'C' fuse numbers F5B & F6B ◦ Removes fuses F5B & F6B ◦ Determines that 2MSS*PSV127 did close by pulling the 'A' and 'B' solenoid fuses ◦ Informs crew that 2MSS*PSV127 closed after pulling the 'A' and 'B' solenoid fuses • If directed, initiates Suppression Pool Cooling per N2-OP-31 section F.4.0: ◦ Notifies Shift Manager to declare RHS A(B) LPCI mode inoperable ◦ Directs radiation protection department to start SWP radiation monitors 2SWP*RE23A(B) ◦ Opens 2SWP*MOV90A(B) ◦ May start 2SWP*P1E(F) ◦ Throttles open 2SWP*MOV33A(B) to establish 7400 gpm service water flow to RHR A(B) ◦ Verifies 2RHS*MOV24A(B) closed ◦ Verifies 2RHS*MOV4A(B) open ◦ Starts 2RHS*P1A(B) ◦ Throttles open 2RHS*FV38A(B) to establish 7450 gpm RHR A(B) total flow ◦ Verifies 2RHS*MOV4A(B) closed ◦ May throttle 2RHS*MOV8A(B) closed

Event Termination Criteria	• 2MSS*PSV127 closed • Suppression Pool Cooling is in service, if required • Technical Specifications have been addressed

Events #6, 7, 8, and 9 – Loss of Offsite Power; Emergency Diesel Generators Fail to Automatically Start; Coolant Leak in Drywell; RCIC Fails to Automatically Start, HPCS Fails to Automatically Inject, Low Pressure ECCS Systems Fail to Automatically Inject

Event Information	• Presumed or required initial plant operating conditions - Plant operating at power • Verifiable Actions - Emergency Diesel Generators started - Reactor water level controlled above top of active fuel - Suppression Pool and Drywell sprays initiated • Final (expected) operating result - Emergency buses energized. - All control rods inserted. - Reactor level being controlled in assigned band. - Containment pressure is being controlled per N2-EOP-PC (1-3). • Mitigation Strategy code (if applicable) - DMS-SE-01, station blackout - DMS-SE-02, loss of off-site power or loss of grid - DMS-RL-01, LOCA; RPV level maintained above TAF using high pressure preferred injection and/or high pressure alternate injection systems; containment parameters controlled; no blowdown
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given a Station Blackout and failure of EDGs to automatically start, the crew will manually start an EDG and re-energize the associated bus, in accordance with N2-SOP-03 and/or N2-SOP-01. A bus must be re-energized within 15 minutes of the start of the Station Blackout. 2. Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -14", in accordance with N2-EOP-RPV (1-3). Reactor water level must not be below -14" for more than 10 minutes.

Instructor Actions / Plant Response	Operator Actions
The following malfunctions are preset to cause the Div I and II EDGs to fail to start: **DG04A**, EDG 1 fail to UV/LOCA Auto-Start, Final = true **DG04B**, EDG 3 fail to UV/LOCA Auto-Start, Final = true *The following malfunctions are preset to cause RCIC, HPCS, LPCS, and LPCI automatic injection to fail:* **RC01**, Reactor Core Isolation Cooling Auto Start Failure, Final = true **CS02**, High Pressure Core Spray Fails to Auto Start, Final = true **RH14A**, ECCS Fails to Initiate (Div I), Final = true **RH14B**, ECCS Fails to Initiate (Div II), Final = true When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG5** **ED02A**, Loss Of Off-Site 115KV Line 5, Final = true **ED02B**, Loss Of Off-Site 115KV Line 6, Final = true, Delay = 30 seconds **EG01**, Main Generator Trip, Final = true, Delay = 30 seconds	
Note: Due to the loss of offsite power, multiple indications, including on R*Time, will be inaccurate or unavailable.	**Crew** • Recognizes / reports loss of offsite power • Recognizes / reports Reactor scram • Recognizes / reports failure of Div I and II EDGs to automatically start

Instructor Actions / Plant Response	Operator Actions
	ATC • Places the Reactor Mode Switch in the SHUTDOWN position • Arms and depresses RPS pushbuttons • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs closed ◦ Feedwater pumps are NOT running ◦ All control rods fully inserted • Reports EOP entry condition on low RPV water level • Performs N2-SOP-101C actions: ◦ Verifies: ○ All rods full in ○ Reactor power lowering ○ Turbine tripped/TSVs & TCVs shut ○ Generator tripped ○ SDV V & D Valves closed ○ RPV pressure on TBVs or SRVs ◦ Fully inserts IRMs and SRMs

Instructor Actions / Plant Response	Operator Actions
	SRO • Acknowledges reports • Directs entry into N2-SOP-101C • Enters N2-EOP-RPV (1-3) on low Reactor water level • Directs entry into N2-SOP-03 • **Directs start of Div I and II EDGs (CT-1.0)** • Performs N2-EOP-RPV (1-3) actions: ◦ Directs scram procedure entered (N2-SOP-101C) ◦ Performs level control actions: ○ Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1 ○ Determines that RPV water level is known ○ **Directs that RPV water level be restored and maintained between 159.3 and 202.3 inches using RCIC, HPCS, and/or SLC (CT-2.0)** ○ Directs an RPV level band of 160 – 200 inches ○ May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: ○ Determines that SRVs are cycling ○ Determines that RPV pressure can be stabilized below 1052 psig using SRVs ○ Directs a pressure band of 800-1000 psig

Instructor Actions / Plant Response	Operator Actions

Booth Operator:
If directed to defeat the HPCS Level 8 interlocks by placing HPCS INJ VLV LEVEL 8 BYPASS TEST SWITCH in the TEST position, **manually insert** the following **remote function**:
 	
• **CS14**, OPS-CSH01 PNL625 TST.SW CSH*MOV107, Final = Test
 	
Wait 1 minute, then report that the HPCS Level 8 interlocks have been defeated.
 	
Booth Operator:
If directed to defeat the RCIC Level 8 interlocks, **manually insert** the following **remote function**:
 	
• **RC02A,** RCIC Level 8 Trip Defeat: Withdraw Trip Units N693A, E, Final = defeated
 	
Wait 1 minute, then report that the RCIC Level 8 interlocks have been defeated.
 	
Booth Operator:
If directed to defeat the RHR group 5 isolation, **manually insert** the following **malfunction**:
 	
• **RH08,** GROUP 5 ISOLATION FAILURE – (RHS*MOV122/113), Final = true
 	
Wait 1 minute, then report that the RHR group 5 isolation has been defeated.
	ATC / BOP
	• May recognize / report failure of RCIC and/or HPCS to automatically start
	• **Control injection to restore / maintain Reactor water level between 159.3 and 202.3 inches using RCIC, HPCS, and/or SLC (CT-2.0)**
	• **May manually start RCIC (CT-2.0)**:
	  ◦ Arms and depresses RCIC MANUAL INITIATION pushbutton
	  ◦ Verifies proper RCIC response
	  ◦ Controls RCIC operation as follows:
	  ◦ Automatic control with thumbwheel
	  ◦ Manual control using the pushbuttons
	  ◦ If injection with reject to CST is required:
	  ▪ Opens 2ICS*MOV124, TEST BYPASS TO CONDENSATE STORAGE TK
	  ▪ Controls injection flow to the Reactor by throttling 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK
	• **May manually start HPCS (CT-2.0)**:
	• At 2CEC*PNL601, ARM AND DEPRESS MANUAL INITIATION
	• At 2CEC*PNL601, VERIFY the following:
	• 2CSH*P1, HPCS PUMP 1, started.
	• 2CSH*MOV107, PMP 1 INJECTION VLV, open.
	• WHEN system flow is greater than 634 gpm, THEN VERIFY 2CSH*MOV105, MINIMUM FLOW BYPASS VLV closed.
	• Flow to RPV as indicated by 2CSH*FI104 flow rising AND indication the following valves are closed:
	  ◦ 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL
	  ◦ 2CSH*MOV110, TEST RETURN TO CONDENSATE TK

Instructor Actions / Plant Response	Operator Actions
	◦ 2CSH*MOV112, TEST RETURN TO CONDENSATE TK • At 2CEC*PNL852, VERIFY DIVISION 3 2EGS*EG2 started
Once at least one EDG has been started and connected to its respective bus, and when directed by the lead evaluator, **insert** the following **malfunction**: **TRG6** **RR20**, RR Loop Rupture – DBA LOCA, Final = 0.1	**Crew** • Recognizes / reports degrading Primary Containment parameters
	SRO • Enters N2-EOP-PC (1-3) on high Drywell pressure and temperature • Re-enters N2-EOP-RPV (1-3) on high Drywell pressure • Performs N2-EOP-PC (1-3) primary containment pressure leg actions: • Determines cannot maintain containment pressure below 1.68 psig • Determines that suppression chamber spray is required before suppression chamber pressure reaches 10 psig • Determines that suppression pool water level is below EL 217 ft. • Determines that suppression chamber sprays are warranted • Directs RO to place RHR in suppression chamber sprays • Directs pneumatics restored to the Drywell • May direct low pressure ECCS pumps terminated and prevented with a high drywell pressure and reactor pressure lowering (N2-EOP-RPV (1-3), step P-1) • Directs Suppression Chamber Spray secured prior to Suppression Chamber pressure lowering to 0 psig

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If requested to place SWP radiation monitor RE23A in service, **insert** the following **remotes**: **RM02-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Online, Final = Online **RM03-040**, SWP23A SWP From RHR 'A' Ht Exch Rad Monit Power, Final = On If requested to place SWP radiation monitor RE23B in service, **insert** the following **remotes**: **RM02-041**, SWP23B Current Radiation Level Online, Final = Online **RM03-041**, SWP23B Current Radiation Level Power, Final = On Then report 2SWP*RE23A(B) is in service	**ATC / BOP** • Initiates Suppression Chamber spray per N2-EOP-6.22: ◦ Opens 2SWP*MOV90A(B) (may be delayed until after sprays are in service). ◦ Verifies 2RHS*MOV24A(B) overridden closed. ◦ Verifies 2RHS*P1A(B) running. ◦ Opens 2RHS*MOV33A(B) to establish suppression chamber spray flow. ◦ Opens 2RHS*FV38A(B) and establish approximately 7450 gpm – If suppression pool cooling is directed. ◦ Throttles open 2SWP*MOV33A(B) to establish flow not to exceed 7400 gpm - Starts 5th SWP pump (post LOCA). ◦ Closes 2RHS*MOV8A(B) - after 10 minute time delay. ◦ Directs RP to place 2SWP*RE23A(B) in service • Restores pneumatics to the drywell using N2-EOP-HC attachment 7 panel 601: ◦ At 2CEC*PNL851: ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV166 to OVERRIDE ◦ OPENS 2IAS*SOV166 ◦ At 2CEC*PNL601 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV164 to OVRD ◦ OPENS 2IAS*SOV164 ◦ PLACES LOCA OVERRIDE VLV 2IAS*SOV165 to OVRD ◦ OPENS 2IAS*SOV165 • If directed, terminates and prevents Low pressure ECCS injection using N2-EOP-HC attachment 4:

Instructor Actions / Plant Response	Operator Actions
	◦ PLACES 2CSL*P1, PMP 1, control switch in PULL-TO-LOCK ◦ IF a LPCI A/LPCS initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI A/LPCS MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24A, LPCI A INJECTION VLV ◦ 2CSL*MOV104, PMP 1 INJECTION VLV ◦ PLACES 2RHS*P1C, PMP 1C, control switch in PULL-TO-LOCK. ◦ IF a LPCI B & C initiation signal is NOT present, THEN ARMS AND DEPRESSES LPCI B & C MANUAL INITIATION pushbutton ◦ CLOSES AND, IF possible, OVERRIDES: ◦ 2RHS*MOV24B, LPCI B INJECTION VLV ◦ 2RHS*MOV24C, LPCI C INJECTION VLV

Event Termination Criteria	• All control rods inserted. • Div I and/or Div II EDG powering their respective bus(es). • Reactor water level being controlled above top of active fuel.

Simulator Scenario Handout Reference Packet



Simulator Scenario Handout Reference Packet

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 100%
- 2MSS*PSV129 is inoperable due to failed 'A' and 'B' solenoids
- Suppression Pool water level is being raised with HPCS per N2-OP-33 section H.3.0. Technical Specification 3.5.1 Condition B was entered 1 hour ago for this evolution.

PART III: Remarks/Planned Evolutions:

- Secure raising Suppression Pool water level with HPCS per N2-OP-33 section H.3.0. The procedure is in progress up to step H.3.7.
- Then, lower Reactor power to 95% using Recirculation flow per the provided ReMA. N2-OP-101D is in progress up to step F.3.6.b.3 in preparation for the power reduction.

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-33

HIGH PRESSURE CORE SPRAY SYSTEM

Revision 01900

Safety Related

CONTINUOUS USE

This procedure requires a 10 CFR 50.59

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
019	00	Minor Revision to Incorporate:

PCR-20-03156

Description: Added an enhancement to P&L D.2.4 to provide information in respect to minimizing time CSH pump is run on min flow. Also, updated embedded cautions for updated P&L. The P&L is updated to read “Minimize the amount of time the CSH pump is run at minimum flow condition to prevent pump damage. A total accumulated time that the CSH pump is run on minimum flow should not exceed 60 hours between overhauls. Reference Calculation Disposition No. A10.1 G 050 01A.”

Reason: AR 04324166-62 directed a review to eliminate vague guidance to perform actions to 'maximize', 'minimize', or 'limit' with no specifics on what it means.

- Section A.1, added a new reference.
- Step D.2.4, added additional information on running on minimum flow.
- Step H.3.3 proceeding caution, aligned to be identical with Step D.2.4.
- Step H.4.2 proceeding caution, aligned to be identical with Step D.2.4.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	5
	A.1. Developmental References	5
	A.2. Performance References	6
	A.3. Commitments	6
B.	SYSTEM DESCRIPTION	7
C.	OPERATING REQUIREMENTS	8
D.	PRECAUTIONS AND LIMITATIONS	9
E.	STARTUP	12
	E.1. Fill and Vent [SOP] [C3]	12
	E.2. Startup to Standby [SOP]	22
F.	NORMAL OPERATION	23
	F.1. Standby Condition Status Checks [EOP]	23
G.	SHUTDOWN	24
	G.1. Shutdown To Standby From Injection [SOP] [EOP] [C1]	24
	G.2. Shutdown to Standby From Test Lineup	25
	G.3. Shutdown To Inoperable	27
H.	OFF NORMAL PROCEDURES	28
	H.1. Automatic Initiation [SOP]	28
	H.2. Suppression Pool Fill By Gravity Drain [EOP]	29
	H.3. Suppression Pool Fill Utilizing CSH Pump [EOP]	30
	H.4. Securing Injection Flow Following Initiation	32
	H.5. Placing CSH in a Test Lineup	33
	H.6. Manual Transfer CSH Pump Suction from CST To Suppression Pool	36
	H.7. Manual Transfer of CSH Pump Suction from Suppression Pool To CST [EOP]	37
	H.8. Manual Initiation Using Initiation Pushbutton [EOP]	38
	H.9. Manual Injection to RPV [EOP]	39
	H.10. CSH Operation with WTR LEG PMP 2 Out of Service	40
	H.11. Defeating Injection Valve (2CSH*MOV107) High Vessel Level Interlock [EOP] [SOP]	41
	H.12. Drain CSH Discharge Header to Suppression Pool	42
	H.13. Placing CSH in Standby Following High Drywell Pressure Initiation	45
	H.14. Large Volume Flush of HPCS Injection Line in Mode 4 or 5	46
	H.15. HPCS Injection to Fill Reactor Refueling Cavity in Mode 5	52
	H.16. HPCS Fill and Vent using MWS Supply with System Keep Fill Pump Inoperable	64
	H.17. HPCS Pump Run Following Maintenance	67

TABLE OF CONTENTS (Continued)

SECTION TITLE PAGE

H.18. Maintaining HPCS Available using MWS Supply with 2CSH*P2 Inoperable..... 69

Attachment 1, Making 2CSH*MOV107, PMP 1 INJECTION VLV, Throttleable 71

Attachment 2, 2CEC*PNL601 (H.13.0) 73

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

- A10.1-G-050-01A, HPCS Pump Min. Flow Requirements
- T.S. 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation
- T.S. 3.5.1, ECCS Operating
- T.S. 3.5.2, Reactor Pressure Vessel (RPV) Water Inventory Control
- T.S. 3.6.1.3, Primary Containment Isolation Valves (PCIVs)
- T.S. 3.8.1, AC Sources - Operating
- T.S. 3.8.2, AC Sources - Shutdown
- USAR Section 6.3, Emergency Core Cooling Systems
- PID 33A High Pressure Core Spray
- PID 33B High Pressure Core Spray
- ESK 6CSH01, HPCS Water Leg Pump
- ESK 6CSH02, High Pressure Core Spray MOVs
- ESK 6CSH03, High Pressure Core Spray MOVs
- ESK 6CSH04, High Pressure Core Spray MOVs
- ESK 6CSH05, High Pressure Core Spray MOVs
- ESK 7CSH02, ERF Computer Input Isolation Circuits
- ESK 11IHS18, ERF Computer Output Isolation Circuits
- GE 807E172TY, HPCS Elementary Diagram and Device Test, Sheets 1 7
- GE 807E183TY, HPCS Power Supply System
- GE 731E950AF, HPCS Functional Control Diagrams
- Borg Warner Byron Jackson Pump Division, Manual No. 8020VMT/1F7820, GEK 64171, SWEC PO NMP2 P800A, Access No. 4300008888.
- Gould Pumps Technical Manual, SWEC P.O. NMP2 P222X 5, Access No. 430000719.
- Anchor/Darling Valve Co., Motor Operated Valve Instruction Manual VPF 3238 724 3, SWEC P.O. NMP2 800A, Access No. 430000178.
- GEK 83335 HPCS Operations and Maintenance Instructions
- FSAR Questions and Responses F430.35 ECCS QA Cat I MOV Duty
- SSM N064-022

A.1 (Continued)

- Safety Evaluation 89-029, 90-145
- NMPC File Code 73598 Dated March 19, 1991
- IR 02083078, RHS Suppression Pool Cooling Mode Inops LPCI mode
- IR 02094174, HPCS System Trip Unit Annunciator 601727 Alarmed After 2CSH*P1 Was Started During N2-OSP-CSH-Q@002
- N2-OSP-CSH-CS001, HPCS Cold Shutdown Valves Operability Test
- N2-OSP-CSH-M001, HPCS Discharge Piping Fill and Valve Line up Verification
- N2-OSP-CSH-Q001, High Pressure Core Spray Valve Operability Test
- N2-OSP-CSH-Q@001, HPCS Pump and Valve Operability and System Integrity Test
- N2-OSP-CSH-R001, High Pressure Core Spray System Functional and Response Time Test
- N2-OSP-CSH-R002, High Pressure Core Spray System Valve Position Indication Operability Test
- N2-OSP-CSH-R301, Logic System Functional Test of HPCS Injection Valve Closure on Reactor Vessel High Level 8 Trip
- N2-OSP-CSH-R@001, HPCS Pressure Isolation Valve Leakage Test

A.2. **Performance References**

- N2-VLU-33-LINEUPS
- N2-OP-4, Condensate Storage and Transfer

A.3. **Commitments**

Sequence Number	Commitment Number	Description
1	SOER 86-1	Verification of trips and valve positions after tests, maintenance and operations.
2	IR 02088103	Trip of RCIC Turbine
3	NRC-GL-2008-01 NCTS 504646	NRC Generic Letter 2008-01. Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems.

B. SYSTEM DESCRIPTION

The High Pressure Core Spray System (CSH) is designed to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. The CSH system is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than one-inch nominal diameter. CSH will provide spray cooling heat transfer during larger breaks in which the core is calculated to become uncovered. CSH will also supply makeup water to the reactor vessel in the event of reactor isolation and failure of the Reactor Core Isolation Cooling (ICS) system. The CSH system can also be used to add water to the Suppression Pool. The system is normally powered from off-site power, or from a dedicated Diesel Generator (EDG2), if off-site power is not available.

The CSH system consists of a single, motor-driven CSH pump (CSH*P1), a smaller motor-driven CSH System Pressure Pump (CSH*P2), and associated piping, valves, controls, and instrumentation to support system operation.

Normal suction for the system is from the "B" Condensate Storage Tank (CST), using the Suppression Pool (SP) as an alternate source. The CST suction piping for other systems is arranged so that a volume of 135,000 gallons is dedicated to the CSH system. The suction flowpath will automatically transfer from the CST to the SP upon either a low CST level or a high SP level signal.

Under normal plant conditions, the CSH system is filled and vented, in the "Standby" mode with the System Pressure Pump running to maintain pressure at approximately 75 psig.

To test the operability of the CSH system during plant operation, two test loops are provided to verify proper discharge pressures and flow. One test loop permits flow from/to the Condensate Storage Tank. The second test loop permits flow from/to the Suppression Pool. In the event the CSH pump is operated with the Injection Valve (CSH*MOV107) and all test valves
(*MOV110 -111, -112) close, the Minimum Flow Valve (*MOV105) opens to permit sufficient flow (500 gpm) to the Suppression pool to prevent pump overheating.

The CSH system initiates automatically on either high drywell pressure or low-low reactor vessel level, starting the CSH pump (*P1) and opening its injection valve (*MOV107). The CSH pump discharges into the vessel through a sparger mounted within the core shroud to direct spray over the fuel bundles. The system can also be manually initiated. If the system is in any mode other than standby, all valves will align to the injection mode. If the system is taking suction from the SP, the SP suction valve (*MOV118) will remain open.

When reactor vessel level is restored to the high level alarm point, the Injection Valve closes, and the Minimum Flow Valve opens, diverting flow to the SP. The CSH automatic initiation signal can be reset when the reactor vessel level is restored above 108.8". A high drywell pressure signal will not prevent resetting the initiation signal.

C. OPERATING REQUIREMENTS

C.1. **The following systems must be in operations to support HPCS system:**

- HPCS Diesel Generator N2-OP-100B
- Standby and Emergency AC Distribution N2-OP-72
- HPCS 125 VDC Distribution N2-OP-74B

D. PRECAUTIONS AND LIMITATIONS

D.1. System Operation

D.1.1. If WTR LEG PMP 2 is out of service, CSH will be inoperable but may remain available provided that the requirements of Subsection H.10, CSH Operation with WTR LEG PMP 2 Out of Service, are met.

D.1.2. Maintain CSH system in automatic operation, unless one of the following is confirmed, by at least two independent indications:

- Misoperation in AUTOMATIC mode is confirmed, or
- Adequate core cooling is assured.

D.1.3. Misoperation includes: inappropriate initiation of CSH system or continued operation of CSH system beyond automatic trip setpoints.

NOTE

When CSH system is in MANUAL, it will not restart automatically.

D.1.4. If CSH system is placed in MANUAL mode:

- Make frequent checks of the initiating or controlling parameter.
- When MANUAL mode is no longer required, restore the CSH system to AUTOMATIC or STANDBY Mode. (SER 90-145)

D.1.5. Only the Condensate Storage and Transfer System may be used to fill or flush the CSH System.

D.1.6. Condensate Storage and Transfer System shall never be used to keep CSH system charged due to the potential for over pressurizing the low pressure CNS piping with CSH pump discharge pressure.

D.1.7. If CSH system is discharging into the vessel under any conditions other than an actual loss-of-coolant accident, CSH suction should be aligned to CST.

D.1.8. If CSH system is operating in full flow test with suction and return aligned to CSTs, CST A level will rise. This may result in an overflow of CST A. ith 2CSH*P1 flow at 6350 gpm, CST levels will stabilize when a differential level of approximately 9 feet is established between CST A and CST B.

D.1.9. Due to concerns that CSH will not inject to the vessel within the analyzed time frame with the full flow test valves open. CSH will need to be declared inoperable with the CSH pump running and test return valves open.

D.1.10. Gas accumulation monitoring is required following maintenance or operational activities that have the potential to drain the system (or portions of the system) or that may cause gas accumulation. This will require Ultrasonic Testing with subsequent fill and vent (if required) prior to restoring system to operable. These actions will ensure piping is full of water

D.2. **CSH Pump and Motor Limitations**

D.2.1. Maintain CSH flow less than 7175 gpm in all operating modes.

D.2.2. CSH pump motor start limitations:

- Two starts in succession from ambient temperature after which a 60-minute wait is required prior to subsequent start attempts.
- One start from rated temperature (established after 30 minutes run time), after which a 60-minute wait is required prior to subsequent start attempts.

D.2.3. Maintain CSH Pump Motor stable full load running current less than 378 amps.

D.2.4. Minimize the amount of time the CSH pump is run at minimum flow condition to prevent pump damage. The NMPNS ASME XI Program implements vibration monitoring and performance monitoring, which will detect any significant effect on pump parameters prior to any significant degradation. No additional administrative actions are required for tracking time that the CSH pump is run at minimum flow. (See Calculation No. A10.1-G-050-01A.)

D.2.5. As an operator good practice, the High Pressure Core Spray pump supply breaker should be checked properly recharged following normal opening operation. Refer to N2-ELU-01 Subsection 3.4.

D.3. **CSH Valve Limitations**

D.3.1. The duty cycle of ECCS MOVs is five cycles per hour. If injection is required and MOV duty cycle limitation is exceeded, ICS or Low Pressure ECCS Systems should be used.

D.3.2. Do not open 2CSH*MOV107, Pump 1 Injection Vlv., while testing 2CSH*V108, Injection Check Valve.

D.3.3. 2CSH*MOV107, PMP 1 INJECTION VLV should not be manually opened with the CSH test return valves to the CST or suppression pool open. This would cause pump runout.

D.3.4. Do not use excessive force to achieve leak tight closure of valve 2CSH*HCV133, Test Bypass to Suppression Pool.

D.3.5. If the reactor is shutdown and depressurized, CSH Hi Water Level Seal in interlock will not reset. This is due to the wide range level instruments reading upscale because their calibration conditions are not met.

D.4. **CSH is a single train safety system:**

D.4.1. Unplanned inoperability of the CSH system must be reported to the NRC (10CFR50.72.b.3.v and 50.73.a.2.v) even though the plant Technical Specifications may allow such a condition to exist.

D.4.2. Inoperability of the CSH system where it is removed from service as part of a planned evolution (such as PM's, ST's or corrective maintenance) in accordance with an approved procedure, and in accordance with the plants Technical Specifications, does not require reporting to the NRC. However, if a condition or event is discovered which would have otherwise rendered CSH inoperable, then D.4.1 applies.

D.5. Prior to venting or draining evolutions, notify Radiation Protection.

D.6. If the CSH system is operated with suction from the suppression pool, it must be shut down prior to returning the suction to the CST's. Swapping the suction with the system in operation could cause a hydraulic transient that is outside the system design basis. **[C2]**

D.7. The CSH System is a designated closed system, which means that the system piping is part of the primary containment. The closed system boundary includes the main process lines from the suppression pool on the suction side of associated pumps through the containment penetration on the discharge side, out to and including the first normally closed branch boundary valve (i.e., drain/vent valves, interface boundary valves, relief valves, etc.). The closed system is credited as one of the two boundaries required for containment isolation under Appendix J and TS 3.6.1.3. Excessive leakage above the minimum suppression pool level (197’-8”) could impact the closed system primary containment boundary function. System leaks identified above elevation 197’-8” greater than 10cc/min (200 dpm) require entry into TS 3.6.1.3 and securing the associated primary containment penetration. Leaks identified less than 10cc/min need to be documented immediately on a work document and corrected on a priority basis (i.e. Pri-3 Necessary or greater).

Initials

H.3. **Suppression Pool Fill Utilizing CSH Pump [EOP]**

H.3.1 **VERIFY** Subsection F.1, Standby Condition Status Checks is complete. OPE

H.3.2. **VERIFY** SM has declared CSH System inoperable. OPE

NOTE

Starting 2CSH*P1 with suction from the CST can result in Gross Failure alarms on trip units E22-N652-HPCS PMP SUCT LO and E22-N653-HPCS PMP SUCT HI.

CAUTION

- Allowing CSH pump to operate in runout flow condition of greater than 7175 gpm may cause pump damage.
- Minimize the amount of time the CSH pump is run at minimum flow condition to prevent pump damage. The NMPNS ASME XI Program implements vibration monitoring and performance monitoring, which will detect any significant effect on pump parameters prior to any significant degradation. No additional administrative actions are required for tracking time that the CSH pump is run at minimum flow. (See Calculation No. A10.1-G-050-01A.)

H.3.3. **START** CSH*P1, HPCS PUMP 1. OPE

H.3.4. **VERIFY** CSH*MOV105, MINIMUM FLOW BYPASS VLV, is OPEN. OPE

H.3.5 **MONITOR** CST **AND** Suppression Pool Levels. OPE

H.3.6 **RAISE** rate of transferring water from CST to Suppression Pool as follows:

a. **THROTTLE OPEN** CSH*MOV111, TEST RETURN TO SUPPRESSION POOL. OPE

b. **WHEN** CSH System Flow is greater than 634 gpm,
THEN VERIFY CSH*MOV105, MINIMUM FLOW BYPASS VLV, is CLOSED. OPE

H.3.7. **WHEN** desired Suppression Pool Level is reached,
THEN CLOSE CSH*MOV111, TEST RETURN TO SUPPRESSION POOL. _____

H.3.8. **WHEN** HPCS SYSTEM FLOW is less than 634 GPM,
THEN VERIFY CSH*MOV105, MINIMUM FLOW BYPASS VLV, is OPEN. _____

H.3.9. **STOP** CSH*P1, HPCS PUMP 1. _____

Initials

H.3.10. **VERIFY** CSH*MOV105, MINIMUM FLOW BYPASS VLV, is CLOSED. _____

H.3.11. **NOTIFY** SM that 2CSH*MOV111 is CLOSED **AND** operability concern **PER** IR 02083078 **no** longer exists. _____

H.3.12. **RESTORE** CST Level, as required **PER** N2-OP-4. _____

Constellation

Nine Mile Point Nuclear Station Unit 2
OPERATING PROCEDURE

N2-OP-101D

POWER CHANGES

Revision 03700

Safety Related

CONTINUOUS USE

10 CFR 50.59 REVIEW REQUIRED

Approval Authority: Director-Operations

SUMMARY OF ALTERATIONS

Minor revision to incorporate the following:

PCR-21-00837:

- **Description:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more.
- **Reason:** Update notification to Public Service Commission (PCS) to perform for all (planned or unplanned) load reductions of 15% or more. It makes the procedure consistent with N1-OP-43B. Also, the PSC desires to be notified for all load reductions.
- Step F.3.2, changed bullets to numbers, added notes, and made improvements for notifications.
- Step G.1.1, changed bullets to numbers, added notes, and made improvements for notifications.

PCR-21-00890:

- **Description:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR.
- **Reason:** Add conditions of MSROOS, PLUOOS, and TCVFASOOS. Deleted “Allowable Modes of Operation with Inoperable Equipment Table” and added 2 P&Ls to align with the U2 COLR. These changes were driven by ECP-20-000231 for the GNF3 fuel modification in RFO18.
- Step D.1.10.b, added MSROOS, PLUOOS, and TCVFASOOS.
- Deleted Allowable Modes of Operation with Inoperable Equipment Table and it’s associated notes.
- Added step D.1.10.c.
- Added P&L D.1.11 added per Reactor Engineering comment.

TABLE OF CONTENTS

SECTION	TITLE	PAGE
A.	REFERENCES AND COMMITMENTS	4
B.	SYSTEM DESCRIPTION	6
C.	OPERATING REQUIREMENTS	6
D.	PRECAUTIONS AND LIMITATIONS	7
E.	STARTUP	15
E.1.	Power Increase From 45% to 100%	15
F.	NORMAL OPERATIONS	23
F.1.	Rated Power Operation	23
F.2.	Routine Reactor Power Maintenance [C12]	25
F.3.	Power Maneuvers Between 50-100%	27
F.4.	Power Maneuvering for Parameter Margin Control	38
F.5.	Trimming of DEHC Pressure Set	43
G.	SHUTDOWN	44
H.	OFF-NORMAL OPERATIONS	51

A. REFERENCES AND COMMITMENTS

A.1. Developmental References

A.1.1. Technical Specifications

Due to the fact that most, if **not** all, of the technical specifications apply to plant power operation, they are **not** listed individually in this procedure.

A.1.2. Licensee Documentation

- Technical Specification Amendment 123 (ARTS/MELLLA+)
- Updated Safety Analysis Report, USAR
- Consult individual operating procedures as necessary.

A.1.3. Technical Information

- Flow Diagrams

  Consult individual operating procedures as necessary.
- Electrical Diagrams

  Consult individual operating procedures as necessary.
- Vendor Manuals

  Consult individual operating procedures as necessary.
- System Instruction Manuals

  Consult individual operating procedures as necessary.

A.1.4. Supplemental References

- NRC Bulletin 88-07 Supplement 1, Power Oscillations in BWR.
- NMPC 865, GE Project Letter.
- Safety Evaluation 91-049, Revision 1, Extended Load Line Limit Analysis.
- Power-Flow Operating Map (EM-950A) Two RCS Loops Operation.
- Power-Flow Operating Map (EM-950B) One RCS Loop Operation.
- EDC 2M00587 Revision to the Power-Flow Operating Map.
- NMPC File Code ESB2-M95-0193, Condensate/Feedwater System Operation Post Update.
- NMPC File Code ESB2-PM96203, Turbine Pressure Control.
- NER-2E-037, APRM/RBM/Technical Specification – Maximum Extended Load Line Limit Analysis (ARTS/MELLLA+) Project.
- SOER 07-1, Reactivity Management. **[C13]**

A.1.4 (Continued)

- RIS 07-21, Revision 1, Adherence To License Power Limit.
- Amendment 140 to the Facility Operating License, Extended Power Uprate.
- IR 02002103, Unit 2 Required Down Power Due to Offgas Flow Excursions.
- ECP-12-000448, NMP2-Implementation of MELLLA+.
- IR 04296001, Received 2RCS*P1B lower mtr bearing oil level low alarm.

A.2. **Performance References**

- LS-CE-1255, Northeast Site Specific - CE (New York)
- N2-OP-101A, Plant Startup
- N2-OP-101C, Plant Shutdown
- N2-OP-10A, Circulating Water System
- N2-OP-27, Generator H_2 and CO_2 System
- N2-OP-29, Reactor Recirculation System
- N2-OP-3, Condensate and Feedwater System
- N2-OP-30, Control Rod Drive
- N2-OP-42, Offgas System
- N2-OP-5, Condensate Demineralizer System
- N2-OP-8, Feedwater Heaters and Extraction Steam
- N2-OP-9, Condensate Air Removal
- N2-OP-91A, Process Computer
- N2-REP-11, Independent Methods of Determining Core Thermal Power
- N2-RESP-12, APRM/LPRM Noise Level Determination During Low Flow Operation
- N2-RESP-4, LPRM Calibration
- N2-SOP-101C, Reactor Scram
- N2-SOP-101D, Rapid Power Reduction
- N2-SOP-29, Sudden Reduction in Core Flow
- N2-TTP-VIB-V001, Turbine Vibration Readings
- OP-AA-106-101, Significant Event Reporting
- OP-AA-106-101, Significant Event Reporting
- OP-AB-300-1003, BWR Reactivity Maneuver Guidance

A.3. **Definitions**

None

A.4. **Commitments**

Sequence Number	Commitment Number	Description
1	SOER 90-2	Nuclear Fuel Defects
2	SOER 90-3	Nuclear Instrumentation Miscalibration
3	DER 2-92-Q-0050	Control Rod Drive Problems Experienced During 12/13 and 12/16/91 Reactor Startup
4	LER 92-17	Rx Scram Due to Loss of Feedwater Pump
5	LER 95-08	Rx Scram Due to High Turbine Vibration
6	DER 2-95-1737	OE Clarification of Basis for Tech. Spec. Neutron Monitoring Flow Referenced Set Point, Recirculation Drive Flow/Core Flow Correlation
7	DER 2-98-3097	Process Computer Failure
8	DER 2-98-3593	Hot Reheat Steam Temperatures Exceeded 50 °F
9	DER 2-99-0421	OE 9387 differences in definition of TBV, OOS between utilities understanding of operability and GE underlying safety assumptions
10	DER-2-99-3922	OE-SEN 208, Reactor Power Ascension with An Unrecognized Loss of Feedwater Heating
11	DER 2-2000-3818	Potential Recirculation System Jet Pump Cavitation
12	CR-2008-000162	Reactor Power Indications Exceeded License Limit
13	CR 2007-004914	INPO SOER 07-1, Reactivity Management
14	CR 2009-002350	Reactor Power spiked above 1850 MWth during Rod Line Adjustment

B. SYSTEM DESCRIPTION

As all plant systems are affected by the plant power operation, individual system descriptions will **not** be discussed in this procedure. See the individual operating procedures for descriptions of systems that are of interest.

C. OPERATING REQUIREMENTS

All systems and their auxiliaries must be in operation, or in a standby readiness per applicable operating procedure.

D. PRECAUTIONS AND LIMITATIONS

D.1.1. Power can be raised by control rod withdrawal or raising reactor recirculation (recirc) flow. Power changes shall **not** be made by simultaneously using reactor recirc flow and control rod withdrawal.

D.1.2. Control rod movement shall be per OP-AB-300-1003, BWR Reactivity Maneuver Guidance.

D.1.3. During power ascension, for each control rod sequence, fuel pre-conditioning rates shall be maintained and per Reactor Engineering instruction. **[C1]**

D.1.4. Recirculation Pump upshift or pump start shall be performed between the 47% Rodline and the 58% Rodline.

D.1.5. Reactor Engineering will provide and update Rodline information during orderly plant startup and shutdown. If a Reactor Engineer is **not** on site, the Unit Supervisor (US) or the STA will verify the Rodline and update the posting at Panel 2CEC*PNL603.

D.1.6. The Main Turbine should **not** be continuously operated with condenser vacuum less than the low vacuum alarm point (23.8 in. Hg). Turbine load should be reduced under deteriorated vacuum conditions in an attempt to restore vacuum. The Turbine should be manually tripped whenever Turbine load is less than 30% (410 MWe) and annunciator 851359 is in alarm with vacuum less than 24.6 in. Hg.

D.1.7. Under normal conditions hotwell water temperature should be maintained less than or equal to 140 °F or as necessary to ensure CY-AB-120-100, Reactor Water Chemistry, Action 1 limits are **not** exceeded.

During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:

- Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
- RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).

D.1.8. Reactor recirculation flow control shall be maintained in LOOP MANUAL.

D.1.9. Technical Specification Requirements During Power Changes

a. When raising Reactor power, N2-RESP-001, Power Distribution Limits Verification, shall be performed once within 12 hours after exceeding 23% Core Thermal Power (CTP) and every 24 hours thereafter.

b. When Reactor power is greater than or equal to 26% Core Thermal Power (CTP) with any recirculation pump in fast speed, and one or more functions with End of Cycle Recirculation Pump Trip (EOC-RPT) trip capability **not** maintained, then action is required within 2 hours to comply with Tech. Spec. 3.3.4.1. **[C9]**

c. When Reactor power is greater than or equal to 23% Core Thermal Power (CTP), and the Main Turbine Bypass System is found to be INOPERABLE, then action is required within 2 hours to comply with Tech. Spec. 3.7.5. **[C9]**

D.1.10. Extended operating domain and equipment out-of-service analyses have been performed for operations within extended operational boundaries and with various plant equipment out-of-service:

a. Extended Operating Domains

1) For Two Recirculation Loop Operation

Operation is allowed above the Maximum Extended Load Line Limit Analysis (MELLLA) Line, provided 100% rated core thermal power is **not** exceeded.

The region of the Power-Flow Map above the MELLLA Line is known as the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Region. Intentional operation beyond the MELLLA+ Boundary is **not** permitted.

The region of the Power-Flow Map between 100% core flow and 105% core flow is known as the Increased Core Flow (ICF) Region.

2) For Single Recirculation Loop Operation

Single Recirculation Loop Operation (SLO) is restricted by two parameters, namely recirculation flow and rod line. The maximum allowable SLO recirculation drive flow is 41,800 gpm due to recirculation piping vibration limitations and the maximum SLO rod line is 89% (Extended Load Line Limit Analysis (ELLLA) Boundary). These restrictions are also described in the SLO Loop Power-Flow Maps (EM-950B-001 and EM-950B-002). Where these two parameters intersect on the Power-Flow Maps restricts SLO maximum power. Intentional Operation beyond the ELLLA Boundary is **not** permitted.

3) Certain plant transients may result in an operating condition beyond the MELLLA+ Boundary. These transients are analyzed - provided the pre-event conditions were within the normal MELLLA+ Region and actions are taken to exit the region above the MELLLA+ Boundary as part of the transient response procedure.

D.1.10 (Continued)

b. Equipment Out-of-Service (EOOS) Analyses

NMP2 has been analyzed for operation in various plant conditions and with various Equipment Out of Service (EOOS). The following conditions may affect the thermal limits as analyzed in the current COLR:

1) Single Recirculation Loop Operation (SLO)
– Per Tech Spec 3.4.1.

2) Safety Relief Valve Out-of-Service (SRVOOS)
– Per Tech Spec 3.4.4.

3) Automatic Depressurization System Out of Service (ADSOOS)
– Per Tech Spec 3.5.1.

4) MSIV Out-of-Service (MSIVOOS)
– Per Tech Spec 3.6.1.3. Refer to N2-OP-1, Main Steam System

5) Turbine Control Valve Out of Service / Turbine Stop Valve Out of Service (TCV/TSVOOS)
– Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

6) Turbine Bypass Valve Out of Service (TBVOOS)
– Per Tech Spec 3.7.5. Refer to N2-OP-23A, Hydraulic Isolation and Restoration of Turbine Stop or Turbine Control or Combined Intercept Valves or Bypass Valves.

7) Recirculation Pump Trip Out of Service, also known as EOC-RPT (RPTOOS)
– Per Tech Spec 3.3.4.1.

8) Pressure Regulator Out of Service (PROOS)
– Per Tech Spec 3.2.2.

9) Moisture Separator Reheaters Out of Service (MSROOS)
– Per the COLR.

10) Power Load Unbalance Out of Service (PLUOOS)
– Per the COLR.

11) Turbine Control Valve Fast Acting Solenoids (TCVFASOOS)
– Per the COLR.

D.1.10 (Continued)

c. GNF3 fuel introduces new equipment out service analyses and combinations of out of service equipment. When out of service equipment is identified, utilize Table 4-2 of the COLR to determine the appropriate penalty to apply for the specific condition that exists. Reactor Engineering assistance may be required to determine the correct penalty to apply for equipment out of service.

d. See the Cycle Specific Core Operating Limits Report for applicable thermal limit impacts for various EOOS conditions [FCMS Search = COLR%].

e. Reference the following to identify the allowable combination of conditions:

- 003N2077-R0, Nine Mile Point 2 GNF NFI T0900 Coincident Equipment out-of-service (EOOS) Report [FCMS Search = 003N2077]
- 006N0918-R0, TSD NF-B756 Nine Mile Point Unit 2 Cycle 18 RPTOOS + TBVOOS Thermal Limits [FCMS Search = 006N0918%]

f. **IF** a specific combination of conditions is **not** identified as allowable, **THEN** it is **not** supported **AND** an orderly plant shutdown to less than 23% power is appropriate.

D.1.11. Operation with Turbine Bypass Valves out of service (TBVOOS) coincident with Turbine Control Valve/Turbine Stop Valve out of service (TCV/TSVOOS) is **not** an allowable condition. It is not supported by the licensing analysis per the COLR.

D.1.12. When Reactor power is greater than 23%, Reactor Engineering notification is required before raising Reactor power with recirculation flow or control rod withdrawal except if power is being raised per Reactor Operating Instructions, Section F.2, Power Maintenance or Section F.4, Power Maneuvering for Parameter Margin Control.

D.1.13. All reactivity changes shall be directly supervised by a Senior Reactor Operator. In order to provide this oversight, the SRO shall be stationed at the Controls Area of the Control Room.

D.1.14. Because of many possible plant conditions, deviation from suggested sequence of operations is permitted as directed by the Shift Manager (SM).

D.1.15. Because the Core Thermal Power Application uses "smooth" feedwater flow inputs, it should **not** be used for Rodline determination during or immediately after power transients. Use APRM power and instantaneous core flow values to compute/plot Rodline from power/flow operating map (Ref. A.1.4, EM-950A or EM-950B Power Flow Operating Map).

D.1.16. If Turbine bearing vibration rises significantly (e.g. to a value of greater than or equal to 5 mils or greater than or equal to 2 mil rise) as a result of Turbine load changes, power should be held steady at the nearest practicable level after the vibration rise. **[C5]**

Once the vibration lowers to an acceptable stable value, Turbine load should be held steady for an additional hour before continuing with load changes. This will allow the Turbine rotor(s) to straighten from any rub induced bowing.

D.1.17. When CTP is greater than or equal to 23% and any APRM indicates outside plus or minus 2% of CTP, an APRM gain adjustment shall be performed prior to additional changes in reactor power.

D.1.18. When power is greater than 90% and prior to any power increase, a Core Thermal Power Report/OD-3 from R*TIME should be demanded to ensure there are **no** unplanned failed sensors inputting into the CTP calculation. Any failed sensor(s) will show on the report under the heading “Failed Sensors List.” (N2-OP-91A has a list of sensors.) **[C6]**

D.1.19. The process computer can fail in ways that are **not** obvious to operators. Operators should **not** add reactivity when the core thermal power calculation is **not** accurate. Operators need to know power level with certainty prior to making adjustments. **[C6] [C7]**

D.1.20. When Reactor Thermal Power is greater than 70% with charcoal adsorbers in service, and Offgas System flow is greater than 25 SCFM the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B sample consistent with the effluent path and maintain proper margin to the trip setpoints.

D.1.21. The Charcoal Adsorber must be in service prior to placing the Offgas Radiation Monitor Delay Pipe in service.

D.1.22. MWth oscillation during steady state operation is a known situation for BWRs using recirculation risers and jet pumps. During steady state operations, if MWth oscillation is experienced, greater than normal expected values, Reactor Engineering should be contacted to adjust rod line or reactor recirculation flow.

D.1.23. The averaging algorithm associated with core thermal power computer points NSSQB001A-01M and NSSQB001A-10M may **not** respond accurately to certain variations in real core thermal power such as oscillations brought about by rapid cycling of recirculation flow. The following computer alarms are provided to ensure that design margins are maintained. Accident analyses provide a 2% margin to account for uncertainties associated with the determination of core thermal power. If any of these alarms are actuated immediate operator action is required to ensure that thermal power is trending below 3988 MWth.

- NSSQB001A_01S, CORE THERMAL POWER – 1-SEC INSTANT Setpoint = 4028 MWt
- NSSQB001A-01M, CORE THERMAL POWER – 1 MIN SMTHD CT POWER Setpoint = 4008 MWt
- NSSQB001A-10M, CORE THERMAL POWER – 10 MIN SMTHD CT POWER Setpoint = 3998 MWt

D.1.24. The R*TIME monitoring above 2CEC*PNL603 is set up to display whole numbers only; decimals are truncated. Caution must be used when monitoring plant parameters using these displays.

D.1.25. Maintain power and flow in the operating domain of the applicable single and two loop power-flow operating maps. Determine expected operating location prior to making reactivity changes. **[C11]**

D.1.26. Primary Containment Oxygen shall be controlled per the requirements of Technical Specification 3.6.3.2.

D.1.27. A CRD Deficiency Log entry should be made at any time during the performance of this procedure when a control rod drive mechanism does **not** perform as expected.

D.1.28. During noble Metals Injection, the plant prerequisite is for core flow greater than or equal to 85% flow and reactor power greater than or equal to 70%. Notify Chemistry Department if outside of these parameters during Noble Metals Injection.

D.1.29. When making core reactivity changes, operators shall monitor redundant nuclear instrumentation as well as other plant indications. Instrumentation to be used but **not** limited to includes SRM count meters, SRM period meters, IRM recorders, APRM recorders/monitors, LPRM meters/monitors, Instantaneous and "Smoothed" Power computer points, Generic Output, Recirc Flow indications, Turbine Bypass and Control Valve positions and Reactor Pressure.

D.1.30. For planned power changes 10% or greater, S.O.S. oversight is required.

D.1.31. Fuel precondition is based on raising fuel bundle nodal power at a rate of less than or equal to 0.35 kW/ft per hour. This translates into raising power at a rate of approximately 3%/hour. This can be accomplished as follows:

a. Raise reactor power by 3% over a one-hour period.

b. Raise reactor power by 3% at any time during the one hour-period, but remain at that power level for the remainder of the hour.

D.1.32. If the 3% rate is exceeded over the first hour, then this can be adjusted for in the next hour. Conversely, if the power increase falls short of the 3% rate over the first hour, this power can be adjusted for over the next hour. However, when adjusting for a shortfall in the first hour, it is **not** permitted to exceed a rate of 3.5% per hour in the second hour.

D.1.33. Operating the reactor in compliance with the Licensed Power Limit is demonstrated by the following process;

a. **No** actions are allowed that would intentionally raise core thermal power above 3988 MWth for any period of time. Small, short-term fluctuations in power that are **not** under the direct control of a licensed Reactor Operator (e.g. fluctuation caused by bi-stable flow) are **not** considered intentional. Due to the change in the reactor bi-stable flow characteristics following N2R16 outage, positive reactivity addition for Power Maintenance shall **not** be made unless instantaneous, one minute and ten-minute thermal power is less than or equal to 3986 MWth and trending lower.

b. Closely monitor thermal power during steady state power operation with the goal of maintaining the rolling 1 hour thermal power average at or below 3987.5 MWth. If the core thermal power average for a rolling 1 hour period is found to exceed 3987.5 MWth, take action to ensure that thermal power is reduced to less than or equal to 3987.5 MWth. Power Maintenance shall be performed before exceeding 3987.5 MWth on the rolling two hour thermal power average.

c. The core thermal power average for a shift is **not** to exceed 3988 MWth. For the purpose of this guidance, a shift is defined as 0600 to 1800 or 1800 to 0600.

d. During End of Cycle Operations, Rx power shall **not** be raised above the Rx power which can be achieved with steady state equilibrium Xenon concentrations in the core (all control rods out, all Recirc Pumps in service, raise Core Flow), as determined by Reactor Engineering.

D.1.34. When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition. When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rate at 3966 MWth and above will be done with recirculation flow. **[C14]**

- The above precaution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, “Reverse Power Effect”, for power maintenance. There should be a minimum of three minute wait prior to any further adjustments with Recirculation Flow.

D.1.35. “Steady State” for the purpose of maintaining licensed power level is defined as:

- **No** control rod motion
- **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
- Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
- Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).

It is **not** intended that “steady state” include equilibrium Xenon or Samarium.

D.1.36. The Feedwater minimum flow valves, 2FWR-FV2ABC, are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:

- Single Pump Operation - 23% to 27% Power
- Two Pump Operation with balanced flows - 46% to 53% Power

D.1.37. Failure to have 3 Condensate Booster Pumps (if **not** inhibited) and 2 Feedwater Pumps in service prior to Steam flow exceeding 11.5 Mlbm per hour (approximately 72% power) will result in a Recirc Flow Control Valve Runback.

D.1.38. With MSRs out of service, special restrictions apply to continuous operation with a heater drain pump out of service. The plant can operate with one heater drain pump out of service provided continuous operation is limited to less than or equal to 69% of rated thermal power. With more than one heater drain pump out of service, reactor power is limited to less than or equal to 56%.

D.1.39. Excessive cycling of the feedwater heater dump valve could damage the associated condenser penetration resulting in an increase in Offgas flow. Immediate actions should be taken to stop/avoid cycling of the heater dump valves.

D.1.40. The following parameters are required to be monitored during each reactivity manipulation:

- Redundant Nuclear Instrumentation
- Core Flow
- Reactor Pressure
- Reactor Temperature

D.1.41. In October 2019, station personnel performed Drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.

D.1.42. During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power), vibration of EHC System Piping and components should be closely monitored. In March 2020, NMP Unit 2 required a manual SCRAM in response to a large EHC leak from to a weld failure caused by piping vibrations after an extended period of operation at this power level. More frequent monitoring of EHC piping and system components should be performed when operating in this “high EHC flow” region. If excessive vibration of EHC piping and system components occur, an evaluation should be made to raise or lower Reactor Power to avoid operation in this region.

Initials

F.3. **Power Maneuvers Between 50-100%**

NOTE

- This section is for performing plant maneuvers between 50% to100%, this section assumes that the plant startup has been completed and that power reduction / ascension is **not** from an outage.
- The steps in this section are a guide for plant maneuvers, if it is determined a plant shutdown is required then entry into the shutdown section of this procedure should occur.
- The steps in this section may be performed out of sequence as determined by the Shift Manager.
- In October 2019, station personnel performed drywell entry at power to resolve a CCP leak, subsequent to the entry a RCS pump lower motor bearing oil level low alarm was received that could have been resolved by observation of the sight glasses.
- During periods of extended operation at power levels in which all 4 Turbine Control Valves are modulating (approximately 84%- 94% Reactor Power) vibration of EHC System Piping and components should be closely monitored. Unit 2 required a manual SCRAM in response to a large EHC leak in March 2020, after a weld failure caused by piping vibrations after an extended period of operation at this power level.
- The Feedwater minimum flow valves, 2FWR-2FVA(B,C), are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:
  - Single Pump Operation 23% to 27% Power
  - Two Pump Operation with balanced flows - 46% to 53% Power

N/A F.3.1 **IF** load reduction is in support of Drywell (DW) entry at power,
THEN NOTIFY Outage Control Center that visual verification of adequate oil level in both Reactor Recirculation Pump (2RCS-M1A and 2RCS-M1B) Motor Bearings (Upper and Lower) must be completed
PRIOR TO Drywell exit,
OTHERWISE MARK this step N/A.

N/A

Initials

F.3.2 **PERFORM** the following power reduction preparatory steps, **AND** Steps **not** required may be marked N/A:

NOTE

Steps F.3.2.a thru F.3.2.l may be performed in any order.

N/A a. **IF** prior to a planned shutdown **OR** significant load drop, **THEN INSTRUCT** Plant Performance Group to install vibration data gathering instrumentation at 2CEC-PNL841 per N2-TTP-VIB-V001, **UNLESS** waived by System Engineer..... ☒ N/A

NOTE

Per OP-AA-102-101, the following expectations/requirements of timeliness for notifications to Generation Dispatch:

- 2-hour notice is desirable
- 30-minute notice is required
- Immediately before is desirable

b. **VERIFY** the following have been notified of planned power changes:

- Generation Dispatch..... ☑
- Power Control..... ☑
- NDO ..... ☑

NOTE

An unplanned load reduction is one that was **not** previously scheduled.

N/A c. **IF** conducting an unplanned load reduction of 15% **OR** greater, **THEN NOTIFY** Public Service Commission Notification per the following:

1) OP-AA-106-101, Significant Event Reporting ..... ☒ N/A

2) LS-CE-1255, Northeast Site Specific - CE (New York) ..... ☒ N/A

Initials

F.3.2 (Continued)

N/A d. **IF** containment will be de-inerted,
THEN VERIFY GSN Electric Vaporizers are available ..... N/A

N/A e. **IF** Condensate Demineralizers will be removed,
THEN VERIFY removal order has been developed..... N/A

N/A f. **IF** Clean Steam Reboilers will be swapped to the Aux Boilers,
THEN PREPARE Aux Boilers to supply Clean Steam Reboilers..... N/A

N/A g. **IF** Intermediate Heat Exchangers will be removed from service,
VERIFY Auxiliary Heat Exchangers are in service. ..... N/A

N/A h. **IF** entering Mode 3,
THEN VERIFY all Mode 3 required surveillances are completed
OR ready for completion ..... N/A

N/A i. **IF** desired,
THEN VERIFY N2-OP-25, H.6.0, Preventing Automatic Swapover of
Reboiler Steam Supply, is complete..... N/A

N/A j. **IF** planning to reject WCS to the Main Condenser,
THEN VERIFY 2NHS-MCC2008-2E, 2WCS-MOV107, RWCU
DISCH TO MAIN COND is ON..... N/A

N/A k. **IF** Cooling Tower will be changed to full bypass,
THEN VERIFY Basin Level is adjusted for Cooling Tower Mode
Change (Bypass) ..... N/A

N/A l. **IF** performing Scram time testing per N2-OSP-RMC-@001,
THEN prior to commencing power reduction,
VERIFY communication equipment is staged
AND prerequisites are performed. ..... N/A

N/A

Initials

F.3.3 Prior to planned power change,
EVALUATE, ASSIGN AND RECORD below any system critical parameter(s) that may be in an abnormal control lineup which may be adversely affected by the load drop (check AIU book; for example, Gland Sealing Steam):

System Parameter	Operator Responsible (i.e., OATC)
None	N/A

OPE

F.3.4 **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation ☑
- Core Flow ☑
- Reactor Pressure ☑
- Reactor Temperature ☑

OPE

F.3.5 **DURING** power change, some of the following systems will require periodic adjustments depending on the significance of the drop.
DIRECT OATC or other available RO(s) to
PERIODICALLY MONITOR the following system parameters
AND NOTIFY the US for adjustment as necessary:

- Reactor Water Cleanup Filter Flows ☑
- Cooling Tower Basin Level/Temperature ☑
- Main Generator Reactive Load (MVARs) ☑
- Turbine Vibrations ☑
- Main Condenser Vacuum ☑
- Gland Sealing Steam Pressure ☑
- Gland Seal Exhaust Pressure ☑
- Clean Steam Reboiler Level Control ☑
- Condensate Demineralizer System/Demin Flow ☑
- Feedwater Level Control Response ☑
- MSR Drain Tank Oscillations ☑
- Containment Parameters ☑

OPE

Initials

F.3.6 Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ..... ☑
- **NOTIFY** NDO if required per OP-AA-102-101. ..... ☑
- **NOTIFY** Chemistry to monitor Offgas chemistry per NF-AB-400-1000, BWR Fuel Integrity Monitoring ..... ☑
- **REVIEW** applicable Precautions and Limitations. ..... ☑
- **OBTAIN** authorization from Shift Manager to reduce power. .. ☑
- **REVIEW** RMI for planned power reduction. ..... ☑
- **COMPLETE** applicable reactivity brief. ..... ☑ OPE

b. **PERFORM** the following actions as needed **DURING** power reduction:

1) At 2CEC*PNL851, **PERIODICALLY MONITOR** Offgas System Flow Rate using 2OFG-FI3A/B. ..... ☑

N/A

2) **IF** automatic make up air to Offgas is **not** available, **THEN WHEN** Offgas flow is less than or equal to 6 SCFM, **PERFORM** the following, **OTHERWISE MARK** these steps N/A:

   a) **REMOVE** pipe cap from 2ARC-V47, ARC SUCTION HDR VENT..... ☑ N/A

   b) **THROTTLE OPEN** 2ARC-V47 to establish approximately 7 to 10 SCFM on each indicator (2OFG-FI3A/3B)..... ☑ N/A

3) **LOWER** power as directed by approved ReMA. ..... ☐

4) **MONITOR** core flow / reactor power on Power to Flow Map... ☐

5) **IF** core flow / power reduction results in entry into the exit region, **THEN ENTER** N2-SOP-29, **OTHERWISE MARK** this step N/A. ..... ☐

6) **MONITOR** rodline as needed based on core conditions. ..... ☐

7) **MONITOR** Feedwater Pump Seal temperatures **AND ADJUST** seal flows per N2-OP-3. ..... ☐

F.3.6 (Continued)

N/A c. **PRIOR TO** going below 80% power perform the following:

1) **MAINTAIN** HVH steam supply on Extraction Steam as follows:

NOTE

If plant trip or power reduction below 50% occurs, building heating could be lost because this step prevents the auto swap from Extraction to Aux Steam.

a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in CLOSE. ..... ☐

2) **MAINTAIN** Clean Steam Reboiler steam supply on Extraction Steam at P851 as follows:

NOTE

If plant trip or power reduction below 50% occurs, the Extraction Steam supply pressure could be insufficient to support clean steam reboiler operations and result in an auto swap to the emergency sealing steam supply.

a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in OPEN. ..... ☐

b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in CLOSE. ..... ☐ N/A

N/A d. **WHEN** Steam flow goes below 11.5 Mlbm/hr (approximately 72% power),
CONFIRM RECIRC RUNBACK INHIBITED lit on 2CEC*PNL602. N/A

Initials

F.3.6 (Continued)

N/A e. **CONTINUE** power reduction
AND PERFORM the following as required:

- **REMOVE** Condensate Demineralizers from service as required
  AND at recommendations of Chemistry per N2-OP-5, Condensate Demineralizer System. ..... ☐
- At approximately 65%, **PLACE** Heater Drain pumps in recirculation per N2-OP-8, G.1.0. ..... ☐ N/A

N/A f. **DURING** power reduction below 65%,
PERIODICALLY MONITOR reheat steam pressures
AND temperatures to ensure LP Turbine nozzle side-to-side inlet temperature differential is maintained less than or equal to 50 °F. N/A

Initials

N/A F.3.7 Power Ascension

a. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation ..... ☐
- Core Flow ..... ☐
- Reactor Pressure ..... ☐
- Reactor Temperature ..... ☐

N/A

b. **PERFORM** the following actions as needed during the power ascension:

1) **MONITOR** core flow / reactor power on power to flow map. ... ☐

2) **IF** core flow / power ascension results in entry into the exit region,
THEN ENTER N2-SOP-29,
OTHERWISE MARK this step N/A. ..... ☐

3) **MONITOR** rodline as needed based on core conditions. ..... ☐

4) **MONITOR** Feedwater Pump Seal temperatures
AND ADJUST seal flows per N2-OP-3. ..... ☐

5) **WHEN** Reactor power is greater than 65%,
THEN VERIFY the following Reheat Steam Load Control Valves are full open:

- 2MSS-PV28A ..... ☐
- 2MSS-PV28B ..... ☐
- 2MSS-PV29A ..... ☐
- 2MSS-PV29B ..... ☐

6) **WHEN** Steam flow exceeds 11.5 Mlbm/hr (approximately 72% power), **CONFIRM** RECIRC RUNBACK INHIBITED light on 2CEC*PNL602 extinguishes ..... ☐

NOTE

Eight condensate demineralizers are required for 100% power with nine condensate demineralizers preferred for additional operational margin.

7) **PLACE** Condensate Demineralizers in service as required per N2-OP-5
AND Chemistry guidance for restoring demins to service. ..... ☐

N/A

Initials

N/A

F.3.7 (Continued)

c. **WHEN** reactor power is greater than 80%,
THEN PERFORM the following:

1) **RESTORE** HVH steam supply automatic swap to Aux Steam as follows:

a) **VERIFY** control switch for 2ESS-STV105, BLDG INTMD HT EXCH ST ISOL, is in AUTO. ..... ☐

b) **VERIFY** control switch for 2ASS-STV143, BLDG INTMD HT EXCH MN ST ISOL, is in AUTO. ..... ☐

2) **RESTORE** Clean Steam Reboiler automatic swap to Aux Steam at P851 as follows:

a) **VERIFY** control switch for 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV, is in AUTO. ..... ☐

b) **VERIFY** control switch for 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV, is in AUTO. ..... ☐

N/A

N/A F.3.7 (Continued)

d. **WHEN** Power is greater than 90%,
PERFORM the following:

1) **WHEN** Power is greater than 90%
AND PRIOR TO any further power increase,
THEN DEMAND an OD-3 to ensure there are **no** unplanned failed sensors that input to CTP calculation. ..... ☐

NOTE [C14]

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.

- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 3966 MWth. The final approach to rated at 3966 MWth and above will be done with recirculation flow. This caution should **not** preclude the use of inserting rods to raise rod line at the end of cycle, "Reverse Power Effect", for power maintenance. There should be a minimum of three-minute wait prior to any further adjustments with Recirculation Flow.

2) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY a smoothed feedwater flow computer point (NSSQB108) is updating,
OTHERWISE MARK this step N/A. ..... ☐

3) **WHEN** Power is greater than 90%
AND PRIOR TO raising power,
THEN VERIFY proper feedwater heating by using N2-OP-8, Feedwater Heaters and Extraction Steam System, Feedwater Temperature / Thermal Power Limit Figure,
OTHERWISE MARK this step N/A. ..... ☐

4) **IF** Offgas make-up air is **no** longer necessary,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a) **CLOSE** 2ARC-V47..... ☐

b) **INSTALL** cap on 2ARC-V47. ..... ☐

N/A

N/A F.3.7 (Continued)

NOTE

- "Steady State" for the purpose of maintaining licensed power level is defined as:
  - **No** control rod motion
  - **No** operator demanded Reactor Recirculation flow changes or flow changes caused by equipment malfunction
  - Feedwater heating has been allowed to stabilize (typically 15 minutes following power change)
  - Computer point NSSQB001A-60M has been allowed to stabilize (about one hour).
- It is **not** intended that "steady state" include equilibrium Xenon or Samarium.

e. **WHEN** reactor has been at constant power for greater than two hours,
THEN DECLARE reactor is at "Steady State". N/A

f. **NOTIFY** NDO per OP-AA-102-101. N/A

g. **PERFORM** Power Maintenance per section F.2 as necessary. N/A

NOTE

- 2ISC*PT108 which outputs to FWSPA101 and TARS-2237, should be used for setting reactor pressure as FWSPA101 inputs into the CTP calculation.
- If 2ISC*PT108 is **not** available or reliable, use PT109 (TARS-2236) or PAM recorders for pressure adjustment. Call Reactor Engineering for guidance regarding substituting Rx pressure value in CTP calculation and System engineering for evaluation.

h. **WHEN** Reactor power is approximately 100% rated power,
THEN using Section F.5 Trimming of DEHC Pressure Set,
ADJUST Digital EHC pressure set **UNTIL ONE** of the following is met, **OTHERWISE MARK** this step N/A.

☐ Reactor dome pressure is 1020 psig ..... ☐

☐ Pressure Averaging Manifold (PAM) pressure is 976.3 psig as indicated by computer point MSSPA16 or DEHC HMI. ..... ☐ N/A

i. **ADJUST** movement restrictions for 2GMC-TV101 via Side-Mounted Handwheel per N2-OP-26. N/A

F.4. **Power Maneuvering for Parameter Margin Control**

NOTE

- This section is intended to be used to maintain condenser vacuum, hotwell temperature, and generator loading parameters within acceptable limits while at or near rated conditions
- Rated generator output at 3988 MWth is 1369 MWe. However, variations in generator output above 1369 MWe are permitted provided generator operation remains within the generator capability curve of N2-OP-68. The 233 MVAR lagging reactive load limit is based on a generator real load of 1380 MWe,
- Minimum condenser vacuum above 55% power is 23.8" Hg Vacuum.
- During Normal Operation, maximum operating hotwell temperature is less than or equal to 140 °F.
- During high environmental temperature conditions, operation of the condensate system with hotwell temperatures up to 145 °F is permitted provided the following:
  - Prior to exceeding 140 °F, notify Chemistry Management to direct monitoring of Reactor Coolant sulfate concentration.
  - RCS sulfates are maintained below three ppb to avoid impacts to Chemistry Effectiveness Index (CEI).
- Power changes using this section may be performed using Reactor Recirculation flow without a Reactivity Maneuvering Instruction (RMI). IF Control Rods are used, an RMI is required.
- This section is written to maneuver the plant between 85% and 100% as needed to control the above parameters in a controlled manner without having to perform a rapid power reduction per N2-SOP-101D. If further power reduction is necessary, use Section F.3.0 of this procedure or N2-SOP-101D as applicable.

F.4.1. Power Reduction

a. **PERFORM** the following actions **PRIOR TO** commencing down power **IF** time permits:

- **NOTIFY** Generation Dispatch of power reduction. ..... ☐
- **NOTIFY** NDO if required per OP-AA-102-101. ..... ☐
- **REVIEW** applicable Precautions and Limitations. ..... ☐
- **OBTAIN** authorization from Shift Manager to reduce power. .. ☐
- **IF** a ReMA is to be used for the power reduction,
  THEN REVIEW ReMA**,**
  OTHERWISE MARK this step N/A. ..... ☐
- **COMPLETE** applicable reactivity brief. ..... ☐ _____

Form 3.3-1 Scenario Outline

Facility:	Nine Mile Point Unit 2	Scenario #:	NRC-4
Scenario Source:	New	Op. Test #:	2023-2
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 7% power during a startup.		
Turnover:	Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. The procedure is complete up to step F.5.4. Then, continue to withdraw control rods per the startup sequence. The next rod is 34-19 going from 12 to 48.		
Critical Tasks:	CT-1: Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC. The scram must be initiated before Suppression Pool water level reaches 192'. CT-2: Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC. The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Transfer Reboiler Steam Supply to Main Steam N2-OP-25
2	N/A	R – ATC, SRO	Withdraw Control Rods N2-OP-96
3	RW01	I – ATC, SRO TS – SRO	Rod Worth Minimizer Fails Inop ARP 603307, N2-OP-95A, Technical Specifications
4	CU06 CU08	C – BOP, SRO TS – SRO MC – BOP	Reactor Water Cleanup Leak into Reactor Building Closed Loop Cooling System, Automatic Isolation Fails ARP 602320, ARP 602313, Technical Specifications
5	RH13A RH15	C – BOP, SRO TS – SRO MC – BOP	Inadvertent Initiation of Div 1 ECCS with Min Flow Failure ARPs, N2-OP-100A, Technical Specifications
6	PC12	M – All	Suppression Pool Leak N2-EOP-PC, N2-EOP-SC, N2-EOP-RPV, N2-SOP-101C, N2-EOP-C2
7	CS:HP:MT:I	C – BOP, SRO	HPCS Pump Trip, Suppression Pool Makeup from RHR Unavailable N2-EOP-PC
8	TC:PA(B):MT:I	C – BOP, SRO MC – BOP	EHC Pumps Trip N2-EOP-RPV

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Facility: **Nine Mile Point Unit 2**	Scenario No.: **NRC-4**	Op-Test No.: **2023-2**
1. Malfunctions after EOP entry (1-2) **Events 7 & 8**	2	
2. Abnormal events (2-4) **Events 4 & 5**	2	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **N2-EOP-C2**	1	
6. Pre-identified critical tasks (≥2)	2	

Constellation

Copy _____ of _____

Training Id: **November 2023 NRC Scenario 4** Revision: **0.0**

Title: **November 2023 NRC Scenario 4**

	Signature / Printed Name	Date
Developed By	Signature on File / Mike Alexander	10/27/23
Validated By	N/A / Mecki Ghabarou	7/12/23
	N/A / Dan Delaney	7/12/23
	N/A / Mike Carno	7/12/23
Facility Reviewer	Signature on File / John Toothaker	10/27/23

References

1. N2-OP-25, Auxiliary Steam, Auxiliary Condensate, and Gland Seal
2. N2-OP-96, Reactor Manual Control and Rod Position Indication System
3. N2-OP-95A, Rod Worth Minimizer System
4. N2-OP-100A, Standby Diesel Generators
5. N2-EOP-PC (1-3), Primary Containment Control - Flowchart
6. N2-EOP-SC (1-3), Secondary Containment Control – Flowchart
7. N2-SOP-101C, Reactor Scram
8. N2-EOP-RPV (1-3), RPV Control – Flowchart
9. N2-EOP-C2 (1-3), RPV Blowdown - Flowchart
10. N2-EOP-HC, NMP2 EOP Hard Cards Procedure
11. NMP Unit 2 EAL Wallboard - EP-AA-1013 Addendum 4 Appendix 1
12. N2-TSPEC, NMPNS Unit 2 Improved Technical Specifications (Volume 1, 2, 3)
13. OP-AA-112-101, Shift Turnover and Relief

Instructor Information

A. Scenario Description

1. Sequence of Events / Expected Crew Response

   a. The scenario begins at approximately 7% power during a plant startup.

   b. Event 1 is the normal evolution performed by the BOP operator to transfer Reboiler steam to Main Steam per N2-OP-25 section F.5.0.

   c. Event 2 is a reactivity evolution. The ATC operator will continue the startup by withdrawing control rods per the startup sequence.

   d. Event 3 occurs when the Rod Worth Minimizer (RWM) fails. This will result in a control rod block. The SRO will address Technical Specifications. The crew will bypass the RWM to allow continued control rod withdrawals.

   e. Event 4 occurs when WCS begins to leak into the CCP system in the NRHX. WCS will fail to automatically isolate. The crew will manually isolate WCS to stop the leak.

   f. Event 5 occurs when an inadvertent Division I ECCS signal is received. This causes the CSL and RHR 'A' pumps to automatically start and run on minimum flow. During the transient, 2RHS*MOV4A (2RHS*P1A minimum flow valve) fails closed. The crew will evaluate using redundant and independent indications that the ECCS signal is not valid and determine that 2RHS*P1A is running at shutoff head. The crew will then place 2RHS*P1A in P-T-L. The SRO will evaluate Technical Specifications for the inoperability of two ECCS injection systems.

   g. Events 6-8 begin when a leak develops from the Suppression Pool. The crew will enter N2-EOP-SC (1-3) and N2-EOP-PC (1-3). The crew will initiate makeup to the Suppression Pool, but RHR will be unavailable for makeup and HPCS will trip shortly after makeup is initiated. Suppression Pool water level will continue to lower. The

crew will scram the Reactor before Suppression Pool water level reaches 192' *(critical task)*. Suppression Pool water level will continue to lower. EHC pumps will trip when Reactor pressure lowers below 800 psig, limiting the crew's ability to depressurize using Turbine Bypass Valves. Suppression Pool water level will not be able to be maintained above 192', so the crew will perform an RPV Blowdown per N2-EOP-C2 (1-3) *(critical task)*.

2. Termination Criteria

   a. All control rods in.

   b. RPV Blowdown in progress or completed.

   c. Reactor level being controlled in assigned band.

3. Critical Tasks

CT-1.0 Justification:	
Safety Significance:	*Critical Task 1.0 is identified as critical because, given lowering Suppression Pool water level, a Reactor scram is required to allow subsequent action to lower Reactor pressure.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Suppression Pool water level is lowering and cannot be maintained above 192'.*
Success Path:	*Arming and depressing RPV pushbuttons and/or rotating the Reactor Mode Switch to SHUTDOWN*
Performance Feedback:	*Control rod position and Reactor power indications will provide performance feedback regarding the success of the scram.*
Measurable Performance Standard:	*Manipulating RPS pushbuttons and/or the Reactor Mode Switch will provide the evaluation team with observable actions.*
Bounding Conditions:	*N2-EOP-PC (1-3) directs entering N2-EOP-RPV (1-3) and scramming the Reactor before Suppression pool water level drops to 192' (bottom of scale).*

   a. CT-1.0, Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC (1-3). The scram must be initiated before Suppression Pool water level reaches 192'.

CT-2.0 Justification:	
Safety Significance:	*Critical Task 2.0 is identified as critical because loss of Suppression Pool water inventory jeopardizes the ability of the Primary Containment to handle the energy contained in the Reactor Coolant system. An RPV Blowdown is required to lower the energy state of the Reactor Coolant system in this situation.*
Initiating Cue:	*Multiple annunciators and indications will provide evidence that Suppression Pool water level is lowering and cannot be maintained above 192'.*
Success Path:	*Manually initiating ADS and/or opening SRVs*
Performance Feedback:	*SRV and/or TBV instrumentation will provide indication that these valves are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.*
Measurable Performance Standard:	*Manipulation of ADS/SRV controls will provide observable actions for the evaluation team.*
Bounding Conditions:	*Time limit based on Operations representative recommendation.*

b. CT-2.0, Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC (1-3). The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.

4. Length

   a. 60 minutes

5. Mitigation Strategy Code

   a. DMS-PC-02, loss of inventory in suppression pool (Unit 2) or torus (Unit 1); RPV blowdown or anticipate blowdown and depressurize to the main condenser

6. Technical Specifications (Applicable actions for initial conditions only)

   a. None

7. EAL Classification

   a. FA1:

      1) Any loss of any potential loss of either fuel clad or RCS

B. Initial Conditions

1. IC Number

   a. IC-012 or equivalent (IC-203 for ILT 22-2)

2. Presets / With Triggers

   a. Malfunctions

1)	**RW01**, Rod Worth Minimizer Failure, Final = true	**TRG1**
2)	**CU08**, Reactor Water Cleanup Isolation Failure, Final = true	**Inserted**
3)	**CU06**, Reactor Water Cleanup Non-regenerative HX Tube Leak, Final = 50, Ramp = 2 minutes	**TRG2**
4)	**RH13A**, ECCS Inadvertently Initiates (Div I), Final = true	**TRG3**
5)	**RH15**, RHS*MOV4A Valve Fails Shut, Final = true	**TRG3**
6)	**PC12**, Suppression Pool To Reactor Building Leak, Final = 10	**TRG4**
7)	**TC:PA:MT:I**, Turbine EHC Pump A Motor Failure, Final = short	**TRG5**
8)	**TC:PB:MT:I**, Turbine EHC Pump B Motor Failure, Final = short	**TRG5**
9)	**CS:HP:MT:I**, HPCS Motor Failure, Final = short, Delay = 30 sec	**TRG30**

   b. Remotes

      1) None

   c. Overrides

      1) None

   d. Annunciators

      1) None

e. Event Triggers

Event #	Event Action	Command
5	rrpdome<800 (Activates when Reactor pressure lowers below 800 psig)	None
14	anntbl(1039)>1	imf pc12 100
30	hzlcshps2(2)==1 (HPCS pump red light on)	None

f. Equipment Out of Service

1) None

g. Support Documentation

1) Provide N2-OP-25 section F.5.0, completed up to step F.5.4. Also provide section F.1.0.

2) Provide ReMA for plant startup

3) Mark up N2-OP-101A with section E.1.0 initialed off, section E.2.0 initialed off, E.3.0 initialed off, and E.4.0 initialed off up to E.4.10. (open circle step E.4.10).

4) Provide N2-OP-96 section F.2.

h. Miscellaneous

1) None

C. Shift Turnover Information

1. Reactor Power: 7%
2. Rodline: Below 100%
3. Technical Specification LCOs in effect:
   a. None
4. Significant Problems / Abnormalities / Equipment Out of Service:
   a. Plant startup is in progress.

NOTES

5. Evolutions / Maintenance Scheduled for this Shift:

   a. Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. An EO is standing by at the Auxiliary Boilers and by 2CES-IPNL204. The procedure is complete up to step F.5.4.

   b. Then, raise Reactor power by withdrawing control rods per the startup sequence. The next rod is 34-19 going from 12 to 48 (continuous).

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 7% during a plant startup
- Reactor pressure is 925 psig with 1 bypass valve open
- Feed pump A is running with level control on LV55A
- A Reactor startup is in progress per N2-OP-101A. Currently on step E.4.10. The Clean Steam Reboiler is still on the Aux Boilers.
- Currently on NM2C19-SU-1 sequence page 19, rod 34-19 withdrawing control rods from position 12 to position 48
- Reactor Engineering is present and available in the Control Room

PART III: Remarks/Planned Evolutions:

- Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. Place 'A' Clean Steam Reboiler in service per engineering recommendation. An EO is standing by at the Auxiliary Boilers and by 2CES-IPNL204. The procedure is complete up to step F.5.4.
- Then, continue to withdraw control rods per the startup sequence. The next rod is 34-19 going from 12 to 48 (continuous).
- Another team of Operators will be responsible for other startup activities in N2-OP-101A.

Event #1 – Transfer Reboiler Steam Supply to Main Steam

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Close 2ASS-AOV145, AUX BLR STM INLET VLV - Place the Reboiler 2ASS-STV112 and 2ESS-STV104 control switches in the AUTO • Final (expected) operating result - Operating with reboiler steam supply on main steam

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs BOP to transfer Reboiler steam supply to Main Steam per N2-OP-25, Section F.5.0, starting at step F.5.4
Role Play: As EO, when asked, provide indications from the Aux Boilers. Use normal operating indications from N2-OP-25, F.1.0 to respond to any additional requests for indications. **Role Play:** As EO, when asked to locally open 2ASS-V23, wait 1 minute and then report that 2ASS-V23 is open. **Role Play:** As EO directed to depress the reset pushbuttons for 2CNA-LV8A, LV-8B and LV-8C, wait 10 seconds and inform the control room that it has been completed. **Role Play:** As EO directed to confirm that 2CNA-LV8A, LV-8B and LV-8C are modulating to control Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed, wait 1 minute and inform the control room that 2CNA-LV8A, LV-8B and LV-8C are modulating and controlling Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed.	**BOP** • Acknowledges direction to transfer Reboiler steam supply to Main Steam • Directs EO to open 2ASS-V23 • Closes 2ASS-AOV145, AUX BLR STM INLET VLV • Places the Reboiler 2ASS-STV112 and 2ESS-STV104 control switches in the AUTO • Directs EO to depress the RESET PUSHBUTTONS for 2CNA-LV8A, LV-8B and LV-8C on the 2CES-IPNL204 • Opens 2CNA-HV34A • Directs EO to confirm that 2CNA-LV8A, LV-8B and LV-8C are modulating to control Reboiler Drain tank level and that 2CNA-LV9A and LV9B are closed • Monitors for proper Reboiler operation • Reports steam supply transferred

Instructor Actions / Plant Response	Operator Actions
Role Play: If asked, inform the candidate that there is no longer any load on the auxiliary boilers. **Role Play:** If asked as the EO about the following, wait 1 minute then report: • At Main Turbine, no steam is blowing from turbine shaft seals • A water level exists in the Gland Exhaust loop seal as indicated by air not being sucked in	

Event Termination Criteria	• Reboiler steam supply on main steam

Event #2 – Withdraw Control Rods

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Withdraw control rods • Final (expected) operating result - Sufficient control rods withdrawn for adequate evaluation

Instructor Actions / Plant Response	Operator Actions
	SRO • Directs ATC to continue withdrawing control rods per the startup sequence • Provides oversight of reactivity manipulation
Note: The next four control rods to be moved are 34-19, 34-43, 26-19, and 18-35 going from 12 to 48 (continuous withdraw). **Note:** The next event may be inserted whenever the lead evaluator determines that sufficient control rods have been withdrawn for an adequate evaluation. It should be inserted when no rod motion is occurring.	**ATC** • Acknowledges direction to continue withdrawing control rods per the startup sequence • Reviews N2-OP-96 actions for first time rod selection and withdrawal actions: ◦ SELECTS the rod to be moved by momentarily depressing the pushbutton for that rod on the Rod Select Module at 2CEC*PNL603 ◦ Monitors RPV, CRD and Nuclear Instruments ◦ OBSERVES the following: - The pushbutton for selected rod is lit - The white select light on full core display is lit for control rod - The rod's position is indicated AND in correct location on 4-Rod Display - IF available, THEN Rod Worth Minimizer indicates correct rod AND rod position OTHERWISE MARK this step N/A

Instructor Actions / Plant Response	Operator Actions
Note: A coupling check may be done separately or as the rod is being initially withdrawn per the sequence.	◦ Continuous Withdrawal is performed as follows: - VERIFIES Reactor mode switch is not in SHUTDOWN AND there are no rod withdrawal blocks - VERIFIES correct rod has been selected per sequence being used AND that continuous rod withdrawal is allowed for given conditions - SIMULTANEOUSLY DEPRESSES WITHDRAW AND CONTINUOUS WITHDRAW push buttons AND OBSERVE WITHDRAW AND CONTINUOUS WITHDRAW lights are on • Continues control rod withdrawal per the startup sequence • Monitors NI indications, reactor level and reactor pressure
	BOP • Monitors plant parameters to verify proper operations • Provides peer checks as needed

Event Termination Criteria	• A sufficient number of control rods have been withdrawn as determined by the Lead Evaluator

Event #3 – Rod Worth Minimizer Fails Inop

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power - RWM in service • Verifiable Actions - Bypass RWM - Technical Specification 3.3.2.1 evaluated • Final (expected) operating result - RWM bypassed

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction**: **TRG1** **RW01**, Rod Worth Minimzer Failure, Final = true *RWM fails*	
The following annunciator alarms: • *603307, RWM ROD BLOCK*	**ATC** • Recognize / report annunciator 603307

Instructor Actions / Plant Response	Operator Actions

Spec	Condition	Applicable Actions
3.3.2.1	C	C.1 or C.2
Action	**Description**	
C.1	Suspend control rod movement except by scram (immediately)	
C.2	Verify ≥ 12 rods withdrawn (immediately) Or Verify by administrative means that startup with RWM inoperable has not been performed in the last calendar year (immediately AND Verify movement of control rods is in compliance with BPWS by a 2nd licensed operator or other qualified member of the technical staff (during control rod movement)	

Role Play:

If asked for assistance with troubleshooting RWM, report that the RWM has a critical fault and will not be available for the remainder of the startup. If asked, recommend/approve bypassing RWM per N2-OP-95A.

Role Play:

If asked if a startup with an inoperable RWM has been performed in the last calendar year, acknowledge request, wait 2 minutes, then report that a startup with an inoperable RWM has NOT been performed in the last calendar year.

Role Play:

If asked for an additional licensed operator or qualified member of the technical staff to verify control rod movement, acknowledge request.

SRO

- Acknowledges report
- Directs ATC to execute ARP
- Refers to TS 3.3.2.1 and determines Condition C applies
- May contact Reactor Engineering and/or plant management
- Directs ATC to bypass RWM per N2-OP-95A section H.1.0

Instructor Actions / Plant Response	Operator Actions
Role Play: If dispatched to verify two position Computer Display keylock mode switch is in OPER, wait 1 minute, then report that the switch is in OPER.	**ATC** • Executes ARP 603307 • Bypasses RWM per N2-OP-95A section H.1.0: • Coordinates with SRO to review restrictions • Dispatches operator to verify two position Computer Display keylock mode switch is in OPER position at H13-P615, RPIS Panel • Places three position Operator Display keylock mode switch in BYPASS position at panel 2CEC*PNL603 • Observes BYPASS mode indication

Event Termination Criteria	• RWM bypassed • Technical Specifications have been addressed

Event #4 – Reactor Water Cleanup Leak into Reactor Building Closed Loop Cooling System, Automatic Isolation Fails

Event Information	• Presumed or required initial plant operating conditions - Reactor at ~7% power - WCS in service • Verifiable Actions - Isolate WCS • Final (expected) operating result - Reactor at ~7% power with WCS isolated

Instructor Actions / Plant Response	Operator Actions
Booth Operator When directed by the lead evaluator, **insert** the following **malfunction** (insert when rods are not being moved): **TRG2** **CU06**, Reactor Water Cleanup Non-regenerative HX Tube Leak, Final = 50, Ramp = 2 minutes The following **malfunction** is **preset** to cause the failure to isolate: **CU08**, Reactor Water Cleanup Isolation Failure, Final = true	
The following annunciator will alarm ~70 seconds after the malfunction is inserted: • *602320, RWCU DIFF FLOW TIMER BYPASS.* *The following annunciator will alarm 35 seconds after 602320 alarms:* • *602313, RWCU DIFFERENTIAL FLOW HIGH*	**Crew** • Recognizes / reports annunciators • Recognizes / reports WCS leak into CCP • Recognizes / reports failure of WCS to automatically isolate

<table>
<tr><th>Instructor Actions / Plant Response</th><th>Operator Actions</th></tr>
<tr><td>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.3.6.1</td><td>A</td><td>A.1</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>A.1</td><td colspan="2">Place channel in trip (24 hours)</td></tr>
</table>

<table>
<tr><th>Spec</th><th>Condition</th><th>Applicable Actions</th></tr>
<tr><td>3.3.6.1</td><td>B</td><td>B.1</td></tr>
<tr><th>Action</th><th colspan="2">Description</th></tr>
<tr><td>B.1</td><td colspan="2">Restore isolation capability (1 hour)</td></tr>
</table>

Note:
Technical Specification 3.3.6.1 Condition F will also become applicable after 1 hour.

</td><td>

SRO

- Acknowledges reports
- Directs execution of ARPs
- Directs isolation of WCS
- May direct entry into N2-SOP-83
- Evaluates Technical Specification 3.3.6.1 for failure of WCS to automatically isolate, as necessary:
  - Determines that Actions A and B are applicable
  - Determines the isolation flow path is already isolated

</td></tr>
<tr><td>

Note:
The crew may choose to not wait for Annunciator 602313 to alarm. The crew may anticipate the RWCU isolation and take manual actions prior to the actual isolation signal. This is an acceptable action.

</td><td>

BOP

- Executes ARPs
- May execute N2-SOP-83
- Isolates WCS:
  - Closes 2WCS*MOV102, Cleanup Suct Inboard Isol Vlv
  - Closes 2WCS*MOV112 Cleanup Suct Outboard Isol Vlv
  - Verifies 2WCS-P1A trips
  - Throttles open 2WCS-MOV110, CLEANUP DEMIN BYPASS VLV

</td></tr>
</table>

Event Termination Criteria	• WCS isolated

Event #5 – Inadvertent Initiation of Div 1 ECCS with Min Flow Failure

Event Information	Enter important information about the event here such as: • Presumed or required initial plant operating conditions - Reactor at ~7% power • Verifiable Actions - Place 2RHS*P1A in P-T-L • Final (expected) operating result - Reactor at ~7% power with 2RHS*P1A in P-T-L

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG3** **RH13A**, ECCS Inadvertently Initiates (Div I), Final = true **RH15**, RHS*MOV4A Valve Fails Shut, Final = true • *Division I low pressure ECCS pumps receive initiation signals and auto start* • *Division I Diesel Generator starts* • *RHS*MOV4A closes* • *LPCI A / LPCS initiation white light illuminates*	
The following annunciators alarm: • *601413, LPCS PUMP 1 AUTO START* • *601426, LPCS SYSTEM ACTUATED* • *601442, RHR PUMP 1A AUTO START* • *601451, RHR A SYSTEM ACTUATED* • *601539, ADS A LPCS / RHR A PERMISSIVE* • *852117, EDG 1 RUNNING*	**BOP** • Silences, acknowledges and reports annunciators • Recognizes / reports Div I low pressure ECCS initiation • Recognizes / reports 2RHS*MOV4A, RHR 'A' minimum flow valve, is closed

Instructor Actions / Plant Response	Operator Actions
Note: The following Technical Specifications apply for placing 2RHS*P1A in P-T-L:	**SRO** • Acknowledges reports • Directs execution of ARPs • Directs 2RHS*P1A placed in P-T-L • May direct 2CSL*P1 placed in P-T-L. • May direct Division I Diesel Generator shutdown per N2-OP-100A • Evaluates Technical Specifications 3.5.1, 3.6.1.6, 3.6.2.3, and 3.6.2.4

Spec	Condition	Applicable Actions
3.5.1	A	A.1
Action	**Description**	
A.1	Restore low pressure ECCS injection/spray subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.1.6	A	A.1
Action	**Description**	
A.1	Restore RHR drywell spray subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.2.3	A	A.1
Action	**Description**	
A.1	Restore RHR suppression pool cooling subsystem to OPERABLE status. (7 days)	

Spec	Condition	Applicable Actions
3.6.2.4	A	A.1
Action	**Description**	
A.1	Restore RHR suppression pool spray subsystem to OPERABLE status. (7 days)	

Note:
The following Technical Specification will also apply if 2CLS*P1 is placed in P-T-L:

Spec	Condition	Applicable Actions
3.5.1	C	C.1
Action	**Description**	
C.1	Restore one ECCS injection/spray subsystem to OPERABLE status. (72 hours)	

Note:
Technical Specification 3.3.5.1 would also possibly apply depending on troubleshooting results.

Instructor Actions / Plant Response	Operator Actions
Role Play: If requested to provide RHR and CSL trip unit indications (2CEC*PNL629), report that no trip units are in alarm and that no gross failures are present. **Role Play:** If dispatched as EO to monitor diesel running parameters, wait 5 minutes and then report that all diesel running indications are normal for the Division I EDG.	**BOP** • Executes ARPs • Verifies the validity of the LOCA initiation using at least two redundant indications • Ensures by two separate and redundant means that the initiation is not valid • Checks trip unit indications for status • Places 2RHS*P1A in P-T-L • Dispatches EO to perform running checks on the Div I EDG • If directed, places 2CSL*P1 in P-T-L • If directed, shuts down the Division I diesel generator per N2-OP-100A, Section H.16.0: ◦ Places EMERGENCY DSL GEN 1 LOCA SIGNAL BYPASS switch to ON ◦ Places DIVISION I 2EGS*EG1 START switch in PULL-TO-LOCK

Event Termination Criteria	• 2RHS*P1A has been placed in P-T-L • Technical Specifications addressed

Events #6, 7, & 8 – Suppression Pool Leak; HPCS Pump Trip, Suppression Pool Makeup from RHR Unavailable; EHC Pumps Trip

Event Information	• Presumed or required initial plant operating conditions - Reactor ~7% power • Verifiable Actions - Scram the Reactor - Depressurize the Reactor • Final (expected) operating result - Reactor power <4% - All control rods in - Reactor pressure lowered • Mitigation Strategy code (if applicable) - DMS-PC-02, loss of inventory in suppression pool (Unit 2) or torus (Unit 1); RPV blowdown or anticipate blowdown and depressurize to the main condenser
Critical Tasks ***(Ensure the task action is bolded and italicized in the "Operator Actions" column where the critical task is performed)***	1. Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will insert a manual Reactor scram, in accordance with N2-EOP-PC (1-3). The scram must be initiated before Suppression Pool water level reaches 192'. 2. Given an un-isolable Suppression Pool water leak and the inability to maintain Suppression Pool water level above 192', the crew will perform an RPV Blowdown, in accordance with N2-EOP-PC (1-3). The depressurization must be initiated within 15 minutes of Suppression Pool water level reaching 192'.

Instructor Actions / Plant Response	Operator Actions
Booth Operator: When directed by the lead evaluator, **insert** the following **malfunctions**: **TRG4** **PC12**, Suppression Pool To Reactor Building Leak, Final = 10 When annunciator 851443, Reactor Building General Area Tank 2A Flooding, is received (anntbl(1039) > 1), verify the following **malfunction modifies**: **PC12**, Suppression Pool To Reactor Building Leak, Final = 100 *The following malfunctions will automatically insert on **TRG5** when Reactor pressure lowers below 800 psig to cause EHC pumps to trip:* **TC:PA:MT:I**, Turbine EHC Pump A Motor Failure, Final = true **TC:PB:MT:I**, Turbine EHC Pump B Motor Failure, Final = true	
The following annunciators alarm: • *851443, REACTOR BLDG GENERAL AREA TANK 2A FLOODING* • *851444, REACTOR BLDG GENERAL AREA TANK 2E FLOODING* • *851453, REACTOR BLDG FLOOR DRAIN SYSTEM TROUBLE*	**Crew** • Recognizes / reports leakage in Reactor Building • Recognizes / reports lowering Suppression Pool water level

Instructor Actions / Plant Response	Operator Actions
Role Play: When dispatched to inspect the Reactor Building for indications of flooding, wait 3 minutes and then report there is a crack in the Suppression Pool wall on RB 175' elevation (~3 feet long and 5 feet above floor level) and there is a lot of water pouring out. **Role Play:** If contacted as Radwaste, report that both pumps are running for each affected tank (2A and 2E).	**SRO** • Acknowledges reports • Enters N2-EOP-SC (1-3) on high RB floor drain and area water levels: ◦ Determines no HVR isolation setpoint has been exceeded ◦ Directs operating sump pumps to restore and maintain drain sump level below high-high setpoint ◦ Determines cannot remove water from area and transitions to 28 ◦ Attempts to isolate all discharges into affected areas ◦ Determines leak is not from primary system ◦ Monitors for 2 areas above Max Safe Water level (flooding alarm) • Enters N2-EOP-PC (1-3) on low Suppression Pool water level • Directs filling suppression pool (N2-OP-31 section H.6.0 or N2-OP-33 section H.2.0 or H.3.0) • Determines cannot maintain Suppression Pool water level above 199.5’ and transitions to 17 • Enters N2-EOP-RPV (1-3) • **Directs manual Reactor scram (CT-1.0)** • Directs entry into N2-SOP-101C • Acknowledges scram report • Performs N2-EOP-RPV (1-3) actions: ◦ Performs level control actions: ○ Verifies needed auto isolations, ECCS starts and diesel generator response occurred per EOP-6.1 ○ Determines that RPV water level is known ○ Determines that RPV water level can be restored and maintained between 159.3 and 202.3 inches using Feedwater, HPCS, RCIC, CRD, and/or SLC

Instructor Actions / Plant Response	Operator Actions
	○ Directs an RPV level band of 160 – 200 inches ○ May direct HPCS level 8 interlocks defeated per N2-EOP-6.20 ◦ Performs Pressure control actions: ○ Determines that no SRVs are cycling ○ Determines that RPV pressure can be stabilized below 1052 psig using the main turbine bypass valves ○ Directs a pressure band of 800-1000 psig
	ATC • Acknowledges SRO direction to scram the reactor • **Arms AND depresses BOTH Manual Scram pushbuttons on either side of 2CEC*PNL603 (CT-1.0)** • **Places the Reactor Mode Switch in the SHUTDOWN position (CT-1.0)** • Provides scram report, by reporting: ◦ Reactor mode switch in shutdown ◦ APRMs downscale ◦ Reactor pressure and trend ◦ Reactor level and trend ◦ MSIVs open ◦ Feedwater pumps are running ◦ All control rods fully inserted • Performs N2-SOP-101C actions: ◦ Verifies: ○ All rods full in ○ Reactor power lowering ○ SDV V & D Valves closed ○ RPV pressure on TBVs or SRVs ○ FWLC controlling level > 159.3 ◦ Fully inserts IRMs and SRMs ◦ Energizes 2WCS-MOV107 ◦ If required, secures makeup to the cooling tower

Instructor Actions / Plant Response	Operator Actions
	◦ Performs Level control leg: ○ Recognizes that level will be stabilized using Feedwater ○ If directed defeats HPCS Level 8 trips by directing a field operator to perform the actions of N2-EOP-6.20
Note: N2-OP-33 section H.3.0 would provide the highest flow rate makeup to the Suppression Pool, however it will not work due to the trip of the HPCS pump. The crew may still attempt to utilize N2-OP-33 section H.2.0 and/or N2-OP-31 to provide some Suppression Pool makeup. **Role Play:** When contacted as field operator to open Condensate Flush Supply valves, wait 3 minutes, then make the report that the valve is stuck, unable to open valve	**BOP** • May attempt Suppression Pool makeup using N2-OP-31 section H.6.0: • Verifies LPCI operability requirements • Performs the following: ◦ Depress RHR A (B, C) MANUALLY OUT OF SVCE pushbutton ◦ Verifies amber light is lit • Verifies annunciator 601431 (601, 631) RHR A (B,C) SYSTEM INOPERABLE is in alarm • Places 2RHS*P1A (B, C) control switch in PULL-TO-LOCK • For the applicable RHR system, unlock AND open the following valve: ◦ For RHR A, 2RHS*V71 ◦ For RHR B, 2RHS*V22 ◦ For RHR C, 2RHS*V21 • May attempt Suppression Pool makeup using N2-OP-33 section H.2.0: ◦ Verify SM has declared CSH System inoperable ◦ Verifies open 2CSH*MOV101, PUMP SUCT FROM CNDS TK ◦ Monitors AND maintains 2CSH*PI117, HPCS SYSTEM PRESS PMP, to greater than or equal to 65 psig ◦ Throttles open 2CSH*MOV111, TEST RETURN TO SUPPRESSION POOL to begin transferring water from CST to Suppression Pool ◦ Monitors CST AND Suppression Pool Levels

Instructor Actions / Plant Response	Operator Actions
Booth Operator: If crew establishes Suppression Pool makeup with HPCS, verify the HPCS pump trips on the following **malfunction**: **TRG30, CS:HP:MT:I, HPCS Motor Failure, Final = short, Delay = 30 sec** **Role Play:** If directed to defeat the RCIC level 8 logic, wait 1 minute then **manually insert** the following **remote**: **RC02A**, RCIC Level 8 Trip Defeat:Withdraw Trip Units N693A,E, Final = Defeated **Role Play** If contacted as field operator to defeat the Offgas high radiation isolation per N2-OP-42 Attachment 6, wait 1 minute and **insert** the following **remote**: **OG03**, Off Gas High Radiation Isolation Defeated, Final = Defeated Then report that the Offgas high radiation isolation has been defeated. **Booth Operator:** If directed to defeat the RHR group 5 isolation, **manually insert** the following **malfunction**: • **RH08,** GROUP 5 ISOLATION FAILURE – (RHS*MOV122/113), Final = true Wait 1 minute, then report that the RHR group 5 isolation has been defeated.	• May attempt Suppression Pool makeup using N2-OP-33 section H.3.0: ◦ Verifies subsection F.1.0, Standby Condition Status Checks, is complete ◦ Verifies SM has declared CSH system inoperable ◦ Starts 2CSH*P1, HPCS PUMP 1 ◦ Verifies open2 CSH*MOV105, MINIMUM FLOW BYPASS VLV ◦ Monitors CST and Suppression Pool Levels ▪ Recognizes / reports trip of HPCS pump • Maintains RPV pressure 800 - 1000 psig using EHC and/or SRVs • Recognizes / reports trip of EHC pump ○ May attempt to start standby EHC pump ○ Recognize / report trip of 2nd EHC pump

Instructor Actions / Plant Response	Operator Actions
	SRO • Determines cannot maintain Suppression Pool water level above 192' • Enters N2-EOP-C2 (1-3) • Answers "Will the reactor stay shutdown without boron?" Yes • Determines Drywell pressure is <1.68 psig • **If Suppression Pool water level is above 192', directs open all 7 ADS valves (CT-2.0)** • **If Suppression Pool water level is at or below 192', then rapidly depressurizes the RPV using one or more RPV Blowdown Systems (CT-2.0)**
	ATC/BOP • Recognize / report Suppression Pool water level cannot be maintained above 192' • **Performs RPV Blowdown as directed by SRO (CT-2.0):** ◦ May open all 7 ADS valves: ◦ If an ECCS pump is operating performs the following to open all 7 ADS valves using N2-EOP-HC attachment 5: ◦ Determines that no SRVs are stuck open and that an ECCS pump is operating ◦ Arms and depresses both ADS logic pushbuttons for Division I: - ADS LOGIC "A" manual pushbutton - ADS LOGIC "E" manual pushbutton ◦ Arms and depresses both ADS logic pushbuttons for Division II: - ADS LOGIC "B" manual pushbutton - ADS LOGIC "F" manual pushbutton

Instructor Actions / Plant Response	Operator Actions
	◦ If an ECCS pump is not operating performs the following to open all 7 ADS valves using N2-EOP-HC attachment 5: ◦ Takes control switches to OPEN at EITHER of the following panels UNTIL a total of 7 SRVs are open: - 2CEC*PNL628 ○ 2MSS*PSV137 ○ 2MSS*PSV127 ○ 2MSS*PSV126 ○ 2MSS*PSV121 ○ 2MSS*PSV134 ○ 2MSS*PSV130 ○ 2MSS*PSV129 - 2CEC*PNL631 ○ 2MSS*PSV137 ○ 2MSS*PSV127 ○ 2MSS*PSV126 ○ 2MSS*PSV121 ○ 2MSS*PSV134 ○ 2MSS*PSV130 ○ 2MSS*PSV129 ◦ May rapidly depressurize the RPV using one or more RPV Blowdown Systems

Event Termination Criteria	• All control rods in. • RPV Blowdown in progress or completed. • Reactor level being controlled in assigned band.

Simulator Scenario Handout Reference Packet

UNIT 2 Control Rod Move Sheet Form

FROM: 08 **TO:** 12

Sequence ID: NM2C19-SU-1 Rev. 0
Page: 17

Concurrence to Proceed ALL				WITHDRAW						INSERT			RECOVERY					
ACD QNE INIT	DJK SRO INIT	Step	Control Rod	From	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	To	Move Complete INIT RO	Move Complete INIT PEER	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	Comments
ACD	DJK	1	26-43	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	2	34-19	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	3	34-43	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch High Worth
ACD	DJK	4	26-19	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	5	18-35	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	6	42-27	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	7	42-35	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	8	18-27	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	9	50-19	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	10	10-19	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	11	50-43	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	12	10-19	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch

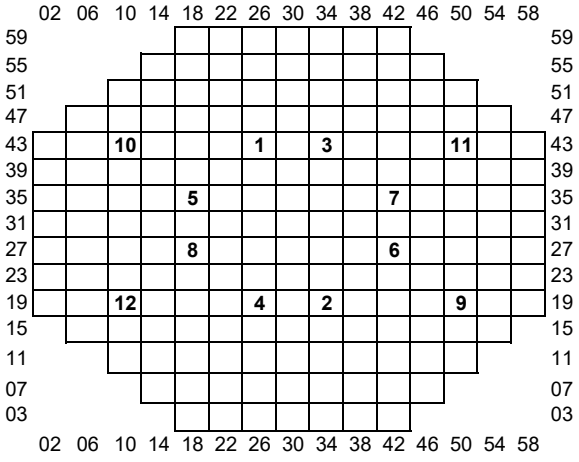
Prepared by: Paul Blejwas

Date: 3/15/2022

Verified by: Kahn, Gerald J

Date: Today

NOTES
GE Rod Group 8



UNIT 2 Control Rod Move Sheet Form

FROM: 08 **TO:** 12

Sequence ID: NM2C19-SU-1 Rev. 0

Page: 18

Concurrence to Proceed ALL				WITHDRAW						INSERT			RECOVERY					
ACD QNE INIT	DJK SRO INIT	Step	Control Rod	From	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	To	Move Complete INIT RO	Move Complete INIT PEER	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	Comments
ACD	DJK	1	18-51	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	2	42-11	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	3	42-51	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch
ACD	DJK	4	18-11	08	12	OPE	VER	N/A	N/A	08	N/A	N/A	12	N/A	N/A	N/A	N/A	Single Notch

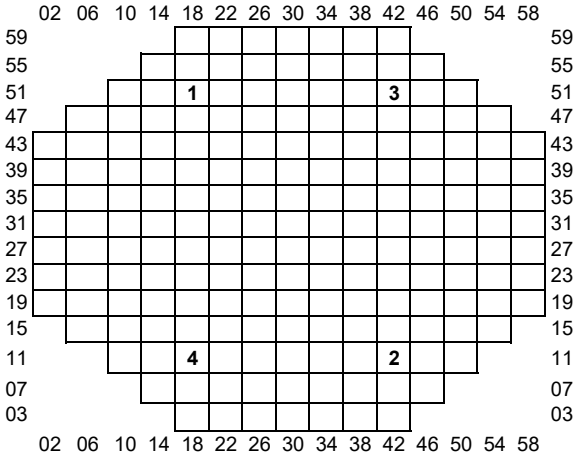
Prepared by: Paul Blejwas

Date: 3/15/2022

Verified by: Kahn, Gerald J

Date: Today

NOTES
GE Rod Group 8



UNIT 2 Control Rod Move Sheet Form

FROM: 12 **TO:** 48

Sequence ID: NM2C19-SU-1 Rev. 0

Page: 19

Concurrence to Proceed ALL				WITHDRAW						INSERT			RECOVERY					
ACD QNE INIT	DJK SRO INIT	Step	Control Rod	From	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	To	Move Complete INIT RO	Move Complete INIT PEER	To	Move Complete INIT RO	Move Complete INIT PEER	Coupling Check INIT	Full Out Light (√)	Comments
ACD	DJK	1	26-43	12	48	OPE	VER	OPE	VER	12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous High Worth
ACD	DJK	2	34-19	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	3	34-43	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous High Worth
ACD	DJK	4	26-19	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	5	18-35	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous High Worth
ACD	DJK	6	42-27	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	7	42-35	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous High Worth
ACD	DJK	8	18-27	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous High Worth
ACD	DJK	9	50-19	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	10	10-43	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	11	50-43	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous
ACD	DJK	12	10-19	12	48					12	N/A	N/A	48	N/A	N/A	N/A	N/A	Continuous

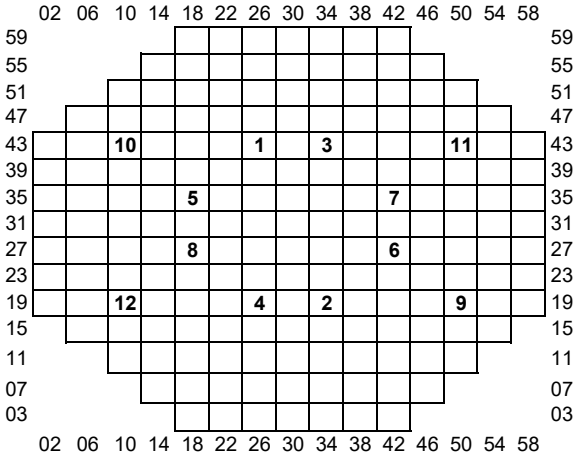
Prepared by: Paul Blejwas

Date: 3/15/2022

Verified by: Kahn, Gerald J

Date: Today

NOTES
GE Rod Group 8



Simulator Scenario Handout Reference Packet

Constellation

SHIFT TURNOVER INFORMATION

ON COMING SHIFT: ☐ N ■ D DATE: Today

PART I: To be performed by the oncoming Operator before assuming the shift.

- Control Panel Walkdown (all panels) (SRO, ROs)

PART II: To be reviewed by the oncoming Operator before assuming the shift.

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 7% during a plant startup
- Reactor pressure is 925 psig with 1 bypass valve open
- Feed pump A is running with level control on LV55A
- A Reactor startup is in progress per N2-OP-101A. Currently on step E.4.10. The Clean Steam Reboiler is still on the Aux Boilers.
- Currently on NM2C19-SU-1 sequence page 19, rod 34-19 withdrawing control rods from position 12 to position 48
- Reactor Engineering is present and available in the Control Room

PART III: Remarks/Planned Evolutions:

- Transfer Reboiler steam supply to Main Steam per N2-OP-25 section F.5.0. Place 'A' Clean Steam Reboiler in service per engineering recommendation. An EO is standing by at the Auxiliary Boilers and by 2CES-IPNL204. The procedure is complete up to step F.5.4.
- Then, continue to withdraw control rods per the startup sequence. The next rod is 34-19 going from 12 to 48 (continuous).
- Another team of Operators will be responsible for other startup activities in N2-OP-101A.

NINE MILE POINT NUCLEAR STATION UNIT 2
OPERATING PROCEDURE

N2-OP-25

AUXILIARY STEAM, AUXILIARY CONDENSATE, AND GLAND SEAL

Revision 01900

10 CFR 50.59 Review Required

Safety Related

CONTINUOUS USE

Approval Authority: Director Operations

SUMMARY OF ALTERATIONS

Minor revision to incorporate the following.

PCR-18-06121
Description: Provide clarifying detail for when to transfer Gland Seal supply to Main Steam when filling the standby Clean Steam Reboiler. Improvement taken to reduce/eliminate instances of vague or non-specific steps per IR 04172818.

- Step F.9 Note, updated the Note's second bullet.
- Step H.1.5.a, added SM direction and split it into two steps.
- Step H.9.1, updated the IF contingency.

PCR-20-01423
Description: Allow Gland Seal to be in service with Main Lube Oil secured to allow for maintenance on Seal Oil with vacuum maintained.

- P&L D.1, added Ops Mgmt. permission for Appling Gland Seals with a rotor not on the Turning Gear.
- Step E.2.1, updated the Caution statement to match P&L D.1 and aligned the step to starting on the Turning Gear.

Operations Review

- Subsection A.5, added numbering to the references. No revision bars used.
- Deleted P&L 1 to remove outdated legacy precaution per IR 04172818. Steps automatically renumbered. No revision bar used.
- Applied formatting per current procedural standards of AD-AA-101-1002 and AD-NM-101-1005.
- Step F.9.1, separated into two separate steps.
- Step H.9, updated wording to correct spelling of "out-of-service" since it is an adjective that immediately precedes a noun.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 4
B. SYSTEM DESCRIPTION 7
C. OPERATING REQUIREMENTS 10
D. PRECAUTIONS AND LIMITATIONS 11
E. STARTUP 12
E.1. Clean Steam Reboiler 1A(B) Startup 12
E.2. Gland Seal Startup 18
F. NORMAL OPERATIONS 21
F.1. Normal Operating Indications 21
F.2. Shifting Gland Seal Supply from Reboiler Steam to Main Steam 23
F.3. Shifting Gland Seal Supply from Main Steam to Reboiler Steam 24
F.4. Shifting Reboiler Steam Supply from Extraction/Main Steam to Auxiliary Boiler Steam [EOP] 25
F.5. Shifting Reboiler Steam Supply from Auxiliary Boiler Steam to Extraction/Main Steam 26
F.6. Swapping Gland Exhaust Fans 28
F.7. Swapping Clean Steam Reboilers with Steam Supply from Main Steam or Extraction Steam 30
F.8. Verifying Automatic Swapover of Reboiler Steam Supply 35
F.9. Blowdown of Clean Steam Reboiler Shell 36
G. SHUTDOWN 37
G.1. Securing Gland Seal [EOP] 37
G.2. Securing Clean Steam Reboiler 1A(B) [EOP] 38
H. OFF-NORMAL OPERATIONS 40
H.1. Steam Blowing from Turbine Seals 40
H.2. Manual Control of Auxiliary Steam Supply Pressure to Clean Steam Reboilers 41
H.3. Manual Initiation of Main Steam Supply to Gland Seal Steam 42
H.4. Manual Control of Main Steam Supply to Gland Seal Steam 43
H.5. Manual Control of Main Steam Supply Pressure to Clean Steam Reboilers 45
H.6. Raising Gland Steam Seal Pressure and Lowering Gland Exhaust Pressure to Minimize Offgas System Flow 46
H.7. Preventing Automatic Swapover of Reboiler Steam Supply 49
H.8. Clean Steam Reboiler 1A(B) Startup with Extraction Steam 50
H.9. Filling the Standby Clean Steam Reboiler 55
Attachment 1, Walkdown Controller Lineup 57
Attachment 2, Using the ABB C310 60

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

None

A.2. Licensee Documentation

Updated Safety Analysis Report (USAR), Volume 24, Section 10.4.3, Turbine Gland Sealing System

A.3. Standards, Regulations and Codes

None

A.4. Policies, Programs, and Procedures

- N2-OP-3, Condensate and Feedwater System
- N2-OP-19, Instrument and Service Air System
- N2-OP-48, Auxiliary Boiler System
- N2-OP-64, Turbine Building Drains
- N2-OP-71C, 600V A.C. Power Distribution

A.5. Technical Information

A.5.1. Flow Diagrams

a. P&IDs 25A - 25G, Clean Steam Reboiler And Auxiliary Cond

A.5.2. Electrical Diagrams

a. ESK-6ASS01, Clean Stm Reboiler Inlet Isol Valves

b. ESK-6ASS02, Cond Air Removal Mn Stm Block Valve

c. ESK-6ASS04, Bldg Heating Intmd Ht Exch Inl Isol V

d. ESK-6TME04, Reboiler Steam Outlet MOVUs

e. ESK-6TME05, Reboiler Gland Seal Steam Feed MOVs

f. ESK-6TME06, Gland Seal Steam Aux Supply Valve

g. ESKs-6TME07 and 08, Steam Pkg Exh Fans

h. ESKs-6TME09 and 10, Steam Pkg Exh Fan Inlet MOVs

A. REFERENCES AND COMMITMENTS (Continued)

A.5.2 Electrical Diagrams (Continued)

i. ESK-6TME11, Reboiler Shell Blowdown MOVs

j. ESKs-7ASS01 - 03, Aux Steam Valves

k. ESK-7CNA01, Clean Steam Reboiler Control V

l. ESK-7CNA03, Bldg Heat Exchanger-Valves

m. ESK-7CNA04, Clean Stm Reboiler Valves

n. ESK-7CNA05, Reboiler Drain Control Valves

o. ESK-7CNA06, Reboiler Drain Tk Cont & Isolation V

p. ESK-7ESS07, Turbine Extraction Steamline Isolation Valve

q. ESKs-7HDL04 - 06, 3rd & 4th Point Heater Control

r. ESK-7HDL10, 1st & 2nd Point Heaters Auxiliary Cont

s. ESKs-7TME01 and 02, Gland Seal Steam Valves

A.5.3. Logic Diagrams

a. LSK-3-1B, Main Steam

b. LSKs-3-4C and 4E, Extraction Steam

c. LSKs-3-9A - 9F, Auxiliary Steam

d. LSKs-4-4.1A - 1D, Auxiliary Condensate

e. LSKs-16-1.1A - 1E, Turbine Generator Gland Seal Steam

f. LSKs-32-5A, 5F, and 5G, Turbine Plant Miscellaneous Drains

A.5.4. Vendor Manuals

a. File Number N20326, Steam Turbine-Generator Instruction Manual (GEK-49966, Volume I, Section 6, Steam Seal System)

A. REFERENCES AND COMMITMENTS (Continued)

A.6. Supplemental References

a. Set Point Data Sheets for 2ASS-PIC106, PIC107, PIC113, PIC125, PIC139, PIC140, and PIC146

b. Set Point Data Sheets for 2CNA-LIC4A/B, LV8A/B/C, LV9A/B, and LIC101

c. Set Point Data Sheets for 2TME-LIC13A/B and PIC122

d. DCP N2-05-021, Add a Spectacle Flange to 2-TME-010-38-4

e. DCP N2-01-164, Turbine Generator Gland Seal and Exhaust Steam

f. ECP-09-000192, ESR-09-003880 NMP-ESR (0000) - Level Controls For 2TME-TK1

g. ECP-11-000977, ESR-11-000636 TME upgrade for operation at EPU conditions.

h. ECP-14-000923, Auxiliary Steam to Radwaste Isolation Enhancement.

A.7. **Commitments**

Sequence Number	Commitment Number	Description
None		

B. SYSTEM DESCRIPTION

The Auxiliary Steam, Auxiliary Condensate and Gland Seal System consists of three major subsystems: Auxiliary Steam (ASS) System, Auxiliary Condensate (CNA) System and the Gland Seal and Exhaust (TME) System.

The Auxiliary Steam (ASS) System supplies reduced pressure steam to the Steam Jet Air Ejectors (SJAEs), the Offgas (OFG) Preheaters, the Building Heating (HVH) Intermediate Heat Exchangers, and the Clean Steam Reboilers. Auxiliary steam is supplied from the Main Steam (MSS) System through a line tapping off one of the Turbine Bypass Chest supply lines. The ASS System supply line is equipped with a motor operated blocking valve, 2ASS-MOV148.

The SJAEs are normally supplied from the MSS System through a branch line from the ASS System supply line via motor operated blocking valve, 2ASS-MOV152. The supply line to the SJAEs branches into two lines and each line has an air operated pressure control valve, 2ASS-PV107 and PV139, to maintain pressure at 128 psig. The two lines are cross-connected downstream of the pressure control valves where they are joined by a supply line from the Auxiliary Boiler (ABM) System. When Main Steam is not available, the ABM System can supply steam to the SJAEs through air operated blocking valve, 2ASS-AOV147. 2ASS-MOV152 and AOV147 are interlocked so that only one valve may be open at a time to prevent cross-connecting Main Steam to the Auxiliary Boilers.

The OFG Preheaters are normally supplied from the MSS System through a line tapping off upstream of 2ASS-MOV152 via solenoid operated valve, 2ASS-SOV142. Air operated pressure control valve, 2ASS-PV125, maintains Offgas Preheater steam pressure at 96 psig. A line from the ABM System connects to the supply line to provide an alternate steam supply via solenoid operated valve, 2ASS-SOV138. The ABM System supply line also has an air operated pressure control valve, 2ASS-PV140, set at 96 psig. The supply line branches to the two preheaters with each line equipped with a solenoid operated valve, 2OFG-SOV14A(B). Auxiliary steam is also used to warm up the OFG System for system startup. 2ASS-SOV142 and SOV138 are interlocked so that only one valve may be open at a time to prevent cross-connecting Main Steam to the Auxiliary Boilers. 2ASS-SOV142 has a normally open bypass valve, 2ASS-V290, around it to provide sufficient flow to the OFG Preheaters.

The HVH Intermediate Heat Exchangers are supplied by a branch line through fast-acting air operated valve, 2ASS-STV143. Air operated pressure control valve, 2ASS-PV106, maintains auxiliary steam supply pressure at 110 psig. The auxiliary steam supply line is joined downstream of 2ASS-PV106 by the normal supply line from the Extraction Steam (ESS) System. The supply line from the ESS System is isolated by fast-acting air operated valve, 2ESS-STV105. The supply line then divides into two lines, each with a motor operated valve (2ASS-MOV8A/B) and a pneumatic temperature control valve (2ASS-TV5A/B), which supply the HVH Intermediate Heat Exchangers.

2ASS-STV143 will auto open and 2ESS-STV105 will auto close on a turbine Trip or when the 5th Point Extraction Steam pressure is less than 105 psig. 2ASS-STV143 will auto close and 2ESS-STV105 will auto open when the 5th Point Extraction Steam pressure is greater than 115 psig. To prevent cross connecting Extraction Steam to Auxiliary Steam, 2ASS-STV143 can only be manually opened when 2ESS-STV105 is closed. 2ASS-MOV8A(B) will auto close when 2HVH-E1A(B) water level is High-High (4" above normal).

B. SYSTEM DESCRIPTION (Continued)

The Gland Seal and Exhaust (TME) System provides steam to the Main Turbine Shaft Sealing System and the Radwaste Evaporator Reboilers. The TME System consists of the following major components: two 100% capacity Clean Steam Reboilers (2TME-E1A/B) and Reboiler Drain Tanks (2TME-TK1A/B), two 100% capacity Gland Exhaust Coolers (2TME-E2A/B), four 100% capacity Gland Exhaust Fans (2TME-FN1A/1B/2A/2B), and one Gland Exhaust Cooler Drain Receiver (2TME-TK1).

The Clean Steam Reboilers and Reboiler Drain Tanks are located inside individual shielded rooms on Turbine Building Elev 306 Northeast corner. Each reboiler can produce 79,000 lbs/hr of non-radioactive steam, at 50 psig and 298°F for use as turbine gland seal steam and for use in the Radwaste Regenerant and Waste Evaporator Reboilers. The Clean Steam Reboiler steam supply to Radwaste is normally isolated by use of two spectacle flanges. 2TME-RV135 provides over pressure protection for the piping downstream of the spectacle flanges and is gagged since the piping protected by it is no longer in service. If the steam line to the Radwaste systems is placed back in service 2TME-RV135 will need to have the gag removed.

Steam to the tube side of the reboilers can be supplied from Extraction Steam (normal supply), Auxiliary Steam, or the Auxiliary Boiler. The Extraction Steam supply line is isolated by fast-acting air operated valve, 2ESS-STV104. The Auxiliary Steam supply line is isolated by fast-acting air operated valve, 2ASS-STV112. 2ASS-PV113 maintains the Auxiliary Steam supply pressure at 95 psig and is located downstream of 2ASS-STV112. The Auxiliary Boiler supply line is isolated by air operated valve 2ASS-AOV145. 2ASS-PV146 maintains Auxiliary Boiler supply pressure at 95 psig and is located upstream 2ASS-AOV145.

2ASS-STV112 will auto open and 2ESS-STV104 will auto close on a Turbine Trip or when 4th Point Extraction Steam pressure is less than 50 psig. 2ASS-STV112 will auto close and 2ESS-STV104 will auto open when 4th Point Extraction Steam pressure is greater than 57 psig. 2ASS-STV112 will also auto close when Main Steam Inlet Header pressure is less than 200 psig. To prevent cross-connecting the Auxiliary Boilers with Extraction or Auxiliary Steam, 2ASS-AOV145 has to be closed to allow 2ESS-STV104 or 2ASS-STV112 to be opened either automatically or manually. To prevent cross-connecting Extraction Steam to Auxiliary Steam, 2ASS-STV112 can only be manually opened when 2ESS-STV104 is closed. 2ASS-AOV145 will auto open when Main Steam Inlet Header pressure is less than 200 psig after a 180 second time delay. 2ESS-STV104 and 2ASS-STV112 have to be closed to allow 2ASS-AOV145 to be opened either automatically or manually.

The steam supply to each reboiler can be isolated by steam inlet valve, 2ASS-MOV3A(B). 2ASS-MOV3A(B) will auto close on reboiler low level (4" below normal) or reboiler drain tank high-high level (14" above normal). Steam condensed in the reboiler tubes is directed to its associated Reboiler Drain Tank, 2TME-TK1A(B). Each Reboiler Drain Tank provides a reference level for drain flow.

Normal drain tank level control permits the condensed steam to drain to the 3rd Point Feedwater Heaters through level control valves 2CNA-LV8A, LV8B, and LV8C. These level control valves modulate to maintain a normal level in the Reboiler Drain Tank. 2CNA-LV8A (B,C) will auto close on high-high level in the associated 1st or 2nd Point Feedwater Heater or high level in the associated 3rd Point Feedwater Heater.

B. SYSTEM DESCRIPTION (Continued)

If high level occurs in either Reboiler Drain Tank, an associated high level control valve, 2CNA-LV9A(B), opens to direct water to the Main Condenser. 2CNA-LV9A(B) are used for drain tank level control during reboiler startup and operation on Auxiliary Boiler steam supply. The drain controls are located on 2CES-IPNL204, Feedwater Heater Auxiliary Control Panel, on Turbine Bldg Elev 306.

Water is supplied to each reboiler from the Condensate System through air operated level control valve, 2TME-LV13A(B). 2TME-LV13A(B) modulates to control reboiler level at 11.75" above tank centerline. The level control valve will auto close when the associated reboiler level is high-high (8" above normal). The reboiler steam outlet valves, 2TME-MOV16A(B), also auto close when the associated reboiler level is high-high.

Gland seal steam is normally supplied from the Clean Steam Reboiler through motor operated valve, 2TME-MOV105. Gland seal pressure is normally maintained at approximately 7 psig by pressure control valve, 2TME-PCV114. Motor operated valve, 2TME-MOV106, can be used to bypass 2TME-MOV105 and PCV114 during system startup.

If the Clean Steam Reboiler is unable to supply gland seal steam, emergency sealing steam is supplied from Main Steam through air operated isolation valve, 2TME-AOV121. 2TME-AOV121 will auto open when steam supply pressure from the reboilers is less than 20 psig or gland seal steam regulated pressure is approximately 6.3 psig. The emergency sealing steam pressure is reduced from Main Steam pressure by air operated pressure control valve, 2TME-PV122, which modulates to maintain emergency sealing steam pressure at approximately 6.1 psig. The emergency sealing steam is then admitted to the gland seal steam header through normally open 2TME-MOV110. 2TME-PV122 starts to open to admit steam after 2TME-AOV121 has opened on low reboiler steam supply pressure of less than 20 psig or a low regulated gland seal pressure of 6.3 psig.

The two Steam Packing Exhausters (SPEs) exhaust the steam/air mixture from the turbine seals. Each Steam Packing Exhauster consists of one Gland Exhaust Cooler, 2TME-E2A(B), and two Gland Exhaust Fans, 2TME-FN1A/1B (2A/2B). 2TME-E2A(B) condenses the steam in the steam/air mixture and is cooled by the Condensate System flowing through the tube side. 2TME-FN1A/1B (2A/2B) maintain a vacuum in their respective Steam Packing Exhauster and associated piping. The SPEs are located on Turbine Building Elev 277 Northwest corner.

Normally, only one SPE is in service with one of the Gland Exhaust Fans running continuously. The fans are controlled by their respective control switches on 2CEC*PNL851 in the Control Room. Each fan is interlocked with its associated motor operated butterfly suction valve, which is required to be off its closed seat before the fan will start. The fans will trip on motor overload or if its associated suction valve is closed.

The air and non-condensable gases from the Gland Exhaust Fans are discharged into a header which is directed to the Main Stack via the Condenser Air Removal System hogging discharge header.

After the steam is condensed in the Gland Exhaust Cooler shell side, it is directed to the Gland Exhaust Cooler Drain Receiver via a loop seal. Level is maintained in the drain receiver by air operated valve, 2TME-AOV137, which directs the water back to the Main Condenser. 2TME-AOV137 modulates to control 2TME-TK1 at 18" (17" to 19").

C. OPERATING REQUIREMENTS

C.1. The following systems must be in operation to support the ASS, TME, and CNA Systems:

- N2-OP-3, Condensate and Feedwater System
- N2-OP-19, Instrument and Service Air System
- N2-OP-48, Auxiliary Boiler System
- N2-OP-64, Turbine Building Drains
- N2-OP-71C, 600V A.C. Power Distribution

D. PRECAUTIONS AND LIMITATIONS

D.1. Prior to admitting Gland Seal Steam, the Main Turbine should be on the Turning Gear to prevent rotor bowing. Operations Management must give permission to **not** be on the Turning Gear when placing Gland Seals in service. See Precautions and Limitations of N2-OP-21 for actions when steam seals are applied without the turning gear in operation.

D.2. Prior to securing Gland Seal Steam, Main Condenser vacuum must be broken to prevent drawing "cold" air across turbine seals and rotor shaft unless an engineering evaluation has determined that “cold" air can be drawn across the turbine seals and rotor shaft.

D.3. As turbine shaft seals are worn, a higher gland exhaust vacuum may be required to prevent steam from blowing out the turbine seals.

D.4. Both reboilers may not be in service at the same time due to the interlock in the 2CNA-LV8A,B,C control logic which locks up the level control valves "As Is" when both 2ASS-MOV3A and MOV3B are open at the same time.

D.5. DCP N2-05-021 AND ECP-14-000923 has isolated the Clean Steam Reboiler steam supply to Radwaste via two spectacle flanges installed in line 2-TME-010-38-4 in the Turbine Building. If it is desired to supply steam to Radwaste, the spectacle flange will need to be shifted. ECP-11-000977 has gagged 2TME-RV135 contingent on the N2-05-021 spectacle flange installation. The gag must be removed should line 2-TME-010-38-4 be placed back in service.

D.6. During Startup of the Sealing Steam System water leakage from Main Turbine Stop, Control, AND bypass valve can be expected. This is due to the cooler metal in the valve blocks condensing the sealing steam then the condensation leaks out of the valve packing gland. This leakage is expected UNTIL the valves are completely warmed up.

F. NORMAL OPERATIONS (Continued) **Initials**

F.4.13. **WHEN** conditions are stable,
THEN SECURE operator at Auxiliary Boiler(s). OPE

NOTE

Actions in this Subsection are performed at 2CEC*PNL851 unless otherwise specified.

F.5. **Shifting Reboiler Steam Supply from Auxiliary Boiler Steam to Extraction/Main Steam**

F.5.1 **IF** Reactor startup is in progress,
THEN VERIFY Reactor pressure is approximately 925 psig
AND one Turbine Bypass Valve is greater than or equal to 20% open. OPE

F.5.2 **VERIFY** an operator is stationed at Auxiliary Boilers. OPE

NOTE

- If 2ASS-MOV148 is **not** already open, then the valve should be opened manually to preclude a power excursion.
- 2ASS-MOV148 is located in the TB Elev 277 Condenser Area by the Bypass valves on elevation 288.

F.5.3 **IF** 2ASS-MOV148, MAIN STM TO AUX STM ISOL VLV, is **not** open,
THEN MANUALLY OPEN 2ASS-MOV148 as follows,
OTHERWISE MARK this step N/A.

a. **DE-ENERGIZE** 2ASS-MOV148 by placing breaker 2NHS-MCC010-21A, MAIN STEAM BLOCK VALVE, in OFF. OPE

b. **SLOWLY OPEN** 2ASS-MOV148. OPE

c. **WHEN** valve is open,
THEN RE-ENERGIZE 2ASS-MOV148 by placing breaker 2NHS-MCC010-21A in ON. OPE

F.5.4 **OPEN** 2ASS-V23. OPE

F.5.5. **CLOSE** 2ASS-AOV145, AUX BLR STM INLET VLV, by placing control switch in CLOSE.

F. NORMAL OPERATIONS (Continued) **Initials**

NOTE

- 2ASS-STV112 will auto open when 4th Point ESS pressure is less than 50 psig. It will auto close when 4th Point ESS pressure is greater than 57 psig or Main Steam Inlet Header pressure is less than 200 psig.
- 2ESS-STV104 will auto open when 4th Point ESS pressure is greater than 57 psig. It will auto-close when 4th Point ESS pressure is less than 50 psig.

F.5.6. **PLACE** the following control switches in AUTO
AND VERIFY one valve opens **AND** the other valve remains closed as applicable:

- 2ASS-STV112, REBOILERS 1A & 1B MAIN STM TRIP VLV.....☐
- 2ESS-STV104, REBOILERS 1A & 1B EXTR SUPPLY ISOL VLV.....☐ _____

NOTE

2CES-IPNL204 is located on TB Elev. 306 North.

F.5.7. At 2CES-IPNL204, **MOMENTARILY DEPRESS** the following black RESET pushbuttons:

- 2CNA-LV8A, REBOILER DR TK NORM WTR LVL DR .....☐
- 2CNA-LV8B, REBOILER DR TK NORM WTR LVL DR .....☐
- 2CNA-LV8C, REBOILER DR TK NORM WTR LVL DR.....☐ _____

F.5.8. **OPEN** 2CNA-HV34A(B), REBOILER 1A(B) DRAIN TK 1A(B) ISOL VLV. _____

F.5.9. Using indications on 2CES-IPNL204, **CONFIRM** the following.

- 2CNA-LV8A, LV8B, **AND** LV8C are modulating to control Reboiler Drain Tank level.....☐
- 2CNA-LV9A **AND** LV9B are closed.....☐ _____

F.5.10. **MONITOR** for proper reboiler operation per Subsection F.1. _____

F.5.11. **IF** required **AND** using N2-OP-48,
THEN secure Auxiliary Boiler(s),
OTHERWISE MARK this step N/A. _____

F.5.12. **SECURE** operator at Auxiliary Boiler. _____

F. NORMAL OPERATIONS

NOTE

Actions in this Subsection are performed at 2CEC*PNL851 unless otherwise specified.

F.1. Normal Operating Indications

F.1.1. **IF** in service,
THEN Clean Steam Reboiler 1A(B) normal operating conditions are as follows.

- Reboiler steam supply pressure is approximately 95 psig as indicated on one of the following:
  - 2ASS-PI116, CLN STM REBLR STM PRESS SUPLY pressure meter ..... ☐
  - Computer Point ASSPA03, CLN STM RCBLR STM PRESS ..... ☐ _____
- Reboiler discharge steam pressure is approximately 55 to 90 psig as indicated on one of the following:
  - 2TME-PI15A(B), REBOILER 1A(B) DISCH STM PRESS meter... ☐
  - Computer Point TMEPA03 (TMEPA04), REBLR 2TME-E1A(B) DIS ST PR ..... ☐ _____
- Reboiler shell level is approximately 7 to 8 inches as indicated on 2TME-LI13A(B), REBOILER 1A(B) SHELL LEVEL meter. _____
- Reboiler Drain Tank 1A(B) level is approximately 16 to 18 inches as indicated on 2CNA-LI9A(B), REBLR DRN TK1A(B) LEVEL meter. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.1.2. Gland Seal normal operating conditions are as follows:

- **IF** Reboiler is in service,
  THEN Gland Seal supply pressure is approximately 55 to 90 psig as indicated on one of the following.
  - 2TME-PI103, GLAND SEAL STM SUPPLY pressure meter..... ☐
  - Computer Point TMEPA01, GLAND SL STM SUPLY PRESS..... ☐ _____
- Gland Seal regulated supply pressure is approximately 7 to 7.5 psig with Reboiler supplying sealing steam,
  OR approximately 6 psig **WHEN** on Main Steam, as indicated on one of the following.
  - 2TME-PI107, GLAND SEAL STM REGLTD SPLY PRESS meter ☐
  - Computer Point TMEPA02, GLD SL STM RGLDT SPLY PR ..... ☐ _____

NOTE

Gland exhaust vacuum should normally be maintained approximately 10 to 14 inches H2O. Due to degraded packing glands, adjustments may be made to gland exhaust vacuum up to 20 inches H2O in order to dampen fan oscillations or maintain steam seal.

- Gland Exhaust vacuum is approximately 10 to 20 Inches H_2O Vacuum as indicated on one of the following.
  - 2TME-PI108, STEAM SEAL EXH VACUUM gauge..... ☐
  - Computer Point TMEPA05, STM SEAL EXH VACUUM ..... ☐ _____

NOTE

Steam blowing from turbine shaft seals indicates a higher Gland Exhaust vacuum is required as referenced in Subsection H.1.

- At Main Turbine, **no** steam is blowing from turbine shaft seals. _____

NOTE

The loop seal is located by the Steam Packing Exhausters on TB Elev 277 West.

- A water level exists in the Gland Exhaust loop seal as indicated by air **not** being sucked in. _____

F. NORMAL OPERATIONS (Continued) **Initials**

NOTE

Actions in this Subsection are performed at 2CEC*PNL851 unless otherwise specified.

F.2. **Shifting Gland Seal Supply from Reboiler Steam to Main Steam**

F.2.1. **VERIFY** Main Steam is available. _____

F.2.2. **USING** 2TME-PI107, GLAND SEAL STM REGLTD SPLY PRESS meter,
MONITOR Gland Seal pressure. _____

NOTE

- 2TME-MOV110 is throttleable.
- Annunciator 851232, GLAND SEAL AUX STM SPLY TME-AOV121 OPEN, may alarm during the next step.

F.2.3. **VERIFY** the following open:

- 2TME-MOV110, GLAND SEAL STM MN STM REGLTD SPLY VLV.. ☐
- 2TME-AOV121, GLAND SEAL STM MAIN STM SPLY VLV ..... ☐ _____

F.2.4. **THROTTLE OPEN** 2TME-MOV106,
REBOILER GLAND SEAL STM BYP V,
UNTIL Gland Seal Pressure as indicated on 2TME-PI107 rises slightly. _____

F.2.5. **WHEN** Gland Seal supply pressure has risen slightly **AND** stabilized,
THEN CLOSE 2TME-MOV105, REBOILER GLAND SEAL STM SPLY V. _____

F.2.6. **WHILE** 2TME-MOV105 is closing, **THROTTLE** 2TME-MOV106 as needed to maintain Gland Seal pressure 4.0 to 7.5 psig as indicated on 2TME-PI107. _____

F.2.7. **WHEN** 2TME-MOV105 is closed,
THEN SLOWLY THROTTLE 2TME-MOV106 full closed. _____

F.2.8. **CONFIRM** Gland Seal pressure on 2TME-PI107 indicates 4.0 to 7.5 psig. _____

F.2.9. **IF** Gland Seal pressure on 2TME-PI107 is **not** 4.0 to 7.5 psig,
THEN TAKE manual control of 2TME-PIC122 per H.4, Manual Control of Main Steam Supply to Gland Seal Steam,
OTHERWISE MARK this step N/A. _____

Simulator Scenario Handout Reference Packet

Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-96

REACTOR MANUAL CONTROL AND ROD POSITION INDICATION SYSTEM

Revision 01100

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-21-02929:

- Description: Add steps in section H.2 to help identify and correct the loss of display for one 4-rod group on the 4-rod display.
- Reason: AR 04446364-05, Loss of Position Indication for 4 Control Rods
- H.2.1 preceding note, added that performance of H.5 Alternate Rod Position Indication Upon Loss of Indication [C2] [C5] is an acceptable method to satisfy the 24-hour surveillance requirement for Control Rod Position Indication required by TSSR 3.1.3.1 (reference IR 04446364-04).
- H.2.6 added new steps to replace a faulty rod display MUX card.

Procedure Writer Changes:

- Updated fonts to bold and lowercase underline for **not**, **cannot** and **no**, bold and uppercase for **ACTION VERBS** and **LOGIC WORDS** per AD-NM-101-1005 and AD-AA-101-1002, **no** revision bars used.
- Removed “open” or “close” from six steps that require throttling 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV, as personnel are trained for operation of this pressure control valve, and “throttle” is the correct action verb to describe it’s intended use. Often, specifying only “open” or “close” is insufficient direction as the throttle valve requires manipulation in both directions to accomplish the desired and specified differential pressure changes, **no** revision bars used. Changes made in the following steps:
  - F.2.1.e and F.2.1.f
  - F.3.1.f and F.3.1.g
  - F.3.2.g and F.3.2.h
- G SHUTDOWN reformatted previous revision paragraph into a note and step.
- For the upcoming transition to Exelon Smart Procedures (ESP), which does **not** display traditional MS Word tables well, and also in which the display of graphics are limited (graphics are also available by simply looking at the panel, are in training material, and are in the RMCS Vendor Manual, and may need to be deleted in the next revision for final conversion to ESP), moved the following previous revision attachments and figures:
  - Previous revision ATTACHMENT 1: ROD BLOCKS WITHDRAWAL BLOCKS, is now B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks, **no** revision bars used.
  - Previous revision FIGURE 1: REACTOR MANUAL CONTROL SYSTEM, is now B.7.1 Reactor Manual Control System, **no** revision bars used.
  - Previous revision FIGURE 2: ANALYZER PAGE INDICATORS, is now B.7.2 Analyzer Page Indicators, **no** revision bars used.
  - Previous revision FIGURE 3: RPIS DISPLAY CARD INDICATIONS, is now B.7.3 RPIS Display Card Indications, **no** revision bars used.
  - Previous revision Attachment 2, ROD POSITION VERIFICATION UPON LOSS OF POWER table information incorporated into written steps of section H.5 Alternate Rod Position Indication Upon Loss of Indication [C2] [C5], **no** revision bars used.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 4
B. SYSTEM DESCRIPTION 6
C. OPERATING REQUIREMENTS 23
D. PRECAUTIONS AND LIMITATIONS 24
E. STARTUP 25
F. NORMAL OPERATIONS 26
F.1. Control Rod Selection 26
F.2. Control Rod Withdrawal 27
F.3. Control Rod Insertion 32
G. SHUTDOWN 35
H. OFF-NORMAL PROCEDURE 36
H.1. Mispositioned Control Rod(s) [SOP] [C1] 36
H.2. Loss of Rod Position Indication or Control Rod Display 38
H.3. Rod Drift Alarm Test 41
H.4. Bypassing a Rod in Rod Drive Control 42
H.5. Alternate Rod Position Indication Upon Loss of Indication [C2] [C5] 45
H.6. Inoperable RDCS [C4] 62
H.7. Preventing Control Rod Withdraw Double Notching 63
H.8. Preventing Control Rod Insert Double Notching 66
H.9. Manually Inserting and Removing a Rod Block Monitor Rod Block 69

A. REFERENCES AND COMMITMENTS

A.1. Technical Specifications

- 3.1.5, Control Rod Scram Accumulators
- 3.9.5, Control Rod OPERABILITY - Refueling
- 3.1.3, Control Rod OPERABILITY
- 3.9.4, Control Rod Position Indication
- 3.3.2.1, Control Rod Block Instrumentation
- 3.9.2, Refuel Position One-Rod Out Interlock
- 3.1.6, Rod Pattern Control

A.2. Licensee Documentation

- USAR Section 7.7.1.1, Reactor Manual Control System
- USAR Section 15.4.1.2, Continuous Rod Withdrawal During Reactor Startup

A.3. Technical Information

A.3.1. Drawings

- 12177-PID-30A, Control Rod Drive Hydraulic System
- 12177-PID-30B, Control Rod Drive Hydraulic System
- 12177-PID-30C, Control Rod Drive Hydraulic System

A.3.2. G. E. Drawings

- 791E406TY, RMCS Elementary Diagram
- 761E354, CRD Hydraulic System Functional Control Diagram
- 828E303AA, Rod and Detector Display Elementary Diagram
- 828E426AA, Rod Position Information System Schematic Diagram
- GEK 83320B, Figure 4-2, Analyzer Page Indicators

A.3.3. Instruction Manuals

- N20287 - GEK 83320B Reactor Manual Control System
- N20269 - GEK 83317 Control Rod Drive

A.3.4. General Electric (GE) Service Information Letters (SILs)

- G.E. SIL 174, Improved Rod Select Push Buttons
- G.E. SIL 407 Preventing Unauthorized Control Rod Patterns

A.3.5. Others

LER 86-10

A.4. **Commitments**

- **[C1]**, SOER 84-02, Recommendation 1, Prompt notification of Nuclear Engineering personnel and General Supervisor Operations prior to mispositioned control rod recovery.
- **[C2]**, 502882-06, Add an Alternate Method to Determine Control Rod Position.
- **[C3]**, SOER 84-02, Recommendation 6, Provide guidance for continuous withdrawal and insertion using continuous insert pushbutton or notch override switch.
- **[C4]**, IR 02062325, Reactor Manual Control System Lockup and RBM operability Tech Spec compliance.
- **[C5]**, IR 02095342, RPIS Failure - Loss of 42 Control Rod Position Indications

B. SYSTEM DESCRIPTION

B.1. Reactor Manual Control System (RMCS)

Composed of four subsystems which allow the operator to select and move control rods as needed for efficient fuel management, and to control core power and flux distribution.

The Rod Drive Control System (RDCS) accomplishes all control rod movements except for a scram. The RDCS monitors a variety of conditions pertaining to the control rods and gives individual indications and summaries of those conditions. The RDCS receives inputs from various plant systems and will prevent rod withdrawals by enforcing rod blocks if selected plant parameters exceed their trip setpoints. See B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks for a list of rod blocks.

The Rod Position Indication System (RPIS) provides information on the vertical position and temperature of the control rod drives.

The Rod Worth Minimizer (RWM) functions to limit control rod worths to minimize the effect of a control rod accident or rod withdrawal errors by enforcing adherence to specified control rod patterns through the use of control rod blocks. The operation of the RWM is covered by N2-OP-95A.

B.2. Rod Drive Control System (RDCS)

The RDCS consists of the controls indicators and circuitry necessary for normal insertion and withdrawal of control rods, one at a time. This system includes interlocks to prevent normal rod motion when rod block signals are present. The RDCS will **not** cause or prevent a scram, which is controlled by the Reactor Protection System N2-OP-97.

The RDCS operates in three modes: 1) The operator control mode, 2) The scan mode, and 3) The Self-Test mode. The Operator Control mode is used to move control rods. The Scan mode is used to gather status data for a rod. The Self-Test mode is used to test the functions of the transponder cards and the related command circuitry. The system spends approximately 100 microseconds in a mode, and alternates between the Operator Control mode and the Scan mode until the status data is gathered for all rods. Each time the scan mode is entered, a different rod is scanned according to a prescribed sequence. When all rods have been scanned, the Self-Test mode will replace the Scan mode for one alternation to test one transponder card. Each time the system enters the Self-Test mode, the next rod in the sequence is scanned so that the circuitry for all rods is tested.

The RDCS is comprised of the following major components:

- Rod Select Module (RSM)
- Rod Drive Control Cabinet (RDCC)
- Branch Junction Modules (BJMs)
- HCU Transponders
- Display Memory Module (DMM)
- Rod and Detector Display (R&DD)

B. SYSTEM DESCRIPTION (Continued)

B.2.1. Rod Select Module

The Rod Select Module is located on the sloping portion of the Operator's console on 2CEC*PNL603. It contains a push button array for selecting:

- The control rod to be moved
- Stabilizer valve set (A or B)
- Drift control (Test, Reset)
- Rod movement commands
- Accumulator Trouble Acknowledge

It also contains two identical transmitter cards, that independently sense which buttons have been pressed by the operator. The transmitter cards then generate duplicate binary REQUEST words which are transmitted to the activity control cards in the Rod Drive Control Cabinet. The RSM also receives inputs from the Rod Sequence Control System and the Rod Worth Minimizer.

B.2.2. Rod Drive Control Cabinet

The Rod Drive Control Cabinet is the heart of the system and is located in panel P616. It contains four major sections. At the top of the cabinet are two identical Activity Control sections. Below this is an Analyzer section, and the Power Supply Section is in the bottom of the cabinet.

The two Activity Control sections are identical and independent. Each section consists of an Input Isolator card, an Activity Control card, and a Rod Motion Timer Card. Each Activity Control section receives a REQUEST word from the Rod Select Module. The REQUEST word, inputs from the rest of the system, plus the rod block signals from other systems enter the Activity Control section through the Input Isolator card, which electrically isolates the signals coming from outside the cabinet. The main function of the Activity Control Card is to monitor the status of the plant to determine if rod motion requested by the operator is permissible. If the request for rod movement is permitted (i.e.; **no** Rod Blocks), the Activity Control Cards each generate a COMMAND word and initiates operation of the respective Rod Motion Timer card. The Rod Motion Timer cards produce a 10 second timing signal to move a control rod one notch.

B. SYSTEM DESCRIPTION (Continued)

The Analyzer section contains an Analyzer card, a Fault Map card, and isolator cards. The Analyzer section is designed to detect errors in control signals that may occur due to electrical noise, equipment faults, etc.; and to prevent inadvertent rod motion due to these errors. The Analyzer section compares the COMMAND words from the Activity Control Sections for equivalency. If they are the same, one of the COMMAND words is transmitted to the transponders through the Branch Junction Modules, and the other is retained in the Analyzer as the REFERENCE word. The REFERENCE word is compared to the ACKNOWLEDGE word (returned to the cabinet from the Transponder of the selected rod) for errors. If a sufficient number of errors occur, the location of the fault (the BJM or the Transponder) will be indicated on the Fault Map card and system operation will be interrupted. The Analyzer section includes a number of controls and display registers which are used to verify proper system operation and to assist IMD personnel in troubleshooting the system. The Display registers hold all of the data available at the time the decision was made to stop system operation.

The Fault Map card contains an array of LEDs laid out in a mimic of the core and is used to indicate faults with the Transponders. The Fault Map card also contains counters that are used to control the Scan mode and the Self-Test mode.

The power section of the RDCC contains the main circuit breakers for the system, the Power Gate and the Contact Power Supply. The Power Gate consists of two large silicon-controlled rectifiers (SCRs) and controls the 120 VAC power to the transponders. If the Analyzer section encounters a problem in the system, it interrupts the control signal to the SCRs, turning them off.

This will disconnect all power to the HCUs and will prevent a failure in the system from causing inadvertent rod motion. The Contact Power supply furnishes power to all external contacts for the system.

B.2.3. **Branch Junction Module**

The Branch Junction Module is a enclosure mounted at the end of a cluster of HCUs. A BJM serves as a termination point for cables from the control room, adjoining BJMs, and for the Transponders on its group of HCUs. The BJM receives the COMMAND word, amplifies and shapes the signal, and transmits it to its associated cluster of transponders, and to the downstream BJM (if applicable). It also receives the ACKNOWLEDGE words from transponders within its cluster and downstream BJMs, and transmits it upstream to the next BJM or the RDCC.

B.2.4. **HCU Transponders**

The Transponders compare the rod address within the COMMAND word (transmitted from the RDCC) to their own. The Transponder that matches (i.e.; that rod has been selected for movement), will process the COMMAND sequence generated by the RDCC. The selected Transponder immediately returns an ACKNOWLEDGE word back to the Analyzer Card in the RDCC. The ACKNOWLEDGE word includes information on the present status of: the solenoid valves, scram valve position, accumulator pressure and liquid level, and the scram test switches. The selected Transponder continues to process the COMMAND work to open and close the Directional control Valves in the specified sequence to move the control rod. The Transponders also provide amplification to signals being passed through it to and from other Transponders.

B. SYSTEM DESCRIPTION (Continued)

B.2.5. Display Memory Module (DMM)

The Display Memory Module receives information from the RPIS and RDCS, stores the information from the RPIS and the RDCS, stores the information in its memory, and prepares the information for display on the Rod and Detector Display (R&DD) and the Four Rod display.

B.2.6. Rod and Detector Display (R&DD)

The Rod and Detector Display, or "full core display", is mounted in the vertical section of 2CEC*PNL603. It is an array of back-lit indicator elements arranged in a mimic of the core. Six parameters are displayed for every control rod:

- FULL IN: Lit when the rod is fully inserted.
- FULL OUT: Lit when the rod is fully withdrawn.
- DRIFT: Lit if the rod is moving and should **not** be.
- SELECTED: Specific rod's push button is lit when the rod is selected on the RSM.
- ACCUM: Lit if the accumulator pressure or water level is abnormal in the HCU.
- SCRAM: Lit if both scram valves on the HCU have opened.

The display also provides indication of Upscale and Downscale alarms for the LPRMs in each location (from the Neutron Monitoring System).

In addition to information given by the full core display, the 4-rod display presents the vertical position of a selected rod and the other rods in it's group, and LPRM readings for the LPRMs located next to the group. The rod positions displayed will be in the form of a two-digit number with an illuminated background. Even notch positions are indicated by numbers and Odd notch positions are indicated by dashes. If RPIS is receiving abnormal data (e.g.; a reed switch stuck open), it will be indicated by a double X in the display. If the selected group has less than four rods, the display window(s) will be blank for the locations that do **not** have rods.

B.2.7. Operator Control Mode

In the operator control mode, the operator initiates movement of a control rod by momentarily depressing the rod select push button on the RSM to select the rod to be moved. Then the operator pushes the INSERT or WITHDRAW push button as desired. If there are **no** rod blocks present, the RDCC initiates the sequence to move the rod in the desired direction.

To insert the selected rod, the RDCC sends a signal to the selected rod's HCU to open the insert supply and exhaust valves to drive the rod inward. After about 3 seconds, during which the rod moves in slightly more than one notch (6 inches), the insert valves close, and the settle valve opens for about 5 seconds to allow the rod to settle back down to the next notch position. Depressing the INSERT push button continuously causes a continuous insertion of the selected control rod. A control rod can also be continuously inserted by depressing and holding the continuous insert push button. This bypasses the rod motion timer cycle and the rod settle feature does **not** operate in this mode.

B. SYSTEM DESCRIPTION (Continued)

To withdraw the selected rod, the RDCC sends a signal to the selected rod's HCU to open the insert supply and exhaust valves for 0.6 seconds to insert the rod far enough to free the collet fingers from the detent and hold them open. After the insert valves shut, the withdraw valves open to drive the rod out. The withdraw supply valve closes after 1.5 seconds and the withdraw exhaust (settle) valve remains open for an additional 6 seconds to allow the rod to "settle" down to the next notch. Continuous withdrawal of a control rod is achieved by simultaneously depressing the CONTINUOUS WITHDRAW and the WITHDRAW push buttons. The dual push button requirement prevents an inadvertent continuous withdrawal of control rods.

B.2.8. **Scan Mode**

In the scan mode, the Analyzer card initiates commands to gather status data from a single HCU. Accumulator pressure and water level, scram valve position, and scram test switch position are checked. The Fault Map card generates a rod I.D. and sends it via the Analyzer card to each of the Activity Control cards.

Each Activity Control card forms a COMMAND word (although in this case, **no** rod motion is called for) which is transmitted in the same manner as in the Operator Control mode. The selected Transponder transmits the HCU status data back to the RDCC. The system does **not** verify correctness of the reply because the ability of the Transponders to reply correctly is checked in the Self-Test mode and, besides there is **no** reference available for the status data. At the next alternation with the Operator Control mode, a counter advances and causes the Fault Map card to generate the next rod I.D., which is processed as above.

B.2.9. **Self-Test Mode**

This mode is used for diagnosis and evaluation. As in the Scan Mode, the Fault Map card generates the rod I.D., but this time it also generates a specific valve control signal determined by a built-in program in the Analyzer. It sends this information to each Activity Control, where a COMMAND word is formed and processed exactly as in the other modes. The Analyzer program briefly exercises the valve driver circuit on the HCU; **not** long enough for mechanical valve motion, but long enough to verify electrical operation by means of a drive monitoring circuit located on the Transponder. The ACKNOWLEDGE word returned to the Analyzer is checked for correct response; both that the correct Transponder replies as indicated by the I.D. and that the valve driver circuit response is correct. In the Self-Test evaluation, a comparison failure of more than a prescribed number of words results in an automatic shutdown of the RDCS and 603303, ROD DRIVE CONTROL SYSTEM INOPERABLE annunciator.

B. SYSTEM DESCRIPTION (Continued)

B.3. Interlocks

The Reactor Manual Control System (RMCS) inhibits movement or selection of control rods under various conditions. The rod blocks caused by inputs from other systems are listed in section B.4 Rod Withdraw Blocks and B.5 Rod Worth Minimizer Withdraw and Insert Rod Blocks.

The existing NUMAC APRMs have an adjustable rate of 0.52 to 0.78 for the slope of the flow biased curves. The single loop flow biased rod block slope **cannot** be set to the desired slope of .50 until a firmware update is completed for MELLLA+ at a later date. The single loop operation EPU APRM rod block setpoint formula is modified by ECP-10-000818 to use the slope of 0.52 and maintains a conservative rod block setpoint.

Rod select blocks are initiated by the RDCC to prevent the selection of a control rod under the following conditions:

- A control rod has already been selected for movement and the movement cycle is **not** complete.
- The rod selected key-lock switch on the Rod Select Module is **not** rotated fully clockwise.
- One rod is selected and withdrawn while in the Refuel mode.

RMCS system failures (i.e.; disagreement between the Activity Control sections, or an error detected by the Analyzer section) and performance of manual testing will impose rod blocks.

B.4. Rod Withdraw Blocks

B.4.1. SRM Downscale Alarm Rod Withdraw Block

a. Setpoint: 3 cps

b. Bypassed:

- IRMs Range 3 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed

B.4.2. SRM Upscale Rod Withdraw Block

a. Setpoint: 1×10^5 cps

b. Bypassed:

- Associated IRM on Range 8 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.3. SRM INOP Rod Withdraw Block

a. Setpoints:

- Module unplugged
- High Voltage Low
- SRM Select Switch **not** in Operate*

b. Bypassed:

- Associated IRM on Range 8 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM bypassed
- * INOP Inhibit toggle switch actuated

B.4.4. SRM Detector **not** Full In Rod Withdraw Block

a. Setpoint: < 100 cps **AND** Detectors **not** fully inserted

b. Bypassed:

- IRMs Range 3 **OR** above **OR** bypassed
- Mode switch in RUN
- SRM Bypassed

B.4.5. IRM Downscale Alarm Rod Withdraw Block

a. Setpoint: 5/125 of scale

b. Bypassed:

- IRM on Range 1
- Mode switch in RUN
- IRM bypassed

B.4.6. IRM Upscale Alarm Rod Withdraw Block

a. Setpoint: 108/125 of scale

b. Bypassed:

- Mode switch in RUN
- IRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.7. IRM Detector **not** Full In Rod Withdraw Block

a. Setpoint: IRM Detector **not** fully inserted

b. Bypassed:

- Mode switch in RUN
- IRM bypassed

B.4.8. IRM INOP Rod Withdraw Block

a. Setpoints:

- Module unplugged
- High Voltage Low
- IRM Drawer Selector Switch **not** in Operate*

b. Bypassed:

- Mode switch in RUN
- IRM bypassed
- *INOP inhibit toggle switch actuated

B.4.9. APRM Downscale Alarm Rod Withdraw Block

a. Setpoint: $\geq$ 4% Average Neutron Flux

b. Bypassed”

- Reactor Mode Switch **not** in RUN
- APRM bypassed

B.4.10. APRM Upscale Alarm Rod Withdraw Block

a. Setpoints:

- $\leq$ 12% Simulated Thermal Power
- $\leq$ 0.55 Wd + 48.5% (two loop operation)
- $\leq$ 0.52 (W-ΔW) + 41.1% (single loop operation) (clamp at 107.5%)

b. Bypassed:

- Reactor Mode Switch **not** in RUN
- APRM bypassed

B. SYSTEM DESCRIPTION (Continued)

B.4.11. APRM INOP Rod Withdraw Block

a. Setpoints:

- Any Self-Test fault detected
- Total LPRM inputs from any one axial level < 3

b. Bypassed:

- APRM bypassed

B.4.12. APRM Flow Upscale Alarm Rod Withdraw Block

a. Setpoint: ≤ 108% Flow

b. Bypassed:

- Associated APRM bypassed

B.4.13. RBM Downscale Alarm Rod Withdraw Block

a. Setpoint: Average RBM Flux for Rod selected LPRMs is less than 75.8% power

b. Bypassed:

- Reactor Mode Switch **not** in RUN
- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B.4.14. RBM Upscale Alarm Rod Withdraw Block

a. Setpoints:

1) Low: (≥ 26% & < 61% RTP) *124.2%

2) Intermediate: (≥ 61% & < 81% RTP) *< 119.2%

3) High: (≥ 81% RTP) *< 114.2%

* Local LPRM Power is Normalized to 100% upon rod selection

b. Bypassed:

- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B. SYSTEM DESCRIPTION (Continued)

B.4.15. RBM INOP Alarm Rod Withdraw Block

a. Setpoints:

- Critical Self-Test Fault detected
- RBM keylock switch in INOP
- < 1/2 required LPRM inputs
- > 1 rod selected
- Firmware/Software Watchdog Timer timed out
- Loss of input power

b. Bypassed:

- RBM Auto Bypass (< 26% Average Neutron Flux)
- RBM Manual Bypass

B.5. **Rod Worth Minimizer Withdraw and Insert Rod Blocks**

The following is a brief summary of the rod blocks imposed by the Rod Worth Minimizer. For more information, refer to N2-OP-95A.

The Rod Worth Minimizer (RWM) applies rod block and permissive signals to the RDCS which generate the rod block or permissive output signals. Rod blocks are applied when the RWM permissive requirements are **not** met and power is below the LPSP. A control rod in a group is in the permissive state when there are two or less insert errors and **no** withdraw errors.

Another RWM system interlock involves the Rod Test sequence. The Rod Test mode may only be entered if **no** more than one control rod is withdrawn from the fully inserted position. If a rod is being tested and is in any position other than notch 00, a rod withdrawal block will be initiated if a second rod is selected on the RSM. If more than one rod is withdrawn beyond the 00 position, and the Rod Test push button is depressed, both insert and withdraw blocks are immediately applied. The Rod Test push button must be depressed a second time to return the system to normal.

Once power is raised above the LPSP, **no** rod blocks are applied, but alarm indications and insert error displays are still provided. At power levels above the Low Power Alarm Point (LPAP), the monitoring functions of the RWM are automatically bypassed. System hardware failures will initiate rod blocks at any power level. Whenever a RWM program abort occurs, the RWM is automatically taken off-line, withdraw and insert blocks are applied and the RWM must be reinitialized before the program can be executed again.

B. SYSTEM DESCRIPTION (Continued)

B.5.1. RWM Withdrawal Rod Block

a. Setpoints:

- Any Withdraw Error
- 3 Existing Insert errors **AND** an Insert Block applied causes a block for any rod selected other than one causing an Insert error
- Rod Test:
  - One rod withdrawn **AND** another rod selected
  - One rod withdrawn **AND** Rod Test Push button is depressed
- RWM program abort.

b. Bypassed:

- Reactor Power above the LPSP (as sensed by steam flow)
- RWM Bypassed

B.5.2. RWM Insert Rod Block

a. Setpoints:

- Two Insert Errors existing **AND** third Insert Error is made
- A Withdraw error existing **AND** a Withdraw Block applied causes a block for any rod selected other than one causing the Withdraw error.
- When power reaches the LPSP while lowering Rx Power **AND**:
  - A Withdraw error exists
  - **OR** the Operating Sequence is **not** latched
- One rod withdrawn **AND** the Rod Test push button is depressed
- RWM program abort.

b. Bypassed:

- Reactor Power above the a LPSP (as sensed by steam flow)
- RWM Bypassed

B. SYSTEM DESCRIPTION (Continued)

B.6. Rod Position Indication System (RPIS)

The RPIS continuously monitors the vertical position and temperature of the control rod drives. The position information is transmitted to the Rod Drive Control System (RDCC) for display on the Rod and Detector Display (R&DD) and the 4-Rod Display, and to the process computer.

A thermocouple is attached to each control rod drive. The thermocouple sends a temperature signal to a chart recorder on panel 2CES*RAK007 (Rx Bldg. elev. 261', North) and energizes an annunciator on 2CEC*PNL603 if a high temperature condition is sensed.

The Position Indication Probes are the primary source of information for the system. Each control rod drive contains a Position Indicator Probe, which contains 53 reed switches connected in an eleven wire array. A closed reed switch will connect two of the eleven wires together and provides a unique output corresponding to the position of the rod. The drive piston in the CRD contains a magnet that closes the reed switch located nearest to the magnet. Forty-nine (49) switches are used to detect the position of the control rod. They are spaced at 3 inch intervals and are numbered from notch position 00 to 48. Notch 00 indicates that the rod is fully inserted into the core and notch 48 indicates that the rod is fully withdrawn. The even numbered switches correspond to the latch positions of the drive. The odd numbered switches correspond to intermediate positions between the notches on the drive. A second switch is located at the same location as the 00 position and is used to light the green FULL-IN indication on the full core display, R&DD. Another switch is located 1 1/4 inches further in than the notch 00 switch and keeps the green FULL-IN light energized when the rod is driven in past position 00 (e.g.; during a scram). Another switch is located at the same location as the notch 48 switch and is used to light the red FULL-OUT indication on the full core display. The last switch is located two inches further out than notch 48 and is used to detect an uncoupled rod. During normal rod motion, the rod is prevented from traveling past notch 48 by the control rod backseating on the stub tube. However, if the control rod drive is uncoupled from the blade, the drive is free to move out past notch 48 and will close the "overtravel" reed switch. This switch is used to initiate an annunciator on 2CEC*PNL603.

All of the wires from the probes terminate in the Rod Position Indication Cabinet (RPIC) located in 2CEC*PNL615 in the control room. The RPIC is the heart of the RPIS. It contains three modules, a Control module and two Multiplex modules. The Control module contains all of the electronics for controlling the system.

There are **no** operator controls or displays directly associated with the RPIS. RPIS supplies the information to the Display Memory Module, which drives the Rod and Detector Display and the 4-Rod Display.

Typical signal flow through the system is as follows: A counter in the RPI Cabinet advances automatically through all of the rod addresses. A multiplexed word, the IDENT word, is distributed to all of the Probe Multiplexer (MUX) cards in the system. These cards (each of which services a 4 rod group) compare the address contained in the IDENT word with their identity as defined by their Row and Column IDENT cards.

The card whose identity matches the address will respond with the position data from the corresponding control rod probe. This data is analyzed by the Probe Data Processor III Card in the RPIC.

B. SYSTEM DESCRIPTION (Continued)

The processor first determines if the data is acceptable. That is, the data must represent a realistic state for the control rod (e.g.; the rod **cannot** be in both positions 12 and 14 at the same time). If the data is **not** realistic, then some defect exists in the system.

If the data is acceptable, the processor will translate it from a pure binary format into a binary coded decimal format that is used to drive the position displays. This code and other data is appended to the IDENT code and transmitted to the Display Memory Module as the POSITION word. The position data is also loaded into a local memory which can be accessed by the plant computer. If the data is **not** acceptable, it is discarded and a fault code is generated. The fault code is transmitted instead, as part of the POSITION word.

After transmitting a POSITION word, the counter will advance to the next address and the process will be repeated. This cycle is repeated continuously, checking the status of all the rods. It takes about 40 milliseconds to cycle through all of the rods. The system does **not** stop if a fault is detected.

Operator inputs into the system are multiplexed into REQUEST words by the Rod Select Module in the RDCS. A REQUEST word is transmitted to the RPI Cabinet. From this word, the RPIC extracts:

- The IDENT of the selected rod.
- The Drift Test command (if applicable).
- The Drift Reset request (if applicable).

One of the bits in the POSITION word is used to identify the selected rod. When, in the process of scanning the probes, the processor comes across the rod whose IDENT is the same as the REQUEST word, it will set this bit into the POSITION word prior to transmitting it.

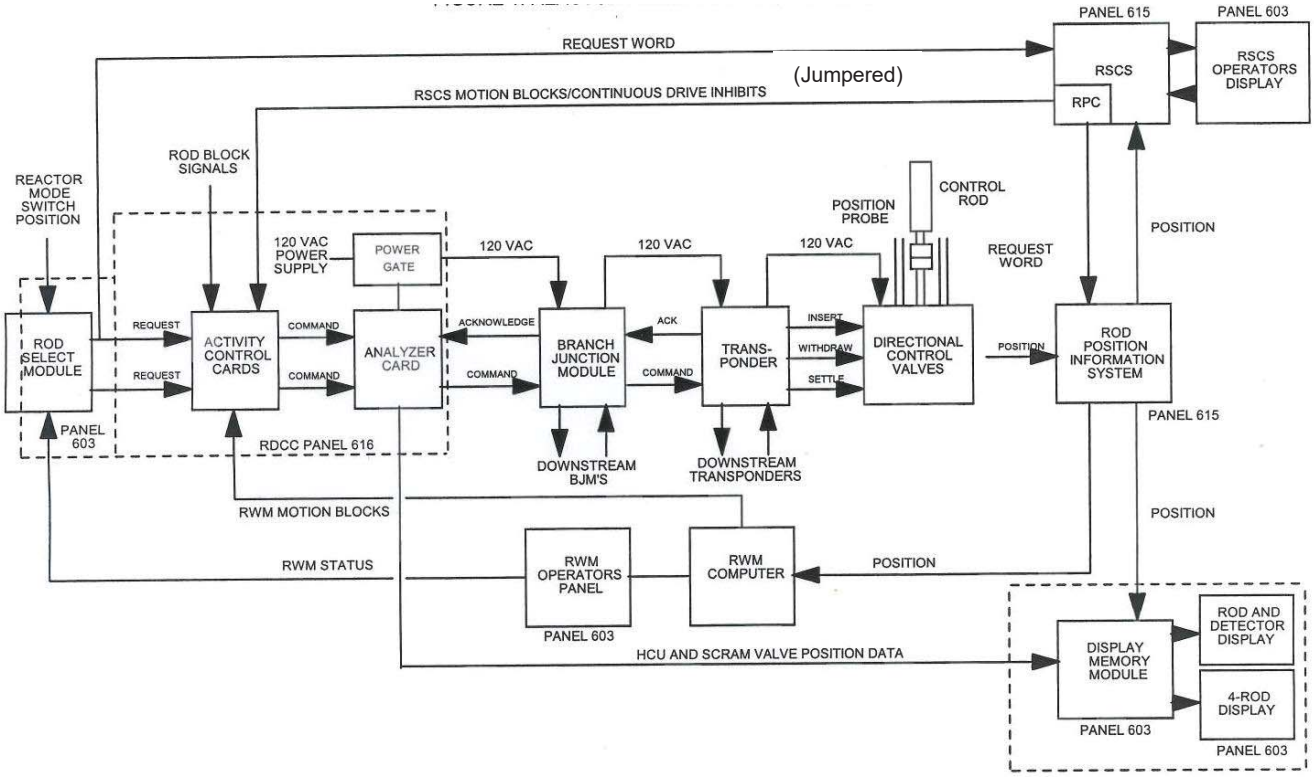
Unless a control rod is being moved, it should only be located at an even-numbered notch position. If a rod is located at an odd-numbered position, it is defined as Drifting (unless the operator is moving that rod). One bit in the POSITION word carries the drift information for that rod, and once set, remains even if the rod stops drifting (i.e.; latches into an even-numbered notch). A "Drift Reset" signal originated by the operator and carried by the REQUEST word, will cause all drift signals to reset unless the rod is still drifting. The "Drift Test" is used to check the drift detection function by temporarily overriding the "rod driving" signal (coming from RDCS) while the operator moves a rod through an odd-numbered position.

The 4-rod display presents the vertical position of a selected rod and the other rods in it's group. The rod positions are displayed in the form of a two digit number. Even notch positions are indicated by numbers and Odd notch positions are indicated by dashes. If RPIS is receiving abnormal data (e.g.; a reed switch stuck open), it will be indicated by a double X in the display. If the selected group has less than four rods, the display window(s) will be blank for the locations that do **not** have rods.

B. SYSTEM DESCRIPTION (Continued)

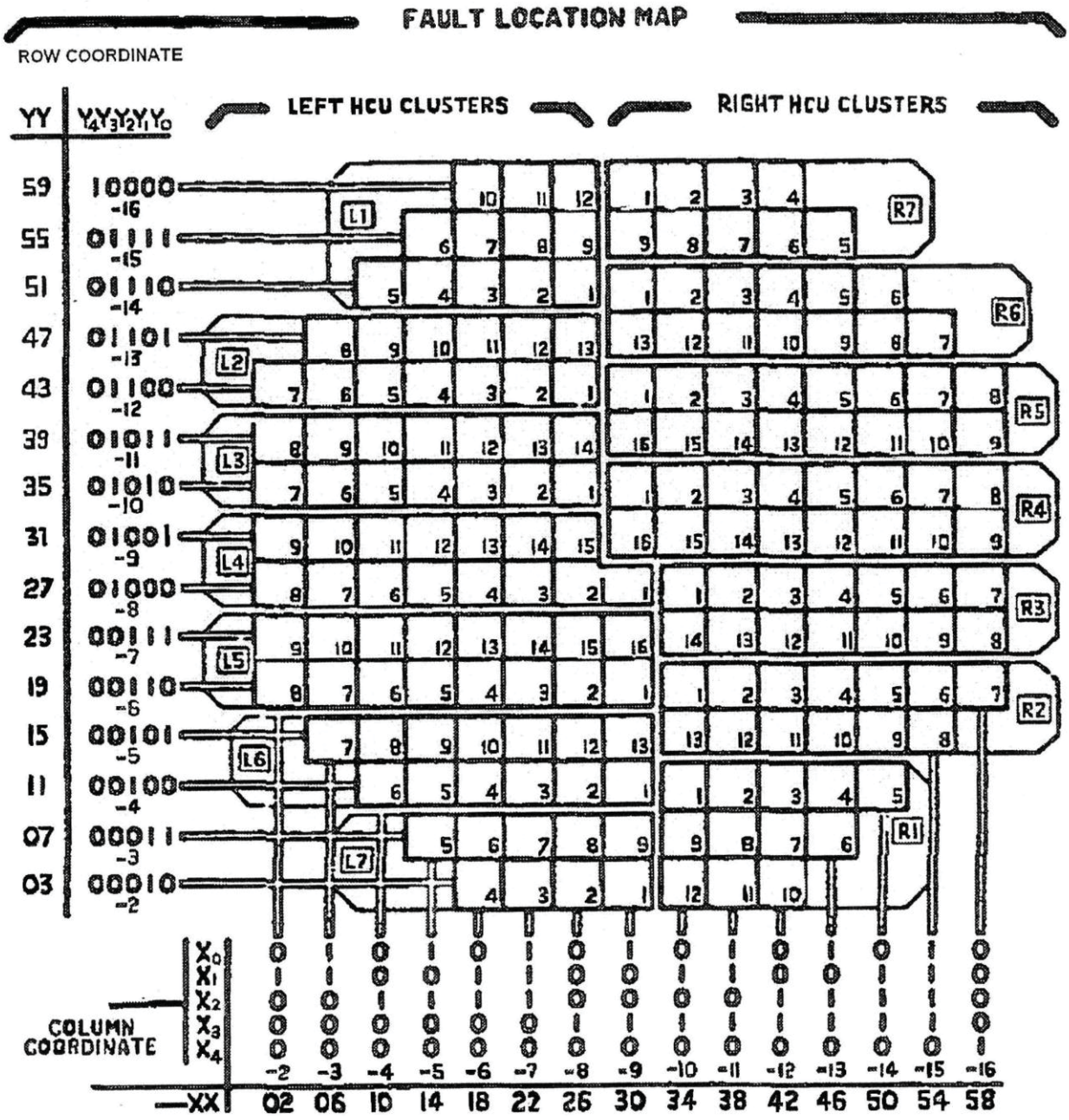
B.7. System Figures and Panel Depictions

B.7.1. Reactor Manual Control System



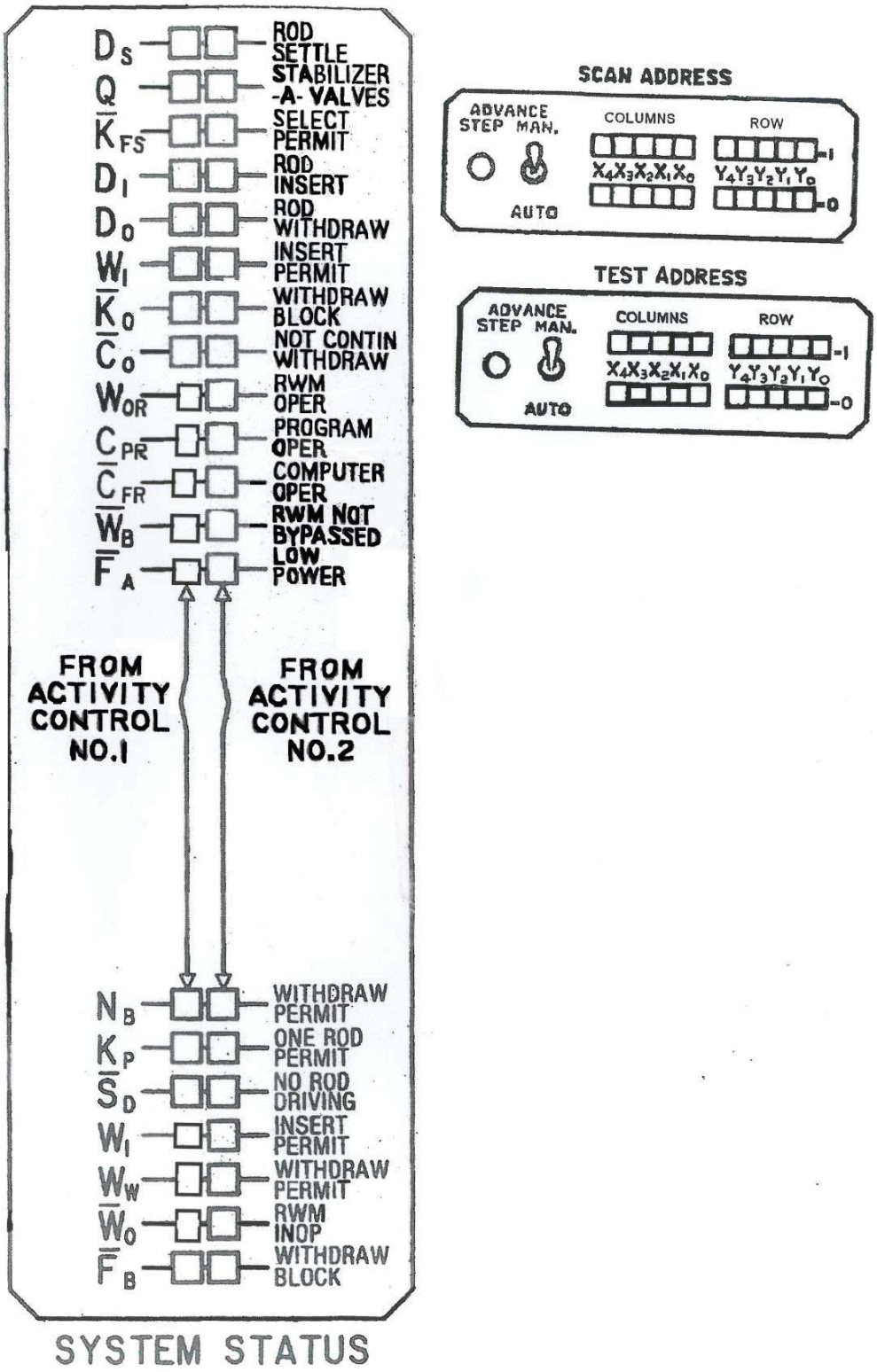
B. SYSTEM DESCRIPTION (Continued)

B.7.2. Analyzer Page Indicators



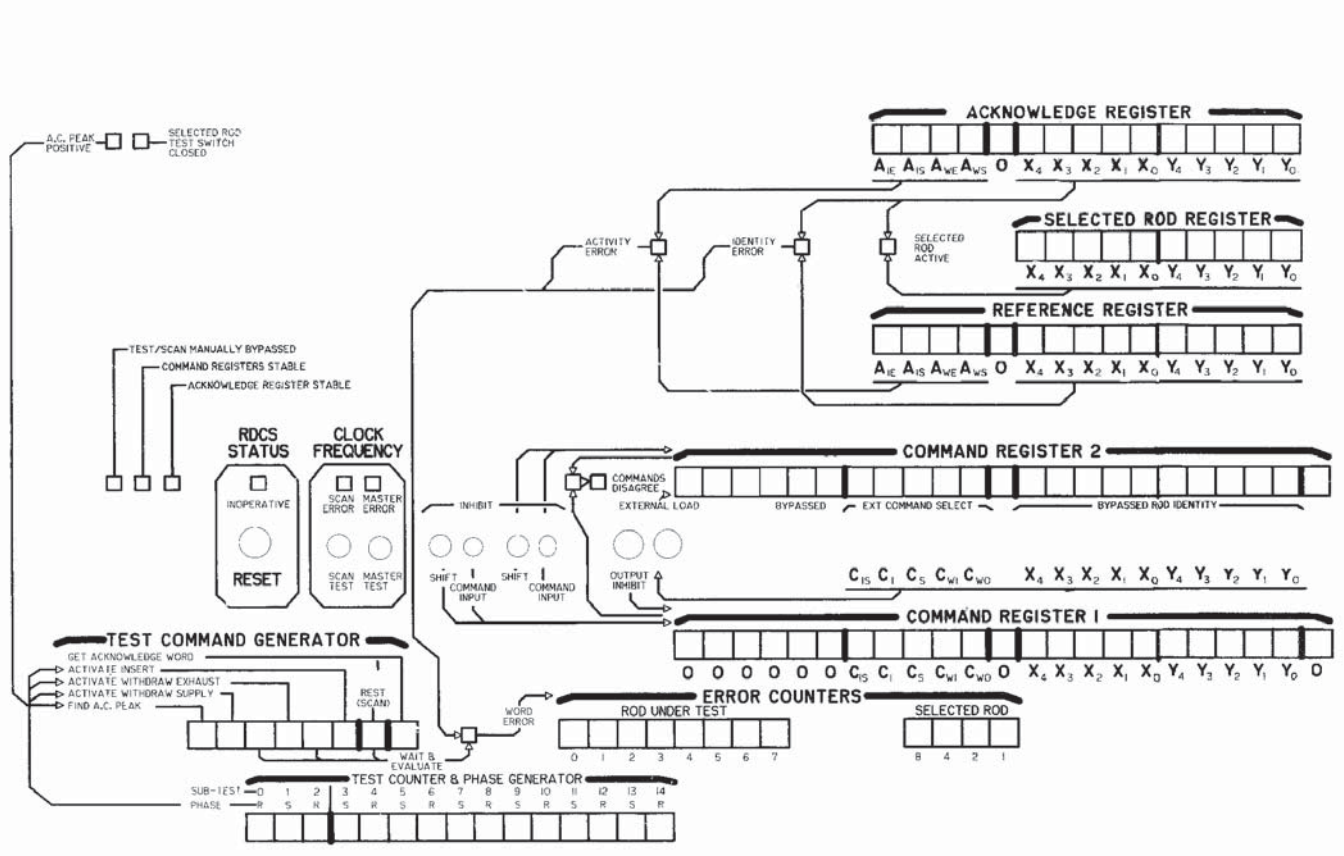
B. SYSTEM DESCRIPTION (Continued)

B.7.2 (Continued)

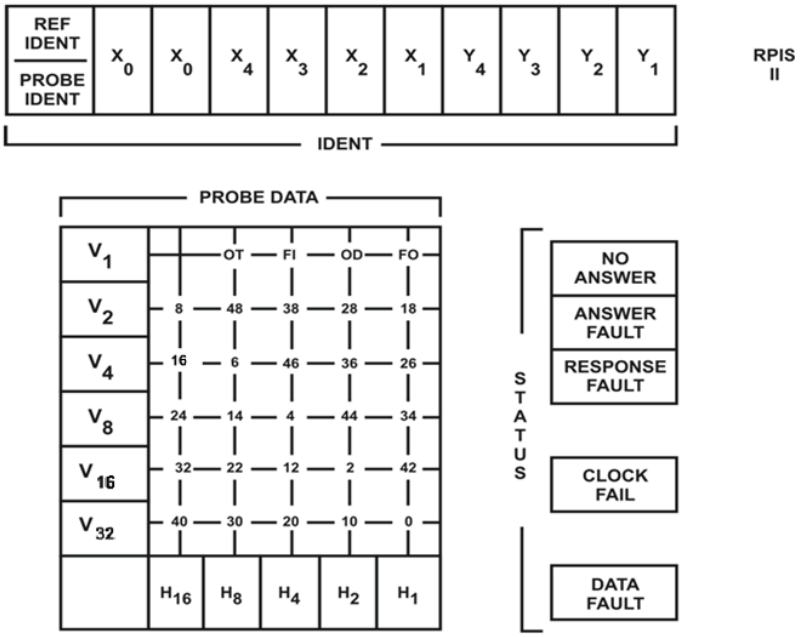


B. SYSTEM DESCRIPTION (Continued)

B.7.2 (Continued)



B.7.3. RPIS Display Card Indications



B. SYSTEM DESCRIPTION (Continued)

C. OPERATING REQUIREMENTS

The following systems should be in operation:

- N2-OP-30, Control Rod Drive
- N2-OP-71C, 600V A.C. Power Distribution
- N2-OP-92, Neutron Monitoring
- N2-OP-95A, Rod Worth Minimizer System
- N2-OP-97, Reactor Protection System

D. PRECAUTIONS AND LIMITATIONS

D.1. Use caution when using Continuous Rod Withdrawal in order to avoid large reactivity transients. Continuous Rod Withdrawal should **not** be used when approaching criticality.

D.2. Monitor the appropriate nuclear instrumentation response closely during **any** rod movement.

D.3. During reactor startup and shutdown, the selected control rod sequence must be strictly adhered to, in order to prevent local fuel damage. If the sequence **cannot** be adhered to, the Reactor Engineer shall be contacted for instructions. Refer to Technical Specification 3.1.6, Rod Pattern Control.

D.4. Whenever a rod is withdrawn to the full-out position, a coupling check must be performed.

D.5. Use CAUTION when using the Continuous Insert push button for rod movement, because the rod motion timer is bypassed and there is **no** settle function. Fully inserting a rod in this manner could cause damage to the CRD seals.

D.6. Under certain conditions, operation may continue with the Rod Worth Minimizer inoperable. However, additional restrictions must be applied. Refer to section H. of N2-OP-95A (RWM).

D.7. Unless specifically directed by the Emergency Operating Procedures, using the individual scram test switches to insert control rods to rapidly reduce power is prohibited. This method of power reduction can result in high rod worths, which could result in fuel damage.

D.8. Do **not** replace a light bulb on the rod select module for a selected rod. With the lens cover removed on the select module and that rod selected, damage could occur to the reactor manual control system should an object fall onto the exposed contacts. A different rod should be selected prior to removing the lens cover for a rod on the select matrix.

D.9. When a rod is bypassed in the RDCS, this rod becomes transparent to the control system. The rod will **not** function in the operator control scan, or test modes. Associated HCU status checks and other electronic checks performed by the system will be unavailable for the bypassed rod (**no** accumulator alarms). Refer to Technical Specification 3.1.5 (Modes 1 and 2) or 3.9.5 (Mode 5) for actions. In lieu of declaring the accumulator inoperable, local monitoring of the accumulator indications is permitted.

D.10. If in Mode 5, the Mode switch must be locked in REFUEL prior to withdrawing a control rod and One-Rod-Out Interlock shall be operable. (N2-OSP-RMC-W@002) (Technical Specification 3.9.2).

D.11. If in Mode 5, whenever a control rod is withdrawn from position 00, the FULL IN light **not** lit must be verified with a valid position observed on the four rod display (Technical Specification 3.9.4).

F. NORMAL OPERATIONS

F.1. Control Rod Selection

F.1.1. At 2CEC*PNL603 **SELECT** the rod to be moved by **MOMENTARILY DEPRESSING** the pushbutton for that rod on the Rod Select Module. _____

CAUTION

Do **not** replace a light bulb on the rod select module for a selected rod. With the lens cover removed on the select module and that rod selected, damage could occur to the reactor manual control system should an object fall onto the exposed contacts. A different rod should be selected prior to removing the lens cover for a rod on the select matrix.

F.1.2. **OBSERVE** the following:

- The pushbutton for selected rod is lit..... ☐
- The white select light on full core display is lit for control rod. ..... ☐
- The rod's position is indicated **AND** in correct location on 4-Rod Display. ..... ☐
- **IF** available, **THEN** Rod Worth Minimizer indicates correct rod **AND** rod position **OTHERWISE MARK** this step N/A..... ☐

_____

F.2. **Control Rod Withdrawal**

F.2.1 **PERFORM** Notch Withdrawal as follows:

a. **VERIFY** Reactor mode switch is **not** in SHUTDOWN
AND there are **no** rod withdrawal blocks. _____

b. **IF** in Mode 5,
THEN VERIFY Mode Switch is locked in REFUEL
AND One-Rod-Out Interlock is operable,
OTHERWISE MARK this step N/A. _____

c. **VERIFY** correct rod has been selected per sequence being used. _____

d. **WITHDRAW** selected control rod as follows:

1) **DEPRESS** WITHDRAW push button. ..... ☐

2) **CONFIRM** Rod motion as indicated by:

- Rod position indication on 4-Rod Display ..... ☐
- **OR** Withdraw light is illuminated. ..... ☐

3) **RELEASE** WITHDRAW pushbutton. ..... ☐

4) **OBSERVE** SETTLE light illuminated. ..... ☐ _____

NOTE

Control rods are more likely to double notch at elevated drive water differential pressures.

e. **IF** control rod failed to withdraw,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.2.1.d,
OTHERWISE MARK this step N/A. _____

f. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.1 (Continued)

g. **IF** control rod fails to withdraw at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

h. **OBSERVE** selected rod moves out one notch to next even-numbered position on 4-Rod Display. _____

i. **IF** in Mode 5
AND withdrawn rod was at position 00,
THEN VERIFY FULL IN light is **not** lit
AND a valid position is observed on four rod display,
OTHERWISE MARK this step N/A. _____

j. **MONITOR** appropriate nuclear instruments for proper response. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.2. **PERFORM** Continuous Withdrawal as follows: **[C3]**

a. **VERIFY** Reactor mode switch is **not** in SHUTDOWN
AND there are **no** rod withdrawal blocks. _____

b. **IF** in Mode 5,
THEN VERIFY Mode Switch is locked in REFUEL
AND One-Rod-Out Interlock is operable,
OTHERWISE MARK this step N/A. _____

c. **VERIFY** correct rod has been selected per sequence being used
AND that continuous rod withdrawal is allowed for given conditions. _____

NOTE

If control rood being withdrawn has a final target position of 48, then coupling check per 0 may be performed concurrently with step F.2.2.d.

d. **SIMULTANEOUSLY DEPRESS** WITHDRAW
AND CONTINUOUS WITHDRAW push buttons
AND OBSERVE WITHDRAW
AND CONTINUOUS WITHDRAW lights are on. _____

NOTE

- Control rods are more likely to double notch at elevated drive water differential pressures.
- Drive Speed will be faster when at elevated drive water pressure, therefore single notch movement should be used.

e. **IF** control rod failed to withdraw,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.2.1.d to perform single Notch Withdrawal of the stuck control rod,
OTHERWISE MARK this step N/A. _____

f. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.2 (Continued)

g. **IF** control rod fails to withdraw at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

h. **OBSERVE** selected rod moves continuously outward on 4-Rod Display. _____

i. **IF** in Mode 5
AND withdrawn rod was at position 00,
THEN VERIFY FULL IN light is **not** lit
AND a valid position is observed on four rod display,
OTHERWISE MARK this step N/A. _____

j. **MONITOR** appropriate nuclear instruments closely for proper response. _____

NOTE

- For control rods with a target position other than position 48, both pushbuttons should be released to allow the control rod to settle 2 notches prior to the target position, to avoid overshooting the target.
- Exceptions to this guidance can be taken as authorized by the reactor engineer when it is undesirable to have a control rod settle in an intermediate position.

k. **RELEASE BOTH** push buttons
AND ALLOW rod to settle into desired notch
AND OBSERVE proper cycling of INSERT, WITHDRAW,
AND SETTLE lights. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.2.3. **WHEN** a control rod is withdrawn to position 48,
THEN PERFORM a Control Rod Coupling Check as follows:

a. **ATTEMPT** to withdraw the Control Rod past position 48 by **DEPRESSING AND HOLDING BOTH** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons
AND OBSERVE the following:

- Position 48 shows on Four Rod Display. ..... ☐
- Annunciator 603444, CONTROL ROD OVERTRAVEL, remains clear..... ☐
- Control rod FULL OUT indicating light illuminated on Full Core Display..... ☐
- Any indicated response on nuclear instrumentation while withdrawing control rod..... ☐ _____

b. **RELEASE** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons. _____

F.3. **Control Rod Insertion**

F.3.1. **PERFORM** Notch Insertion as follows:

a. **VERIFY** there are **no** rod insert blocks. _____

b. **VERIFY** correct rod has been selected per sequence being used. _____

c. **INSERT** selected control rod as follows:

1) **DEPRESS** INSERT push button. ☐

2) **CONFIRM** Rod motion as indicated by:

- Rod position indication on 4-Rod Display ☐
- **OR** INSERT light is illuminated ☐

3) **RELEASE** INSERT pushbutton. ☐

4) **OBSERVE** SETTLE light illuminated. ☐ _____

d. **OBSERVE** selected rod moves in one notch to next even-numbered position on 4-Rod Display. _____

e. **MONITOR** appropriate nuclear instruments for proper response. _____

f. **IF** control rod failed to insert,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.3.1.c,
OTHERWISE MARK this step N/A. _____

g. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

h. **IF** control rod fails to insert at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.2. **PERFORM** Continuous Insertion as follows: **[C3]**

CAUTION

Do **not** use the Continuous Insert push button for normal rod movement. The rod motion timer is bypassed and there is **no** settle function, fully inserting a rod in this manner could cause damage to the CRD seals.

a. **VERIFY** there are **no** rod insert blocks. _____

b. **VERIFY** correct rod has been selected per sequence being used. _____

c. **DEPRESS AND HOLD** INSERT push button
AND OBSERVE INSERT light remains on as long as push button is held down. _____

d. **OBSERVE** selected rod moves continuously inward on 4-Rod Display. _____

NOTE

- For control rods with a target position other than position 00, the pushbutton should be released to allow the control rod to settle 2 notches prior to the target position, to avoid overshooting the target.
- Exceptions to this guidance can be taken as authorized by the reactor engineer when it is undesirable to have a control rod settle in an intermediate position or when performing control rod exercising.
- For control rods with a target position of 00, the pushbutton should be released to allow the control rod to settle into position 00.

e. **RELEASE** push button
AND ALLOW rod to settle into desired notch
AND OBSERVE proper cycling of INSERT
AND SETTLE lights. _____

f. **MONITOR** appropriate nuclear instruments for proper response. _____

F. NORMAL OPERATIONS (Continued) **Initials**

F.3.2 (Continued)

NOTE

- Control rods are more likely to double notch at elevated drive water differential pressures.
- Drive Speed will be faster when at elevated drive water pressure, therefore single notch movement should be used.

g. **IF** control rod failed to insert,
THEN RAISE drive water pressure in less than 50 psid increments, up to approximately 350 psid by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV
AND RETURN to step F.3.1.3 to perform single Notch Withdrawal of the stuck control rod,
OTHERWISE MARK this step N/A. _____

h. **WHEN** control rod motion is achieved
OR it is determined motion will **not** be achieved with 350 psid Drive Water pressure,
THEN RESTORE Drive Water pressure to normal band (approximately 260 psid) by
THROTTLING 2RDS-PV101, DRIVE WTR PRESS CONTROL MOV,
OTHERWISE MARK this step N/A. _____

i. **IF** control rod fails to insert at elevated drive pressure,
THEN PERFORM N2-OP-30 H.1.0, Failure to Insert/Withdraw Control Rod,
OTHERWISE MARK this step N/A. _____



Nine Mile Point Nuclear Station UNIT 2
OPERATING PROCEDURE

N2-OP-101A

PLANT START-UP

Revision 05400

Safety Related

CONTINUOUS USE

10 CFR 50.59 Review Required

Approval Authority: Director - Operations

SUMMARY OF ALTERATIONS

PCR-22-00445:

- **Description:**
  - Update the point at which 2FWS-LV55A/B will be closed during power ascension, and power range when low power mode to high power mode transition occurs.
  - Correct procedure section referenced for defeating lock low power mode
- **Reason:** Upgraded 2FWS-LV55A/B valves will be installed during RFO N2R18 per ECP-20-000442.
- E.5.6, removed previous revision step E.5.6.c to **VERIFY** 2FWS LV10 is controlling in automatic with 2FWS LV55A closed **AND** in Auto.
- E.5.6.f revised verification that LOW POWER MODE to HIGH POWER MODE transition occurs from 15% - 20% reactor power to 20% - 25% reactor power.
- E.5.6.g revised from "**IF REQUIRED AND**" to just "**IF**", as this is always required.
- E.5.6.g corrected step referenced in N2-OP-3.
- E.5.15 preceding note that with 2FWS-LV55A(B) and 2FWS-LV10A(B) controllers in Auto, 2FWS-LV55A(B) will automatically close when Total Steam Flow exceeds 3.7 Mlbm/Hr (about 27% RTP).
- E.5.15 added new step to **BEFORE** Reactor Recirculation Pump Upshift, **VERIFY** 2FWS-LV55A(B) is CLOSED **AND** 2FWS-LV10A(B) is controlling level per N2-OP-3, Section E.3.

PCR-22-00446:

- **Description:** Editorially update Attachment 7, Startup Guide Flowchart.
- **Reason:** Match startup sequence.
- Page 3 of 5, 4 of 5, and 5 of 5 updated per PCR markup.
- **No** change to technical content of steps in procedure body directing actions.

PCR-22-00698:

- **Description:** Editorially add independent verification steps for any tech spec decision.
- **Reason:** IR 04451632, Declining trend in Fleet Operations Performance
- A.7 added **C26**, IR 04451632, Declining Trend in Fleet Operations Performance
- Throughout, changed Equipment Status Log (ESL) to LCO Tracker, **no** revision bars used.
- E.2.5, E.2.45.b.2), E.3.24.b, and E.5.13, added **[C26]** and IV placekeeping.
- E.8.6 added step for US **AND** SM to **REVIEW** section E.8, Establishing Primary Containment for Mode 2, for completeness.
- **No** change to technical content of steps in procedure body directing actions.

TABLE OF CONTENTS

SECTION PAGE

A. REFERENCES AND COMMITMENTS 4
B. SYSTEM DESCRIPTION 7
C. OPERATING REQUIREMENTS 7
D. PRECAUTIONS AND LIMITATIONS 8
D.1. General 8
D.2. Reactor Startup and Heatup 10
E. STARTUP 12
E.1. Prerequisites 12
E.2. Approach to Criticality, Heatup, and Vessel Pressurization 21
E.3. Transfer of Mode Switch to RUN 42
E.4. Turbine Startup 49
E.5. Raise Power to 45% Rated 51
E.6. Reactor Start-up Short Form Checklist 57
E.7. Master Startup Checklist 68
E.8. Establishing Primary Containment for Mode 2 89
E.9. Pressure Adjustments Using Automatic Heatup Rate 97
E.10. Pressure Adjustment Using Manual Rate 98
F. NORMAL OPERATIONS 99
G. SHUTDOWN 99
H. OFF-NORMAL OPERATIONS 100
H.1. End-Of-Cycle Recirculation Pump Trip or Main Steam Bypass INOP 100
Attachment 1, Reactor Startup Preparation and Briefing Checklist 101
Attachment 2, Primary Containment AC Circuits 110
Attachment 3, SRM Count Rate Posting 112
Attachment 4, Appendix R Valve/Switch Lineup 113
Attachment 5, N2-OP-101A Procedure Variance Form 117
Attachment 6, Initials to Name Log 118
Attachment 7, Startup Guide Flowchart 119
Attachment 8, IRM Range at the Point of Adding Heat 124
Attachment 9, Startup Procedure Guidance 125

A. REFERENCES AND COMMITMENTS

A.1. Technical Specification

SR 3.3.1.1.5-1.a, IRM/SRM Overlap

A.2. Licensee Documentation

- Technical Specification Amendment 123 (ARTS / MELLLA)
- Amendment 140 to the Facility Operating License, Extended Power Uprate
- Technical Specification Amendment 145, Pressure/Temperature Limit Curves (TAC No. MF0345)

A.3. Standards, Regulations, and Codes

Consult individual operating procedures as necessary.

A.4. Policies, Programs, and Procedures

- OP-AA-108-115, Operability Determinations (CM-1)
- CY-AB-120-100, Reactor Water Chemistry
- CY-AB-120-110, Condensate and Feedwater Chemistry
- CY-AB-120-300, Spent Fuel Pool
- CY-AB-120-320, Control Rod Drive Water Chemistry
- WC-AA-104, Integrated Risk Management
- OP-AA-108-114, Post Transient Review
- Consult individual operating procedures as necessary

A.5. Technical Information

- Consult individual operating procedures as necessary.

A.6. Supplemental References

- NMP-AMP-ELECT02, Non-EQ Cables and Connections used in Instrumentation Circuits Program.
- PTLR-2, Nine Mile Point Unit 2, Pressure and Temperature Limits Report (PTLR) Revision 0 (ECP-12-000055)
- AR 02410659, Reactivity Management Performance Gaps During Low Power Operations (ACE) and NER NC-14-022
- ECP-12-000448, ESR-12-000241 NMP-ESR (0000) – NMP2-IMPLEMENTATION OF MELLA PLUS
- IR 04432785, BWR-NEB-21-9 SRNI Condition Prohibited by Tech Specs.

A. REFERENCES AND COMMITMENTS (Continued)

A.7. **Commitments**

Sequence Number	Commitment Number	Description
1	NCTS 502568-00	2CNM-LV137 Capacity
2	NCTS 700138-00	APRM Calibration During Plant Startup
3	NCTS 003772-01	Actions for low load and high back pressure (low vacuum) conditions
4	SOER 90-2	Nuclear Fuel Defects
5	SOER 90-3	Nuclear Instrumentation Miscalibration
6	IR 02062400 (Legacy DER 2-92-Q-0050)	Control Rod Drive problems experienced during 12/13 and 12/16/91 reactor startup
7	SOER 88-02	Premature Criticality Events During Reactor Startup
8	NRC SSER 3 & 5	Item 2, pgs. 6-17, 18 and pg. 6-6 respectively. De-energizing 2MSS*SOV97s
9	NCTS 503896-00	Requirements for evaluation of leakage within containment (DER 2-94-2456)
10	IR 02072703 (Legacy DER 2-95-0473)	Incorporate foreign material exclusion concerns as identified by NRC Information Notice 95-06, "Potential Blockage of Safety-Related Strainer by Material Brought Inside Containment"
11	NMPIL 1178	NMPC Response to GL 96-06
12	IR 02080793 (Legacy DER 2-97-1698)	Deleted
13	IR 02079427 (Legacy DER 2-97-0312)	For Operational Condition 2 and 3, Operators must manually ensure that the Vacuum-Low Isolation Actuation Logic is Fully Operational by placing all four bypass switches to the NORMAL position prior to opening any or all TSVs
14	IR 02080716 (Legacy DER 2-97-1621)	Fluid Entrapment in Valve Bonnets
15	IR 02079099 (Legacy DER 2-96-3449)	Turbine Trip-POT Drawer open
16	PRNM MOD SER	Verify APRM Setpoints after Mode SW Change
17	IR 02086101 (Legacy DER 2-98-3593)	Hot Reheat Steam Temperatures exceeded 50 °F
18	IR 02088958 (Legacy DER C-1999-2451)	Potential Positive Moderator Temp. Coefficient

A. REFERENCES AND COMMITMENTS (Continued)

Sequence Number	Commitment Number	Description
19	IR 02089952 (Legacy DER 1-1999-3456)	Unexpected Criticality at NMP1
20	IR 02094651 (Legacy DER 2-2000-3952)	Tech. Spec. LCO Action Statement Entered Because Primary Containment **not** Inerted Within 24 Hours After Reaching 15%. [TS Amendment 184 changed TS 3.6.3.2 LCO A to 72 hours.]
21	IR 02094523 (Legacy DER 2-2000-3818)	Potential Recirculation System Jet Pump Cavitation
22	NCTS 504583-29	NMPIL 1962, July 14, 2005 License Renewal Application. Annotated sections of this document implement an NRC commitment to manage the effects of aging for cables and connections within the scope of License renewal during the period of extended operation
23	IR 02031007 (Legacy CR-2007-2646)	Control Rod 22-03 Triple Notched In During Weekly Rod Exercising
24	USAR, Appendix C	Section C.1.7, BWR Feedwater Nozzle Program. Action Tracking 04257377-06, License Renewal BWR Feedwater Nozzle Program.
25	IR 04285674	OPEX from Browns Ferry IRM Scram when withdrawing SRM's
26	IR 04451632	Declining Trend in Fleet Operations Performance

B. SYSTEM DESCRIPTION

This procedure accomplishes a Reactor and Balance of Plant startup. This procedure is written in a logical, sequential order to bring the plant from a Cold or Hot Shutdown condition to power operations and is **not** intended to cover every possible eventuality that could occur during this evolution. Situations that prevent a procedure step from being signed off in the procedure will be documented and justified as stipulated in the next section.

C. OPERATING REQUIREMENTS

C.1. When situations/conditions arise that prevent or require a procedure step to **not** be completed, continuation past that step is permitted provided the reason the step **cannot** be completed and the justification for why it is acceptable to proceed is documented on a procedure variance form (Attachment 5, N2-OP-101A Procedure Variance Form).

The variance form shall include a brief explanation as to why the step **cannot** or need **not** be signed off and shall be signed by both the Control Room Supervisor and Shift Manager. A "V" shall then be placed to the left of the associated step(s) signifying a variance has been completed.

If a step is logged on the procedure variance form and at some time later, while still performing this procedure, the step can be completed, then upon completion of the step, the Control Room Supervisor will draw a single line through the procedure step number on the variance form and will print ‘Completed’ above the line followed by his initials and date. In addition, the Control Room Supervisor will initial, date and time the step within the procedure upon completion of the step.

Each time a new section is entered, the previous sections shall be reviewed for completeness. Any step that was **not** previously signed off shall be re-evaluated for applicability. If any steps in a section have **not** been completed, the next section shall **not** be entered unless the current variances are reviewed by the Control Room Supervisor and the Shift Manager.

Several pre-startup preparations are expected to be completed or previously satisfied upon entry into this procedure. Signoff of these steps is permitted after it has been verified that all of the actions required by the step have been completed.

C.2. If required to discontinue the startup, then exit this procedure and refer to N2-OP-101C, Plant Shutdown.

C.3. The Shift Operations Superintendent shall ensure that as a minimum, two additional Reactor Operators and one additional Senior Reactor Operator above the normal operating shift manning are available during reactor startup. **[C19]**

C.4. The Reactor Engineer on duty shall be present in the Control Room during reactor startup commencing with control rod withdrawal, and up to the point of adding heat. **[C19]**

C.5. Reactor startup activities shall be administratively controlled per the requirements stipulated in WC-AA-104, Integrated Risk Management, for placing the proper level of focus and attention for these evolutions.

C.6. Each person who places their initials in this procedure should provide their printed name and initials on Attachment 6, Initials to Name Log.

D. PRECAUTIONS AND LIMITATIONS

D.1. General

D.1.1. Verify Thermal Limits once within 12 hours after ≥23% RTP and 24 hours thereafter.

D.1.2. Recirculation pump upshift or start must be performed below 58% rodline.

D.1.3. Main Turbine operation with load less than 30% (410 MWE) with condenser vacuum less than 24.6" Hg is prohibited. **[C3]**

D.1.4. Condenser vacuum should be established prior to placing fresh condensate demineralizers in service.

D.1.5. The Mechanical Vacuum pump should **not** be operated if there is any indication of fuel failure (except as directed by EOPs, or if permitted by Engineering and Chemistry evaluation).

D.1.6. For Operational Condition 2 and 3, Operators must manually ensure that the Vacuum-Low Isolation Actuation Logic is fully operational by placing all four bypass switches to the NORMAL position prior to opening any one or all TSVs. **[C13]**

D.1.7. When returning potential transformer drawers to their normal configuration, care must be taken to ensure that the drawer is fully engaged such that all circuit connections are properly connected. **[C15]**

D.1.8. When Reactor Thermal Power is greater than 70%, with charcoal adsorbers in service, and Offgas System flow is greater than 25 SCFM, the Offgas System Sample Delay Pipe should be in service in order to maintain 2OFG-RE13A/B sample consistent with the effluent path and maintain proper margin to the trip setpoints.

D.1.9. Caution needs to be observed when at reduced reactor power and pressures. Leakage past the LV10s and/or LV55s can cause level and associated power excursions.

D.1.10. Maintain power and flow in the operating domain of the applicable single and two loop power-flow operating maps. Determine expected operating location prior to making reactivity changes. **[C21]**

D.1.11. Primary Containment Oxygen shall be controlled per the requirements of Technical Specification 3.6.3.2.

D.1.12. The Feedwater minimum flow valves, 2FWR-2FVA(B,C), are susceptible to seat erosion at certain reactor power levels. Operation of the following Feed Pump combinations at these reactor power levels should be minimized:

- Single Pump Operation - 23% to 27% Power
- Two Pump Operation with balanced flows - 46% to 53% Power

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.1.13. To facilitate shortened refueling outages, the maximum allowable Spent Fuel Pool temperature from the time the Spent Fuel Pool gates are installed for cavity drain down can be determined from the graph seen as Attachment 8 to N2-OP-38. With the temperature of the Fuel Pool water below the maximum allowed temperature (115° Fahrenheit 8 Days After Reactor Shutdown), the station will be assured of being within the 4-hour coping time for a station blackout were it to occur concurrently with commencement of drain down. (Refer to Figure 9.1 of NMP Calculation A10.1-J-038, Rev 3.)

D.1.14. A CRD Deficiency Log entry should be made at any time during the performance of this procedure when a control rod drive mechanism does **not** perform as expected.

D.1.15. Continuation of the evolution must be evaluated whenever any of the following criteria is met (Hold Criteria):

- Equipment or systems do **not** operate as expected.
- The status of test and Operations personnel readiness for Startup is **not** as expected.
- Conditions are **not** as expected.
- The margin to safety is significantly reduced.
- Any unexpected reactivity transient.
- Any unexpected Startup related event that requires entry into an unplanned LCO.
- Any unexpected Startup related event that is reportable or potentially reportable to the NRC.
- Unplanned/Unexpected entry into any EOP/SOP.

D.1.16. Reactor Building unit heaters, except as noted below, and 2HVR-CH2 and CH3 in the Secondary Containment Boundary shall be de-energized while the plant is in Operational Conditions 1, 2 and 3 to comply with the Secondary Containment Drawdown Analysis.

Installed Unit Heaters in the following rooms have been approved for use during Operation Condition 1, 2 and 3.

- RHR A, B and C Pump Rooms
- RHR A and B Hx Rooms
- LPCS Pump Room

D.1.17. In Modes 4 and 5, 2HVR*TIS31A/B may be adjusted to actuate at 85 °F vs. the Mode 1, 2 and 3 setpoint of 70 °F. IF adjusted in this manner, 2SWP*AOV97A/B will **not** open until inlet temperature to 2HVR*UC413A/B reaches 85 °F. This should eliminate the need to isolate Service Water to a running UC413 in Mode 4 or 5.

D.1.18. Excessive vacuum in the condenser can increase deflection of mid-span steam path components, this combined with changing temperatures, can bring the rotating element in contact with stationery components. Elevated condenser vacuum is more likely at low power levels and thus makes these conditions more likely. Rub Mitigation per N2-OP-9 is recommended per the BWROG Scram Reduction documentation, 1NER-1E-034.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.1.19. During Reactor startup the Reactor Water Cleanup System (WCS) Flow Rate should be maximized by the use of two WCS Pumps and four WCS Filter-Demins. Maximizing WCS filtration minimizes crud impact on the Control Rod Drive Mechanism (CRDM) collet piston, stop piston, and drive piston seals.

D.2. Reactor Startup and Heatup

D.2.1. Precautions and Limitations associated with movement of control rods during startup are contained in Attachment 1, Reactor Startup Preparation and Briefing Checklist.

D.2.2. During plant heatup with a negative moderator temperature coefficient, it is possible that periods of low control rod worth withdrawal or short delays in rod movement may result in temporary subcriticality. IF the reactor goes subcritical, insert control rods until only Group 1 control rods are withdrawn, review the flowchart in Attachment 9, Startup Procedure Guidance (AR 02410659 and NER NC-14-022).

D.2.3. Per the Pressure and Temperature Limit Report (PTLR) Vessel Heatup and Cooldown rates shall **not** exceed 100 °F/hr (TS 3.4.11).

D.2.4. The definition for subcriticality is defined as:

1. **BEFORE** reaching the Point of Adding Heat – Multiple SRM/IRM readings continuously lowering without rod insertions.

   OR

2. **AFTER** reaching the Point of Adding Heat – Multiple SRM/IRM readings continuously lowering without rod insertions.

   AND

   Multiple IRMs are at least 2 ranges below the levels established when initially at the Point Of Adding Heat (POAH).

D.2.5. The definition of the subcritical hold point is 3 Count Rate Doublings. This hold point is where the core is sufficiently subcritical to provide a reasonable assurance that the reactor will remain subcritical with the reactor plant parameters within the evaluated startup conditions.

D.2.6. Although the reactor is expected to remain subcritical at the subcritical hold point, changing plant conditions can reduce the margin to criticality. When the reactor is maintained at the subcritical hold point, reactor parameters shall be routinely monitored and significant changes assessed for impact on the margin to criticality.

D.2.7. When beyond Middle-of-Cycle exposure, moderator temperature coefficient may become positive. This could shorten reactor period during heatup, after achieving initial criticality. Increasing moderator temperature during approach to criticality can pull up the expected ECP when operating with a +MTC.

D.2.8. Criticality should be expected at any time. Close attention to all indications of neutron flux and period is necessary. The point of criticality may be impacted by control rod withdrawal, changing reactor coolant temperature or changing Xenon concentrations.

D. PRECAUTIONS AND LIMITATIONS (Continued)

D.2.9. A +MTC can be observed as a period that gradually becomes shorter after the initial criticality, without any further rod withdrawals. If a +MTC is observed, the period should be monitored such that IRM range changes remain controllable. Additionally, heatup rate should be monitored to remain less than 20F every 15 minutes. If period or heatup rate become excessive, single notch insertion may be used, backwards in the sequence, to maintain the reactor critical on a controllable period. **[C18]**

D.2.10. Extended periods of operation at low power (such as during operation in hot standby), should be discouraged. Under these conditions a small change in reactivity (poison build-up, pressure change) can have a large effect on reactor parameters including driving the core subcritical and varying bypass valve position. The Control Room staff should monitor for any unexpected indications and take action as directed by Control Room Supervisor.

D.2.11. The following parameters are required to be monitored during each reactivity manipulation:

- Redundant Nuclear Instrumentation
- Core Flow
- Reactor Pressure
- Reactor Temperature

D.2.12. 2MSS*MOV111, 112, and 208 are isolation valves in the Main Steam Line Drains between or upstream of the Main Steam Isolation Valves (MSIVs). Because the closure times of these valves are longer than the MSIV closure times, the duration that 2MSS*MOV111, 112, or 208 are open shall be minimized to remain within the bounds of the current isolation time analysis. The total allowable time that valves may be open during a given year is less than 135 hours, when Primary Containment Isolation function is required (reference Calculation Dispositions No. ES-177-05E and ES-109-01B).

D.2.13. When the Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig, the valves are subjected to a differential pressure greater than the engineering defined limit. Therefore, the valves must be declared INOPERABLE and Technical Specification 3.6.1.3, Condition B entered. Once entered, the associated action is to isolate the penetration with at least one valve within one hour (reference Calculation Disposition No. A10.2-O-026).

D.2.14. To minimize thrusting the Reactor Coolant Pump Motor Thrust Bearings:

- While raising Reactor Pressure, minimize time in the thrust transition zone; between 315 and 500 psig.
- When lowering Reactor Pressure, minimize time in the thrust transition zone; between 315 and 260 psig.
- Reactor Pressure should pass efficiently through the thrust transition zone so the RCS pump seal is **not** subject to repeated large, axial displacements.
- If the RCS pressure remains in the transition range for extended periods, the seal faces can be damaged.
- It is a good practice to be aware of the thrust transition zone, and to pass through it as efficiently as possible.

E. STARTUP

NOTE

- Because of many possible plant conditions, deviation from suggested sequence is permitted as directed by the Shift Manager.
- Steps **not** required to be performed shall be documented per Operating Requirement C.1.

E.1. Prerequisites

E.1.1. Shift Operations Superintendent, **OR** designee, **PERFORM** the following:

a. **VERIFY** the Plant General Manager
OR Manager Operations has designated a Senior Manager to oversee the evolution
AND RECORD
Designated Senior Manager Name: Designated Senior Manager — SOS / Today
S.O.S.

b. **ASSIGN AND RECORD** a Dedicated SRO for Reactivity Management evolution:

Dedicated SRO Name: Dedicated SRO — SOS / Today
S.O.S.

E.1.2. Using HU-AA-1211, Pre-Job Briefings, **PERFORM** a Pre-Job Brief including Attachment 9, Startup Procedure Guidance. — MCA / Today

a. **DISCUSS** the roles and responsibilities of the Rod Mover, Peer Checker, Reactivity Manager (RM) SRO, and Qualified Nuclear Engineer (QNE). — MCA / Today

b. **DISCUSS** the following:

- **DISCUSS** ±1.0% bands based on current Temperature and Xenon Conditions ..... ☑
- **DISCUSS** high worth steps and their proximity to the ECP..... ☑
- **DISCUSS** ECP including adjustments due to count rate doubling information obtained during approach to criticality.... ☑ MCA / Today

E.1.3. **INDICATE** permission to perform this Activity:

Director Shift Operations
Director Site Operations

Shift Operations Superintendent
Shift Operations Superintendent

E. STARTUP (Continued) **Initials/Date**

E.1.4. **VERIFY** one of the following startup checklists is completed and attached:

☑ E.6 Reactor Start-up Short Form Checklist

☑ E.7 Master Startup Checklist MCA / Today

E.1.5. **VERIFY** the following for ALL IRMs:

	Fully Inserted	Range Sw. at Position 1	Recorder set to IRM
IRM A	☑	☑	☑
IRM B	☑	☑	☑
IRM C	☑	☑	☑
IRM D	☑	☑	☑
IRM E	☑	☑	☑
IRM F	☑	☑	☑
IRM G	☑	☑	☑
IRM H	☑	☑	☑

MCA / Today

E.1.6. **VERIFY** the following for ALL SRMs:

	Fully Inserted	Indicates ≥ 3 cps	Attachment 3 complete, Posted on P603 adjacent to SRM meters	Recorder selected to highest 2 channels
SRM A	☑	☑	☑	☑
SRM B	☑	☑	☑	☑
SRM C	☑	☑	☑	☑
SRM D	☑	☑	☑	☑

MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.1.7. **IF** any of the following valves have
OR may have been exposed to hydrostatic testing
OR flooding of associated piping,
THEN STROKE affected valve to ensure fluid is **not** trapped in valve bonnet,
OTHERWISE MARK these steps N/A: **[C14]**

- 2ASS-STV112, REBOILERS 1A & 1B MAINSTM TRIP VLV ..... ☑
- 2ASS-STV143, BLDG INTMD HT EXCH MN STM ISOL ..... ☑
- 2ESS-STV105, 5th POINT EXTRACTION TO AUX STEAM ISOL ..... ☑
- 2MSS-AOV92A, STM SUPPLY VLV ..... ☑
- 2MSS-AOV92B, STM SUPPLY VLV ..... ☑ SM / Today
SM

E.1.8. **VERIFY** Narrow Range Reactor Water Level Instruments indicate normal operating range (178.3" to 187.3"):

- C33-R606A ..... ☑
- C33-R606B ..... ☑
- C33-R606C ..... ☑
- C33-R608 ..... ☑ MCA / Today

E.1.9. **VERIFY** the following Reactor Head Vents **OPEN**:

- 2MSS*MOV118, Reactor vessel vent throttle ..... ☑
- 2MSS*MOV119, Reactor vessel vent throttle ..... ☑ MCA / Today

E.1.10. **IF** startup is from cold conditions,
AND RPV is depressurized,
THEN VERIFY outboard MSIVs closed,
OTHERWISE MARK this step N/A. MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

Post shutdown cooling flush should be deferred until in Operational Condition 2.

E.1.11. **VERIFY** the following:

a. WCS, RHR and RCIC HIGH AREA TEMPERATURE isolations have been restored per N2-OP-83, Attachment 4, Defeating/Restoring WCS or RHR and RCIC High Temperature Isolations ..... ☑

b. **VERIFY** Level 3 SDC isolation instruments robust barriers are removed from the following on RB 261':

- At Az 280°, 2CES*RAK026, RACK (RV LEVEL/PRESS INSTRUMENTATION/RACK C)..... ☑
- At Az 280°, 2CES*RAK027, RACK (RV LEVEL/PRESS INSTRUMENTATION/RACK B)..... ☑
- At Az 160°, 2CES*RAK005, RACK (RPS/WCS INSTRUMENTATION)..... ☑
- At Az 80°, 2CES*RAK004, RACK (RRS/CSL INSTRUMENTATION)..... ☑ MCA / Today

E.1.12. **VERIFY** HVR/GTS Isolations have been restored per N2-OP-52. MCA / Today

E.1.13. **VERIFY** all Mode dependent Tagouts have been removed to support Mode change. MCA / Today

E.1.14. **PRIOR TO** RPV temperature reaching 180 °F, **VERIFY** the following per N2-OP-3, Section E, Initial Condensate Booster Pump Startup and Lining Up CNM/FWS to the Reactor:

- At least one Condensate Pump in service..... ☑
- At least one Condensate Booster Pump in Service ..... ☑
- 2 to 4 demineralizers through 2CNM-LV137 **OR** 2FWS-LV55A(B) ..... ☑
- Long cycle flush is shut down..... ☑ MCA / Today

E.1.15. **VERIFY** reactor cavity shield plugs E, F, G, and H (lower shield plugs) are installed. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.1.16. **VERIFY** turbine is on turning gear per N2-OP-21, Main Turbine. MCA / Today

CAUTION

Prior to placing steam on turbine seals, establishing condenser vacuum, or admitting steam to turbine, Turbine Turning Gear should be in service to prevent rotor bowing due to uneven heating. See N2-OP-21 Precautions and Limitations.

E.1.17. **ESTABLISH** Main Turbine Seals by placing in service Gland Seal per N2-OP-25, Auxiliary Steam, Auxiliary Condensate and Gland Seal System. MCA / Today

NOTE

- Condenser Vacuum may be drawn when the Turbine is Cold (approximately ambient temperature) with or without Steam Seals as directed by the Shift Manager. Steam Seals shall be in service prior to commencing warming of the main turbine.
- Condenser Vacuum shall **not** be drawn when the Turbine is HOT, until steam seals are established, or an Engineering Evaluation has been completed.
- If the Main Turbine is reset, a turbine trip may occur as vacuum reaches the low vacuum trip setpoint.
- If Starting-Up from Main Steam Supply to Gland Seals, Step E 1.21 can be delayed until Gland Seal Regulated Supply Pressure Starts to Rise.

E.1.18. **DRAW** condenser vacuum using Condenser Air Removal pumps per N2-OP-9, Condenser Air Removal. MCA / Today

E.1.19. Manually **CONTROL** condenser vacuum per N2-OP-9, H.5.0, Main Condenser Vacuum Control for Potential Rub Mitigation. MCA / Today

NOTE

The following step increases the Low Energy Clean Up flow to mitigate corrosion product transport to the fuel and should be performed after a refuel or extended outage.

E.1.20. **IF** High Velocity Flush is required,
THEN ESTABLISH a High Velocity Flush per N2-OP-3, Condensate and Feedwater System, Step E.1.57,
OTHERWISE MARK this step N/A. MCA / Today

E.1.21. **VERIFY** Off Gas Preheater is warmed up and ready for service per N2-OP-42, Off Gas System. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.1.22. **VERIFY** the following per N2-OP-29, Reactor Recirculation System:

- 2RCS*P1A in low speed on Low Frequency Motor Generator set. ☑
- 2RCS*P1B in low speed on Low Frequency Motor Generator set. ☑
- 2RCS*HYV17A, A LOOP RECIRCULATION FLOW CONTROL VALVE, full open. ☑
- 2RCS*HYV17B, B LOOP RECIRCULATION FLOW CONTROL VALVE, full open. ☑
- 2RCS*P1A BRKR 3A, RECIRC PMP 1A MOTOR A, CLOSED. ☑
- 2RCS*P1A BRKR 4A, RECIRC PMP 1A MOTOR A, CLOSED. ☑
- 2RCS*P1B BRKR 3B, RECIRC PMP 1B MOTOR B, CLOSED. ☑
- 2RCS*P1B BRKR 4B, RECIRC PMP 1B MOTOR B, CLOSED. ☑ MCA / Today

E.1.23. **VERIFY** N2-OSP-LOG-D001, Attachment 8, Recirc Loop Flow Mismatch Surveillance is complete. MCA / Today

E.1.24. **VERIFY** the following steam drain valves are OPEN at Panel P824:

- Group I ☑
- Group II ☑
- Group III ☑ MCA / Today

CAUTION

In Operational Mode 2 or 3, the Condenser Low Vacuum Bypass switches must all be placed in Normal before any or all TSV's are open.

E.1.25. At Panel H13-P609,
VERIFY the following keylock switches are in BYPASS position:

- B22H-S24A, CNSR LOW VAC BYPASS MCA / Today
- B22H-S24C, CNSR LOW VAC BYPASS MCA / Today

E.1.26. At Panel H13-P611,
VERIFY the following keylock switches are in BYPASS position:

- B22H-S24B, CNSR LOW VAC BYPASS MCA / Today
- B22H-S24D, CNSR LOW VAC BYPASS MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

There are other factors that have contributed to WCS delta flow instability. Varying WCS reject flow may allow flow to stabilize.

E.1.27. **IF** WCS Delta Flow instruments are unstable,
THEN VERIFY IMD has vented or backfilled the following WCS flow transmitters to stabilize delta flow timers,
OTHERWISE MARK these steps N/A:

- 2WCS*FT67X/Y ..... ☑
- 2WCS*FT68X/Y ..... ☑
- 2WCS*FT69X/Y ..... ☑ MCA / Today

E.1.28. **IF STARTING UP** with one Recirc Loop out of service,
THEN VERIFY compliance with N2-OP-29, Section H, Single Recirculation Loop Operation,
OTHERWISE MARK this step N/A. MCA / Today

E.1.29. **PERFORM** the following for Drywell H2/O2 analyzers:

a. **IF** Primary Containment was de-inerted
OR surveillance requirement SR 3.6.3.2.1 is **not** current,
THEN VERIFY Drywell H2/O2 Analyzers in service per N2-OP-82, Containment Atmospheric Monitoring System,
OTHERWISE MARK this step N/A. MCA / Today

b. **IF** Primary Containment was **not** de-inerted
AND surveillance requirement SR 3.6.3.2.1 is current,
THEN VERIFY Drywell H2/O2 Analyzers in stand-by per N2-OP-82,
OTHERWISE MARK this step N/A. MCA / Today

c. **VERIFY** at least one of Drywell H2/O2 Analyzers has been set to 0 to 10% for oxygen concentration per N2-OP-82, Attachment 2,
AND ENTER other in LCO Tracker as required for TRM 3.3.3.1.1 requirements **IF** **not** set down. MCA / Today

E.1.30. **VERIFY** Redundant Reactivity Control System (RRCS) is in standby per N2-OP-36B, Redundant Reactivity Control System. MCA / Today

E.1.31. **VERIFY** shift staffing is per the following:

- TS 5.2.2 ..... ☑
- USAR Section 13.1 ..... ☑
- Operating Requirements C.3 and C.4 ..... ☑ MCA / Today

E. STARTUP (Continued) — **Initials/Date**

E.1.32. **OBTAIN** authorization for Reactor startup from Plant Manager **OR** designee. MCA / Today

E.1.33. **VERIFY** the following to ensure requirements of Secondary Containment Drawdown Analysis in Modes 1, 2, and 3 are met:

- Reactor Bldg Unit Heaters are shut off (Refer to D.1.16) ☑
- Setpoint for 2HVR*TIS31A/B is set at 70 °F (Refer to D.1.17) ☑ MCA / Today

E.1.34. **VERIFY ALL** Control Room chart recorders required for plant operation are turned on. MCA / Today

E.1.35. **VERIFY** Gaitronics in Merge **AND ANNOUNCE** the following:

"Attention all personnel, Reactor startup at Unit 2 is about to commence. Primary and Secondary Containments are in effect." MCA / Today

E.1.36. **DIRECT** the OCC to make startup notifications:

- System Power Control ☑
- Central Regional Control ☑
- Constellation Load Dispatcher ☑ MCA / Today

E.1.37. **IF** required by Reactor Engineering,
THEN COMMENCE N2-RESP-10, Shutdown Margin Determination,
OTHERWISE MARK this step N/A. MCA / Today

E.1.38. **VERIFY** computer points FWSFE102 and FWSFE103 equal 1.000. MCA / Today

E.1.39. **WHEN** at least one booster pump is in service,
THEN NOTIFY IMD to flush the following sensing lines per N2-OP-3, Attachment 2, Feedwater Flow Venturi Instrument Sensing Line Flush:

- 2FWS-FT1A/B ☑
- 2FWS-PDI1A/B ☑ MCA / Today

E.1.40. **CONFIRM** that delta flows are stable
AND VERIFY the following switches in NORMAL:

- E31A-S5A, RWCU OUTBD DELTA FLOW BYPASS WCS*MOV112, at 2CEC*PNL632 ☑
- E31A-S5B, RWCU INBD DELTA FLOW BYPASS WCS*MOV102, at 2CEC*PNL642 ☑ MCA / Today

E.1.41. **VERIFY** Residual Heat Removal System removed from Shutdown Cooling mode and operable for LPCI mode per N2-OP-31, Residual Heat Removal System. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.1.42. **VERIFY** RHR Shutdown Cooling HIGH REACTOR PRESSURE isolations (Group 5) have been restored per N2-OP-83, Attachment 5, Defeating/Restoring RHR Shutdown Cooling High Reactor Pressure Isolations (Group 5). MCA / Today

E.1.43. **VERIFY** RHR Shutdown Cooling Suction Isolation Valves (Group 5), are energized if required per N2-OP-83, Attachment 6, De-Energizing/Re-Energizing SDC Suction MOVS to Prevent Spurious Isolations. MCA / Today

E.1.44. **VERIFY** the following databases have been reviewed and items within them evaluated for mode change and start-up/mode restraints:

- LCO Tracker ..... ☑
- NewCap ..... ☑ MCA / Today

NOTE

A copy of the Mode 2 required surveillances can be obtained from Outage Management.

E.1.45. **VERIFY ALL** Mode 2 required surveillances are complete. MCA / Today

IVP / Today
IV

E.1.46. **IF** SCRAM timing is required,
THEN VERIFY N2-OSP-RMC-@001 is complete for control rods requiring Scram timing per TS SR 3.1.4.3 and SR 3.1.3.4,
OTHERWISE MARK this step N/A. MCA / Today

E.1.47. **POST** signage to **RESTRICT** Control Room access during Reactivity Manipulations. MCA / Today

E.1.48. Section E.1 reviewed for completeness
AND ALL variances evaluated for applicability during approach to criticality and vessel heatup:

Unit Supervisor	Today Now
Unit Supervisor	Date/Time

Shift Manager	Today Now
Shift Manager	Date/Time

E. STARTUP (Continued) — **Initials/Date**

E.2. Approach to Criticality, Heatup, and Vessel Pressurization

E.2.1. The reactivity team and Control Room staff have been briefed on Attachment 1, Reactor Startup Preparation and Briefing Checklist. — USM / Today (US/SM)

NOTE

If N2-OSP-RMC-Q@003 is **not** current (has **not** been successfully performed in the past 92 days), N2-OSP-RMC-Q@003 must be performed to satisfy TS SR 3.3.2.1.1.

E.2.2. **IF** N2-OSP-RMC-Q@003 is **not** current,
THEN PRIOR TO withdrawing Control Rods for Criticality,
PERFORM N2-OSP-RMC-Q@003, Rod Worth Minimizer Test
OTHERWISE MARK this step N/A. — MCA / Today

E.2.3. **WITHIN** 15 minutes
PRIOR TO control rod withdrawal to bring Reactor to criticality,
VERIFY AND LOG following Reactor coolant parameters are to right of critical line per N2-OSP-RCS-@001, Pressure/Temperature Verification (PTLR-2):

- Reactor coolant pressure ..... ☑
- Reactor coolant temperature ..... ☑ — MCA / Today

E.2.4. **IF** starting up with one recirc loop out of service,
THEN PRIOR TO commencing rod withdrawal
OR recirc flow increase
AND every 15 minutes **UNTIL** thermal power is greater than 30% power and operating loop flow is greater than 50% of rated loop flow,
INITIATE S-OSP-LOG-@001 to record recirc loop differential temperatures (SR 3.4.11.5, SR 3.4.11.6),
OTHERWISE MARK this step N/A. — MCA / Today

E.2.5. **VERIFY ALL** Control Rods are fully inserted. (SR 3.1.6.1) **[T/S] [C26]** — MCA / Today

IVP / Today (IV)

E.2.6. **VERIFY** Rod Worth Minimizer in service per N2-OP-95A, Rod Worth Minimizer. — MCA / Today

E.2.7. **IF** Rod Worth Minimizer is INOP,
THEN VERIFY N2-OP-95A, Section H, RWM Bypass conditions met,
OTHERWISE MARK this step N/A. — MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

Within 72 hours after entering MODE 2, Drywell and Suppression Chamber atmosphere oxygen concentration must be less than 4% by volume. **[C20]**

E.2.8. **PLACE** Reactor Mode Switch in START/HOT STANDBY position. MCA / Today

E.2.9. **VERIFY** Gaitronics in merge
AND ANNOUNCE:

"Attention all personnel, Unit 2 has entered Mode 2, Startup." MCA / Today

E.2.10. At 2CEC*PNL608, **VERIFY** the following on APRM 1-4: **[C16]**

a. SETDOWN is displayed:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA / Today

b. The double upward pointing triangles on FLUX bar graph are aligned to approximately 15%:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA / Today

c. The double upward pointing triangles on STP bar graph are aligned to approximately (0.55 Wd + 57.5), with a maximum of less than or equal to 113.5%:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA / Today

E.2.11. **VERIFY** annunciator 603442, CONTROL ROD OUT BLOCK, is clear. MCA / Today

E.2.12. **PLACE** SRM recorder speed selection switch in FAST (second speed) position. MCA / Today

E.2.13. **PLACE** speed selection switch for at least one IRM recorder in each Reactor Protection System in FAST (second speed) position. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.14. **PRIOR TO** commencing rod withdrawal for Reactor startup, **PERFORM** the following actions:

a. **VERIFY** IRM range switches in Position 1. MCA / Today

b. **RECORD** Range 1 **OR** 2 IRM readings indicated on recorders C51-R603A, B, C, and D
AND VERIFY IRM readings are less than 10 on recorder:

	READING	RANGE		READING	RANGE
IRM A	0	1	IRM B	0	1
IRM C	0	1	IRM D	0	1
IRM E	0	1	IRM F	0	1
IRM G	0	1	IRM H	0	1

MCA / Today

c. **IF** any IRM is reading greater than 10 on range 1 **OR** 2,
THEN NOTIFY SM
AND REQUEST IMD assistance,
OTHERWISE MARK this step N/A. MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

- Criticality should be expected at any time. Close attention to all indications of neutron flux and period is necessary. The point of criticality may be impacted by control rod withdrawal, changing reactor coolant temperature or changing Xenon concentrations.
- When operating beyond mid cycle exposure, the moderator temperature coefficient (MTC) may turn positive resulting in a shortened period during the heatup. Increasing moderator temperature during approach to criticality can pull up the expected ECP when operating with a +MTC. **[C18]**
- A +MTC can be observed as a period that gradually becomes shorter after the initial criticality, without any further rod withdrawals. If a +MTC is observed, the period should be monitored such that IRM range changes remain controllable. Additionally, heatup rate should be monitored to remain less than 20F every 15 minutes. If period or heatup rate become excessive, single notch insertion may be used, backwards in the sequence, to maintain the reactor critical on a controllable period. **[C18]**

CAUTION

When approaching criticality, Operators shall be guided by redundant or corroborating instrumentation (such as, SRMs, IRMs, and period meters) when available, and in the absence of other definite operational evidence, they shall always believe instrument indications. **[C2] [C5] [C7]**

E.2.15. **PERFORM** the following:

a. **BEGIN** Control Rod withdrawal per verified copy of approved Startup Control Rod Sequence
UNTIL Reactor criticality is indicated by the following:

- Rising neutron count rate
- **No** control rod motion
- Constant positive period

MCA /Today

b. **DIRECT** CRO to make a CRD Deficiency Log entry per N2-OP-30, Subsection F, Documenting Control Rod Drive Operational Problems, at any time during performance of this procedure when a Control Rod Drive Mechanism does **not** perform as expected.

MCA /Today

c. **MONITOR** the following parameters during each reactivity manipulation:

- Redundant Nuclear Instrumentation
- Core Flow
- Reactor Pressure
- Reactor Temperature

MCA /Today

F. STARTUP (Continued) **Initials/Date**

NOTE

SRMs are to be withdrawn such that noise from induced fluctuations on IRMs **cannot** result in FULL Scram. **[C25]**

E.2.16. **MAINTAIN** a count rate between 10^2 and 10^5 CPS by **WITHDRAWING** SRMs 'A' **AND** 'C' as required. **[C25]** MCA / Today

E.2.17. **MAINTAIN** a count rate between 10^2 and 10^5 CPS by **WITHDRAWING** SRMs 'B' and 'D' as required. **[C25]** MCA / Today

E.2.18. **WHEN** Reactor criticality is indicated,
THEN VERIFY Gaitronics in Merge
AND ANNOUNCE the following:

"Attention all personnel the Unit 2 Reactor is critical." MCA / Today

E.2.19. **RECORD** Critical Data below:

- Time Criticality was achieved: This Morning
- Rod Number: XX-XX
- RWM Step: Step X
- Rod Position: 14
- Reactor Period (1.44 x Doubling Time): In Spec. Doubling Time
- Reactor Water Temperature: In Spec. Temp
- Operating Recirc Pump Suction Temperature: In Spec. Temp
- Person Pulling Critical: MCA

MCA / Today

E.2.20. **LOG** critical data in Control Room Log. MCA / Today

E.2.21. **IF** reactor goes subcritical,
THEN REVIEW flowchart in Attachment 9, to determine what action is required,
OTHERWISE MARK this step N/A. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.22. **IF** Reactor goes RE-CRITICAL un-intentionally
(Reactor from a SUBCRITICAL condition to a CRITICAL condition),
OR Rod insertion **cannot** overcome the positive reactivity insertion due to moderator temperature changes
AND/OR Xenon changes due to low rod worth
OR other equipment issues,
THEN SCRAM reactor,
OTHERWISE MARK this step N/A. MCA / Today

E.2.23. **ESTABLISH** a stable positive period between 100 and 300 seconds using control rods as required. MCA / Today

E.2.24. **IF** reactor period is less than or equal to 300 seconds,
THEN do **not** withdraw rods between criticality and the POAH unless:

- Predicted rod withdrawal will **not** exceed 100 second period.
- SM Permission is obtained,

OTHERWISE MARK this step N/A. MCA / Today

E.2.25. **IF** a positive period of less than 50 seconds is experienced,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **INSERT** the last control rod which was withdrawn to its previous position ..... ☑

b. **NOTIFY** the US/SM..... ☑

c. **REQUEST** further direction from Reactor Engineering..... ☑ MCA / Today

E.2.26. **IF** Reactor startup is following operating cycle outage,
THEN CONFIRM with Reactor Engineering that shutdown margin procedures are being performed, as required,
OTHERWISE MARK this step N/A. MCA / Today

E.2.27. **IF** criticality is obtained outside the estimated critical acceptance criteria band at beginning of cycle,
THEN PRIOR TO continuing reactor startup,
RETURN to Group 1 hold point
AND CONSULT with Nuclear Fuels BWR design group,
OTHERWISE MARK this step N/A. MCA / Today

E.2.28. **IF** criticality is obtained outside the estimated critical acceptance criteria band **AFTER** beginning of cycle,
THEN INITIATE an IR
AND CONTINUE reactor startup,
OTHERWISE MARK this step N/A. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.29. **UPDATE** Safety Parameter Display System (SPDS) per N2-OP-91B, Safety Parameter Display System (SPDS). MCA / Today

NOTE

Failure of the following steps credited for NRC commitment to manage the effects of aging require generation of an IR for evaluation by a qualified individual to ensure actions are taken to evaluate and mitigate the effects of aging on cables and connections prior to the loss of the cable systems intended function. **[C22]**

E.2.30. **PRIOR TO** fully withdrawing SRM detectors, **VERIFY** proper SRM/IRM overlap as follows: **[C22]**

a. **RECORD** IRM readings from P606 and P633:

	READING	RANGE		READING	RANGE
IRM A:	Acceptable	Acceptable	IRM B:	Acceptable	Acceptable
IRM C:	Acceptable	Acceptable	IRM D:	Acceptable	Acceptable
IRM E:	Acceptable	Acceptable	IRM F:	Acceptable	Acceptable
IRM G:	Acceptable	Acceptable	IRM H:	Acceptable	Acceptable

MCA / Today

b. **RECORD** the following indications using SRM monitors at P606 and P633: **[T/S]**

SRM CPS		DETECTOR POSITION	Yes	No
SRM A: Good	cps	SRM A FULL OUT	☐	☑
SRM B: Good	cps	SRM B FULL OUT	☐	☑
SRM C: Good	cps	SRM C FULL OUT	☐	☑
SRM D: Good	cps	SRM D FULL OUT	☐	☑

Acceptance Criteria:

- SRM detectors **not** fully withdrawn
- IRM reading greater than or equal to 15 on Range 1 **OR** 2
- SRM reading less than or equal to 1 x 10^5 cps

CHANNEL	SAT	UNSAT	CHANNEL	SAT	UNSAT
IRM A	☑	☐	IRM B	☑	☐
IRM C	☑	☐	IRM D	☑	☐
IRM E	☑	☐	IRM F	☑	☐
IRM G	☑	☐	IRM H	☑	☐

MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.30 (Continued)

c. **IF** Acceptance Criteria is exceeded for any channel above,
THEN immediately **NOTIFY** SM,
OTHERWISE MARK this step N/A. MCA / Today

d. **IF** proper SRM/IRM overlap **cannot** be verified,
THEN generate an IR for evaluation,
OTHERWISE MARK this step N/A. MCA / Today

E.2.31. **MAINTAIN** IRM indications between 25 and 75 on 125% scale by ranging IRM range switches as required. MCA / Today

E.2.32. **PLACE** SRM recorder speed selection switch in Slow (normal speed) position. MCA / Today

E.2.33. At least every 30 minutes during heatup,
LOG Reactor coolant temperature and pressure per N2-OSP-RCS-@001, RCS Pressure/Temperature Verification. MCA / Today

NOTE

- Identify method(s) for pressure control to be used near the POAH other than control rods insertions.
- Reactivity team must be informed of any activity that will change steam demand. As steam demand increases, control rods will need to be withdrawn to maintain the desired pressurization rate.

CAUTION

- When beyond mid cycle exposure, moderator temperature increase may result in a shortened reactor period because of the potential for a positive moderator temperature coefficient. **[C18]**
- Failure to maintain vessel heatup and cooldown rates less than 100 °F/hr, will violate the requirements of the Pressure and Temperature Limits Report (PTLR).

E.2.34. **WHEN** Reactor Vessel Temperature begins to rise as part of plant startup,
THEN DOCUMENT on Attachment 8, IRM Range at the Point of Adding Heat,
AND POST on P603 console for reference. MCA / Today

E.2.35. **WHEN** heating range is reached, as evidenced by reduction in IRM indication,
THEN POSITION control rods to maintain a heatup rate less than 100 °F in any one-hour period. MCA / Today

E. STARTUP (Continued) **Initials/Date**

CAUTION

Reactor heatup from approximately 200 °F to 350 °F (1 to 150 psig) may be limited by reject flow capacity if WCS-P1B is in service. Failure to limit heatup to approximately 40 °F/hour until Reactor Pressure greater than 150 psig could result in a high RPV level condition. Careful monitoring of RPV level at heatup rates greater than 40 °F/hr is warranted. This limit does **not** apply with WCS-P1A in service.

E.2.36. **PRIOR TO** RPV temperature exceeding 285 °F,
VERIFY dissolved oxygen is less than 300 ppb. MCA / Today

NOTE

Dissolved Oxygen concentrations >300 ppb does **not** Hold up Power Ascension

E.2.37. **IF** Dissolved Oxygen is >300 ppb,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

- **TAKE ACTIONS** to correct per CY-AB-120-100 ..... ☑
- **NOTIFY** Chemistry..... ☑
- **GENERATE** an IR ..... ☑ MCA / Today

E.2.38. **PRIOR TO** RPV temperature reaching 325 °F,
VERIFY Reactor Water Cleanup System is lined up per N2-OP-37, Section F, WCS Operation to Reduce Feedwater Stratification with Return to a Single Feedwater Line,
OR Section F, WCS Operation in Full Reject to Reduce Feedwater Stratification. MCA / Today

CAUTION

- If any steam line paths (including drains) are open before the Condensate and Condensate Booster Pumps are online at 212 °F or above, the steam flow may exceed CRD cooling flow which could result in a low-level Reactor SCRAM.

E.2.39. **MAINTAIN** Reactor Water Level between 178.3" and 187.3" by **REJECTING** water as necessary using WCS system per N2-OP-37, Reactor Water Cleanup System. MCA / Today

E.2.40. **VERIFY** RPV Instrument Reference Leg Backfill Station Flow Rates have been **ADJUSTED** per N2-OP-34 for Mode 1, 2, 3 Operation. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.41. **PERFORM** RCIC warmup preparations per N2-OP-35, Steps E.1.1 through E.1.4.4. MCA / Today

> **NOTE**
>
> It is recommended that Suppression Chamber and Drywell pressure be maintained less than 0.41 psig in order to meet purge prerequisites.

E.2.42. **DURING** reactor heatup, **PERFORM** the following:

- **MAINTAIN** drywell average temperature less than 150 °F per N2-OP-60, Drywell Cooling. ..... ☑
- **MAINTAIN** drywell pressure between −0.5 and +0.75 psig per N2-OP-61A, Primary Containment Ventilation/Purge & Nitrogen System, for drywell venting. ..... ☑ MCA / Today

E.2.43. **WHEN** Reactor pressure reaches 5 psig,
THEN PERFORM the following:

> **NOTE**
>
> Small perturbations in RPV level may occur when MSIVs are opened.

a. **VERIFY** Inboard **AND** Outboard MSIVs open per N2-OP-1, Main Steam System:

2MSS*AOV6A	☑	2MSS*AOV7A	☑
2MSS*AOV6B	☑	2MSS*AOV7B	☑
2MSS*AOV6C	☑	2MSS*AOV7C	☑
2MSS*AOV6D	☑	2MSS*AOV7D	☑

MCA / Today

b. **CLOSE** the following:

- 2MSS*MOV118, Reactor Vessel Vent Throttle ..... ☑
- 2MSS*MOV119, Reactor Vessel Vent Throttle ..... ☑ MCA / Today

c. **OPEN** 2MSS*MOV108, Reactor Head Vent to Main Steam Line. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.43(Continued)

d. **VERIFY** the following valves per N2-OP-1, Main Steam System OPEN:

- MSS-AOV87A, MSL Low Pt Drain Isol..... ☑
- MSS-AOV87B, MSL Low Pt Drain Isol..... ☑
- MSS-AOV87C, MSL Low Pt Drain Isol ..... ☑
- MSS-AOV87D, MSL Low Pt Drain Isol ..... ☑
- MSS-AOV88A, MSL Header Drain ..... ☑
- MSS-AOV88B, MSL Header Drain ..... ☑ MCA /Today

e. **IF** required,
THEN PERFORM N2-OSP-MSS-002,
OTHERWISE MARK this step N/A. MCA /Today

f. **VERIFY** 2ASS-MOV148, Main Steam to Aux Steam Block Vlv open. MCA /Today

g. **VERIFY** HWC H2
AND O2 piping is pressurized per N2-OP-112, Sections E.1, Startup from Isolated Conditions through E.3, Pressurization of Oxygen Piping. MCA /Today

NOTE

Notify Rad Protection of intent to use Main Steam to Supply Sealing Steam

h. **IF** Main Turbine Seals are Lined up to Auxiliary Steam,
AND Reactor Pressure is approximately 40 psig,
THEN ESTABLISH Main Turbine Seals using Main Steam, per N2-OP-25 Section E.2.0.
OTHERWISE MARK this step N/A. MCA /Today

i. **WHEN** Reactor pressure exceeds 50 psig,
THEN VERIFY a Turbine Bypass Valve opens to regulate Reactor pressure. MCA /Today

j. **WHEN** Turbine Bypass Valve opens,
THEN INITIATE Condenser Neck Sprays per N2-OP-3, Section F., Condenser Neck Spray Operation. MCA /Today

E. STARTUP (Continued) **Initials/Date**

E.2.43(Continued)

NOTE

- Reactor Pressure is raised by adjusting PRESSURE SET with Section E.9, "Pressure Adjustments using Automatic Heatup Rate" (preferred) or Section E.10, "Pressure Adjustments using Manual Rate".
- The use of the next step for raising pressure (heatup) may result in changes to main turbine bypass valve position and changes in reactor pressure, level, and power.

k. **RAISE** Reactor pressure to 150 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10.

MCA / Today

CAUTION

During shell warming, exceeding 100 PSIA first stage pressure may result in a Reactor SCRAM. Particular attention is required if shell warming is being performed in conjunction with raising Reactor pressure.

E.2.44. **WHEN** Reactor pressure is approximately 50 psig,
THEN PERFORM the following:

a. **COMMENCE** Turbine Shell/Chest warming per N2-OP-21, Main Turbine System.

MCA / Today

b. **WHILE CONTINUING**,
CONCURRENTLY PERFORM the following:

- **PERFORM** Main Generator and Main Transformer startup checks per N2-OP-68 Subsections E.1.0 through E.3.4
AND Exciter Startup preparations per N2-OP-68A..... ☑
- **PERFORM** N2-OP-71A Section F.1.0, Preparing 2STX-XNS1 Normal Station Service Transformer for Service..... ☑
- **PERFORM** Turning Gear and Turbine Lube Oil System Operational Checks per N2-OP-22A, F.1.0. ..... ☑
- Per Radiation Protection for access to High Pressure Hood Locked High Rad Area,
CHECK Turbine Bearing Oil Flow by
PERFORMING a local visual inspection at each of the ten Bearing Oil Inspection Caps..... ☑ MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.44(Continued)

c. **WHEN** DIV I **AND** DIV II Low Pressure Isolation signals have cleared,
THEN PLACE RCIC in STANDBY per N2-OP-35, Reactor Core Isolation Cooling. MCA / Today

NOTE

Prior to exceeding 150 psig, RCIC shall be lined up in the standby condition per N2-OP-35, Reactor Core Isolation Cooling.

E.2.45. **PRIOR TO** RPV pressure reaching 150 psig, **PERFORM** the following:

a. **AFTER** reaching 100 psig
AND PRIOR TO exceeding 150 psig,
FILL RCIC piping per N2-OP-35, Section H,
RCIC Manual Keep Fill of Piping Downstream of MOV126. MCA / Today

b. **IF** RPV pressure approaches 140 psig and RCIC is **not** yet operable,
THEN PERFORM the following:
OTHERWISE MARK these steps N/A:

1) **HOLD** pressure ascension
AND CONTINUE control rod withdrawal
UNTIL at least one BPV partially open (approximately 50%). . ☑

2) **WHEN** RCIC is operable,
THEN CONTINUE pressure ascension. **[C26]**..... ☑ MCA / Today

IVP / Today
IV

c. **PERFORM** N2-OSP-ICS-M001, RCIC System Piping Fill and Valve Lineup Verification Test. MCA / Today

d. **PERFORM** required valve testing per N2-OSP-ICS-Q@002,
OTHERWISE MARK this step N/A. MCA / Today

e. **NOTIFY** Chemistry to sample Drywell and Suppression Chamber for purge/vent acceptance criteria,
OTHERWISE MARK this step N/A. MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

- N2-OSP-ICS-R002, RCIC System Flow Test, is required if plant startup is subsequent to end of operating cycle outage or as PMT.
- Reactor pressure between 150 and 165 psig will be maintained during performance of N2-OSP-ICS-R002, RCIC System Flow Test.

E.2.46. **WHEN** RPV pressure is approximately 150 psig,
THEN PERFORM the following:

a. **PREPARE** Offgas system and Steam Jet Air Ejectors (SJAE) for service as follows:

1) **PERFORM** N2-OP-42, E.4.1 through E.4.19 to prepare Offgas system for use. MCA / Today

2) **PERFORM** SJAE steam line warmup per N2-OP-9, Steps E.2.1 through E.2.16. MCA / Today

b. **IF** required,
THEN PERFORM N2-OSP-ICS-R002, RCIC System Flow Test,
OTHERWISE MARK this step N/A. MCA / Today

NOTE

- Suppression Chamber pressure and Drywell pressure should be maintained less than 0.41 psig in order to meet purge prerequisite.
- To minimize the chance of lifting a Drywell Vacuum Breaker, Suppression Chamber pressure should be maintained less than or equal to Drywell pressure.

E.2.47. **IF** Drywell **OR** Suppression Chamber pressure is greater than or equal to 0.41 psig,
THEN PERFORM containment venting per N2-OP-61A, Primary Containment Ventilation, Purge and Nitrogen System
UNTIL Drywell and Suppression Chamber pressure are less than or equal to 0.41 psig,
OTHERWISE MARK this step N/A. MCA / Today

E.2.48. **RAISE** Reactor pressure to 250 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10. MCA / Today

F. STARTUP (Continued) **Initials/Date**

E.2.49. **IF** a Reactor pressure of 250 psig **cannot** be obtained
WHILE MAINTAINING at least one BPV partially open (approximately 50% open),
THEN HOLD pressure ascension
AND CONTINUE control rod withdrawal
UNTIL sufficient BPV opening is obtained.
OTHERWISE MARK this step N/A.

MCA / Today

CAUTION

When controlling RPV water level using 2CNM-LV137, steam loads and water level should be monitored to ensure capacity of 2CNM-LV137 is **not** exceeded. 2CNM-LV137 should be less than 70% open prior to a Feed Pump start. **[C1]**

E.2.50. **WHEN** RPV pressure is approximately 250 psig,
THEN PERFORM the following:

a. Valve lineups and prestart checks necessary to start a Reactor Feed pump.

MCA / Today

b. **If** 2CNM-LV137 is more than 70% open,
THEN AUGMENT flow per N2-OP-3, Section H.3.0,
OTHERWISE MARK this step N/A.

MCA / Today

CAUTION

The first feedwater pump should be started as close to 480 psig Reactor pressure as possible to allow 2CNM-LV137 to pass enough flow on initial feedwater pump start to maintain Reactor Level while allowing 2FWS-LV55A(B) to open against the differential pressure across the valve.

E.2.51. **RAISE** Reactor pressure to 500 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10

MCA / Today

E.2.52. **IF** a Reactor pressure of 500 psig **cannot** be obtained
WHILE MAINTAINING at least one BPV partially open (approximately 50% open),
THEN HOLD pressure ascension
AND CONTINUE control rod withdrawal
UNTIL sufficient BPV opening is obtained,
OTHERWISE MARK this step N/A.

MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.53. **WHEN** RPV pressure is approximately 500 psig,
THEN PERFORM the following:

a. **PLACE** SJAEs into service per N2-OP-9, Section E, Placing the Steam Jet Air Ejectors (SJAEs) In Service. MCA / Today

b. **PLACE** the Offgas system into service per N2-OP-42, Section E, OFG System Startup. MCA / Today

c. **START** a Feedwater Pump per N2-OP-3, Condensate and Feedwater System, Section E, Initial Feedwater Pump Startup. MCA / Today

d. **VERIFY** HWC System is ready to be placed in service per N2-OP-112, Hydrogen Water Chemistry System. MCA / Today

E.2.54. **RAISE** Reactor pressure to 600 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10. MCA / Today

E.2.55. **IF** a Reactor pressure of 600 psig **cannot** be obtained
WHILE MAINTAINING at least one BPV partially open (approximately 50% open),
THEN HOLD pressure ascension
AND CONTINUE control rod withdrawal
UNTIL sufficient BPV opening is obtained,
OTHERWISE MARK this step N/A. MCA / Today

E.2.56. **PRIOR TO** RPV pressure exceeding 600 psig,
PERFORM the following:

a. **RESET** HPCS Reactor high water level at Panel P601. MCA / Today

b. **OBSERVE** HPCS reactor high level seal-in white light extinguishes. MCA / Today

F. STARTUP (Continued) **Initials/Date**

CAUTION

Failure to place the Reactor Mode Switch in RUN position before exceeding 11% on APRM will cause control rod block. Exceeding 14% will cause Reactor SCRAM. Actual set points are set 1% conservative to nominal trip set points due to instrument tolerances.

E.2.57. **WHEN** Reactor power is about 2 to 3%,
THEN PERFORM the following: **[C5]**

a. **VERIFY** APRMs come on scale during power rise:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA / Today

b. **IF** IRM range correlation is required,
THEN NOTIFY IMD Supervision to
PERFORM required IRM range correlations,
OTHERWISE MARK this step N/A:

Person Contacted
Person Contacted

MCA / Today

E.2.58. **RAISE** Reactor pressure to 900 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10.

MCA / Today

E.2.59. **IF** a Reactor pressure of 900 psig **cannot** be obtained
WHILE MAINTAINING at least one BPV partially open (approximately 50% open),
THEN HOLD pressure ascension
AND CONTINUE control rod withdrawal
UNTIL sufficient BPV opening is obtained,
OTHERWISE MARK this step N/A.

MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.60. **WHEN** Reactor pressure is greater than 821 psig,
THEN PERFORM the following:

a. **VERIFY** the following annunciators are clear:

- 603127, DIVISION I MN STEAM LINE PRESSURE LOW ..... ☑
- 603427, DIVISION II MN STEAM LINE PRESSURE LOW ..... ☑

MCA / Today

NOTE

The following step in **not** required if plant startup is from a planned or forced outage, **no** rise in containment leakage was observed prior to or during plant shutdown, **no** work was performed requiring a 900 psig PMT and **no** low-level RCS Pump Oil alarms (602123, 603124) were received.

b. **IF** DRYWELL 900 PSIG INSPECTION IS REQUIRED
THEN VERIFY Drywell 900 psig inspection team has been dispatched (Section E.8 **AND** Attachment 2),
OTHERWISE MARK this step N/A.

MCA / Today

NOTE

Prior to exceeding 5% power as indicated by APRM, Condenser vacuum shall be established and maintained by SJAEs and condenser air removal pumps secured.

E.2.61. **PRIOR TO** Reactor Power exceeding 5%, **PERFORM** the following:

a. **VERIFY** Steam Jet Air Ejectors in service per N2-OP-9, Condenser Air Removal.

MCA / Today

NOTE

For the first startup following a refueling outage, PMST credit needs to be taken for the following step. A work order should exist for taking this credit.

b. **WHEN** Condenser Air Removal Pumps have been secured,
THEN CONFIRM 2ARC-AOV105, AIR REMOVAL PMPS INLET ISOL VALVE, has closed.

MCA / Today

NOTE

The OFG charcoal adsorbers should be placed in service as OFG system parameters allow to prevent unnecessary release to the main stack.

c. **VERIFY** Off-gas in service per N2-OP-42, Off-gas System.

MCA / Today

F. STARTUP (Continued) **Initials/Date**

E.2.62. **PLACE** HWC System in service per N2-OP-112, Hydrogen Water Chemistry System. MCA / Today

E.2.63. **RAISE** Reactor pressure to 925 psig
WHILE MAINTAINING an approximate heatup rate of < 100 °F/hr using Section E.9 **OR** Section E.10. MCA / Today

E.2.64. **IF** a Reactor pressure of 925 psig **cannot** be obtained
WHILE MAINTAINING at least one BPV partially open (approximately 50% open),
THEN HOLD pressure ascension
AND CONTINUE control rod withdrawal
UNTIL sufficient BPV opening is obtained,
OTHERWISE MARK this step N/A. MCA / Today

E.2.65. **WHEN** Reactor pressure reaches approximately 925 psig,
THEN PERFORM the following:

a. **SELECT** MANUAL RATE
AND VERIFY Turbine Bypass valves maintain stable Reactor pressure. MCA / Today

b. IPTE portion of plant startup is now complete,
LOG completion in Control Room Log. MCA / Today

c. **WHEN** Reactor pressure is approximately 925 psig,
AND one Turbine Bypass valve is open 20% **OR** greater,
THEN CHANGE Reboiler Steam Supply to Main Steam per N2-OP-25, Auxiliary Steam, Auxiliary Condensate and Gland Seal System. ____ / ____

NOTE

Withdrawal of SRMs prematurely can cause numerous alarms and significantly delay control rod operations. The SM/US may choose to defer this step until after mode switch to RUN.

d. **VERIFY** SRM detectors fully withdrawn:

- SRM A..... ☑
- SRM B..... ☑
- SRM C..... ☑
- SRM D..... ☑ MCA / Today

e. At P851, **PLACE** 2TME-AOV121, GLAND SEAL STM MAIN STM SUPPLY VLV, control switch in AUTO. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.65 (Continued)

f. **IF** any VT-2 exams are required,
THEN DIRECT the OCC to **NOTIFY** QA to **PERFORM** required VT-2 exams,
OTHERWISE MARK this step N/A. MCA / Today

E.2.66. **WHEN** Main Condenser vacuum is greater than 25" Hg (annunciators 603128 and 603428 clear),
THEN PERFORM the following:

a. At H13-P609,
PLACE the following keylock switches to NORMAL position:

- B22H-S24A, CNSR LOW VAC BYPASS ..... ☑
- B22H-S24C, CNSR LOW VAC BYPASS ..... ☑ MCA / Today

b. At H13-P611,
PLACE the following keylock switches to NORMAL position:

- B22H-S24B, CNSR LOW VAC BYPASS ..... ☑
- B22H-S24D, CNSR LOW VAC BYPASS ..... ☑ MCA / Today

c. **CONFIRM** the following annunciators remain clear:

- 603128, DIVISION I CONDENSER VACUUM LOW. ..... ☑
- 603428, DIVISION IICONDENSER VACUUM LOW. ..... ☑ MCA / Today

d. **WHEN** vacuum is stable,
THEN CONTROL vacuum between 26.5 and 28 in. Hg. per N2-OP-9, Section H, Main Condenser Vacuum Control for Potential Rub Mitigation. MCA / Today

E.2.67. **WHEN** power is between 5 to 10%,
THEN PLACE Oxygen Feedwater Injection System in service per N2-OP-3, Section E., Oxygen In Feedwater Injection (OFI) System Startup. MCA / Today

E.2.68. **IF** Drywell entry has been made,
THEN VERIFY final Drywell inspection completed per Section E.8,
OTHERWISE MARK this step N/A. MCA / Today

E.2.69. **VERIFY** Primary Containment AC Circuits are de-energized per Attachment 2, Primary Containment AC Circuits. MCA / Today

E.2.70. **PERFORM** Appendix R valve/switch lineup per Attachment 4, Appendix R Valve/Switch Lineup. MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.2.71. **IF** Drywell **OR** Suppression Chamber entry was performed
OR containment was purged,
THEN PERFORM the following
AFTER final Drywell inspection and closure is complete **AND** satisfactory,
OTHERWISE MARK these steps N/A:

a. **VERIFY** Instrument Nitrogen System is supplied by Nitrogen per N2-OP-61A, Primary Containment Ventilation/Purge and Nitrogen System, Section E., Supplying Nitrogen Gas to Instrument Nitrogen System, and Section F., Transfer from Instrument Air to Instrument Nitrogen Supply. MCA / Today

b. **BEGIN INERTING** Primary Containment per N2-OP-61A, Primary Containment Ventilation Purge and Nitrogen System,
UNTIL Oxygen concentration is less than 4% by volume. MCA / Today

E. STARTUP (Continued) — **Initials/Date**

E.3. Transfer of Mode Switch to RUN

E.3.1. **PLACE** Condensate demineralizers in service as required in order recommended by Chemistry. MCA /Today

E.3.2. **MAINTAIN** individual demineralizer flows between 2000 and 3300 gpm and a system delta-P less than or equal to 55 psid (60 psid maximum). MCA /Today

E.3.3. **VERIFY** DEHC pressure regulator controlling Bypass valves to regulate pressure as required. **[C24]** MCA /Today

E.3.4. **CONTINUE** to **WITHDRAW** control rods: next target: APRM downscale lights have cleared and less than 11% power. **[C5]** MCA /Today

E.3.5. **VERIFY** APRMs reading greater than 5% by placing IRM/APRM recorder select switch to APRM:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA /Today

E.3.6. **IN EACH** Reactor Protection Division
LEAVE one IRM/APRM recorder select switch **PLACED** in APRM position:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA /Today

E.3.7. **CONFIRM** the following chemistry parameters:

- Operational Condition 1 per TRM Table T3.4.1-1..... ☑
- Reactor coolant DOSE EQUIVALENT I - 131 specific activity is less than or equal to 0.2 µCi/gm..... ☑
- CDE and Feedwater dissolved oxygen are greater than or equal to 30 ppb and less than or equal to 200 ppb (CY-AB-120-100)..... ☑ MCA /Today

E.3.8. **PERFORM** APRM gain adjustments per N2-OSP-NMS-@004, APRM Gain Adjustment, at intervals recommended by Reactor Engineering. MCA /Today

E. STARTUP (Continued) **Initials/Date**

E.3.9. **VERIFY** the following:

- Steam Line pressure is greater than 821 psig ..... ☑
- Annunciator 603127, DIVISION I MN STEAM LINE PRESSURE LOW, is clear. ..... ☑
- Annunciator 603427, DIVISION II MN STEAM LINE PRESSURE LOW, is clear ..... ☑
- APRMs on scale between 5% and 11% ..... ☑
- MSIVs are open ..... ☑
- Condenser vacuum greater than 25" Hg **[C5]** ..... ☑
- **No** Mode 1 restrictions exist ..... ☑

a. At 2CEC*PNL873, OVERRIDE Switches to RESET:

1) UNIT COOLER FANS GR1 LOCA ..... ☑
2) UNIT COOLER FANS GR2 LOCA ..... ☑
3) DRYWELL UNIT COOLER WTR. DIV I LOCA ..... ☑
4) DRYWELL UNIT COOLER WTR. DIV II LOCA ..... ☑
5) (DIV I CMS) ISOL VLV ..... ☑
6) (DIV I HCS) LOCA ..... ☑
7) PURGE OUTBD VALVES ..... ☑

b. At 2CEC*PNL875, OVERRIDE Switches to RESET:

1) (DIV II CMS) ISOL VLV ..... ☑
2) (DIV II HCS) LOCA ..... ☑
3) PURGE INBOARD VLVS ..... ☑
4) At Panel 2CEC*PNL625, HPCS INJ VLV LEVEL 8 BYPASS TEST SWITCH in NORMAL ..... ☑ MCA / Today

E.3.10. **REVIEW** Section E.2 and Steps E.3.1 through E.3.9 for completeness **AND EVALUATE** all variances for applicability during Mode Switch transfer to RUN:

Unit Supervisor _____ Today/Now _____
Unit Supervisor Date/Time

Shift Manager _____ Today/Now _____
Shift Manager Date/Time

E. STARTUP (Continued) — **Initials/Date**

E.3.11. **PLACE** Reactor Mode switch in **RUN** position. MCA /Today

E.3.12. **VERIFY** Gaitronics in Merge **AND ANNOUNCE**:

"Attention all personnel, Unit 2 has entered Mode 1." MCA /Today

E.3.13. At 2CEC*PNL608, **VERIFY** the following on APRM 1-4: **[C5] [C16]**

a. RUN MODE is displayed:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA /Today

b. The double upward pointing triangles on FLUX bar graph are aligned to approximately 118%:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA /Today

c. The double upward pointing triangles on STP bar graph are aligned to approximately (0.55 Wd + 57.5), with a maximum of less than or equal to 113.5%:

APRM 1	☑	APRM 3	☑
APRM 2	☑	APRM 4	☑

MCA /Today

E.3.14. **UPDATE** SPDS per N2-OP-91B, Safety Parameter Display System (SPDS).

MCA /Today

E. STARTUP (Continued) **Initials/Date**

E.3.15. **PERFORM** the following:

a. **SELECT** APRM position on **ALL** IRM/APRM recorders:

- R603A-2NMPN01 ..... ☑
- R603B-2NMPN01 ..... ☑
- R603C-2NMPN01 ..... ☑
- R603D-2NMPN01 ..... ☑ MCA / Today

b. **SELECT** RBM position on BOTH IRM/RBM recorders:

- R603C-2NMPN01 ..... ☑
- R603D-2NMPN01 ..... ☑ MCA / Today

E.3.16. **FULLY WITHDRAW** IRMs from core per N2-OP-92, Section E, APRM/RBM Startup:

IRM A	☑	IRM E	☑
IRM B	☑	IRM F	☑
IRM C	☑	IRM G	☑
IRM D	☑	IRM H	☑

MCA / Today

E.3.17. **VERIFY ALL** IRM recorders in SLOW (normal) speed:

- R603A-2NMPN01 ..... ☑
- R603B-2NMPN01 ..... ☑
- R603C-2NMPN01 ..... ☑
- R603D-2NMPN01 ..... ☑ MCA / Today

E. STARTUP (Continued) **Initials/Date**

NOTE

When reactor pressure exceeds 935 psig with 1 turbine bypass valve full open and N2-OSP-ICS-Q@002 is required, a 12-hour clock for test performance will start. (SR 3.5.3.3)

E.3.18. **IF** N2-OSP-ICS-Q@002 is required,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. **SLOWLY RAISE** DEHC pressure setpoint to between 935 and 945 psig MCA / Today

b. **PERFORM** N2-OSP-ICS-Q@002, RCIC Pump and Valve Operability Test and System Integrity Test,
WITHIN 12 hours
AFTER Reactor pressure is between 935 and 1035 psig with one Turbine Bypass Valve full open. MCA / Today

E.3.19. **SLOWLY RAISE** DEHC pressure setpoint to 935 psig per Section E.10. MCA / Today

CAUTION

Observe caution when opening 2FWS-MOV47A, B, C valves. Leakage past the LV10s can cause level and associated power excursions particularly at lower reactor pressures.

E.3.20. **VERIFY OPEN** discharge MOV associated with non-operating feedwater pumps. MCA / Today

NOTE

When reactor pressure exceeds 950 psig with 2 turbine bypass valves full open and N2-OSP-ADS-R001 is required, a 12-hour clock for test performance will start. (SR 3.5.1.7)

E.3.21. **CONTINUE** to **WITHDRAW** control rods: next target: 1½ to 2 bypass valves open to support turbine roll. MCA / Today

E.3.22. **IF** N2-OSP-ADS-R001is required to be performed,
THEN WITHIN 12 hours
AFTER reaching a reactor pressure of 950 psig
AND at least two Turbine Bypass valves open
AND PRIOR TO exceeding 976 psig,
PERFORM N2-OSP-ADS-R001,
OTHERWISE MARK this step N/A. MCA / Today

F. STARTUP (Continued) **Initials/Date**

CAUTION

A differential pressure greater than the engineering defined limits will be encountered if Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig.

F.3.23. **PRIOR TO** Reactor pressure exceeding 950 psig, Close Main Steam Line Drains per N2-OP-1, Main Steam System. MCA / Today

F.3.24. **IF** Main Steam Line Drains 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 are opened with steam dome pressure greater than 950 psig,
THEN PERFORM the following,
OTHERWISE MARK these steps N/A:

a. Declare 2MSS*MOV111, 2MSS*MOV112, and 2MSS*MOV208 inoperable. MCA / Today

b. Enter Technical Specification 3.6.1.3, Condition B. **[C26]** MCA / Today

IV / Today
IV

F.3.25. **VERIFY** the following valves de-energized: **[C8]**

- 2MSS*SOV97A..... ☑
- 2MSS*SOV97B..... ☑
- 2MSS*SOV97C ..... ☑
- 2MSS*SOV97D ..... ☑ MCA / Today

NOTE

- 2FWS-LV10A(B) will begin to open based on system flow demand, ~ 1.75% output as read on the Master Controller output, prior to exceeding the capacity of 2FWS-LV55A(B).
- 2FWS-LV10A(B) will operate in parallel with 2FWS -LV55A(B) to control water level.

F.3.26. **VERIFY** 2FWS-LV10A(B) is operating in parallel with 2FWS -LV55A(B). MCA / Today

E. STARTUP (Continued) **Initials/Date**

E.3.27. At 2CEC-IPNL202, **VERIFY** the following:

- 2HVT-UC210A in NORMAL-AFTER-START..... ☑
- 2HVT-UC210B in NORMAL-AFTER-START..... ☑
- 2HVT-FN11 **OR** FN12 in service per N2-OP-55, E, Main Steam Tunnel Transfer Fan Startup. ..... ☑ MCA / Today

F. STARTUP (Continued) — **Initials/Date**

E.4. Turbine Startup

E.4.1 **NOTIFY** System Power Control at least one hour **PRIOR TO** 345 KV yard switch operation to allow time for traveling Operator to reach yard. MCA /Today

E.4.2. **VERIFY** Main Generator and Main Transformer startup checks have been completed per N2-OP-68, Subsections E.1.0 through E.3.4. MCA /Today

E.4.3. **VERIFY** Exciter startup checks complete per N2-OP-68A. MCA /Today

E.4.4. **DISPATCH** Operators to the Unit 2 345kV Switchyard and Scriba Switchyard to **PERFORM** N2-OP-68, Subsection E.3.5. MCA /Today

E.4.5. **VERIFY** Normal Station Service Transformer has been prepared for service per N2-OP-71A, Section F.1.0, Preparing 2STX-XNS1, Normal Station Service Transformer for Service. MCA /Today

E.4.6. **PRIOR TO** turbine roll, **VERIFY** condenser vacuum between 26.5 and 28 inches of Hg per N2-OP-9, Section H., Main Condenser Vacuum Control for Potential Rub Mitigation. MCA /Today

E.4.7. **VERIFY** Turbine shell warming has been completed per N2-OP-21, Section E.3.0, Shell Warming. MCA /Today

E.4.8. **VERIFY** steam chest warming is complete or in progress per N2-OP-21 Section E.4.0, Chest Warming. MCA /Today

E.4.9. **PREPARE** Turbine for roll by **PERFORMING** N2-OP-21, Subsections E.5.1 through E.5.3.9, Turbine Roll. MCA /Today

E.4.10. **CONTINUE** to **WITHDRAW** control rods to next target: 2½ to 3 bypass valves to support Main Generator sync. _____/_____

E. STARTUP (Continued) **Initials/Date**

E.4.11. **WHEN** 2½ to 3 bypass valves are open
OR as directed by SM,
THEN PERFORM the following: **[C24]**

a. **VERIFY** Turbine chest warming has been completed per N2-OP-21 Section E.4.0, Chest Warming. _____/_____

b. **ROLL** Main Turbine per N2-OP-21 Section E.5.0, Turbine Roll. _____/_____

c. **SYNCHRONIZE** Main Generator to grid per N2-OP-68, Section E.3.0, Preparing the Main Generator for Synchronization. _____/_____

d. **IF REQUIRED**,
THEN PERFORM N2-OP-3, Section H., Isolating 2CNM-AOV109, CONDENSATE DEMIN BYPASS LINE ISOL,
OTHERWISE MARK this step N/A. _____/_____

e. At Panel 2CEC*PNL851, **SHUT DOWN** Condenser Neck Spray by **CLOSING** 2CNM-MOV126. _____/_____

E.4.12. **TRANSFER** house load from Reserve to Normal per N2-OP-71A, 13.8KV A.C. Power Distribution, F., Transfer Station Service from Reserve to Normal. _____/_____

E.4.13. **VERIFY** positive interlock switch is made up for the following breakers:

- 2NPS-SWG001-1..... ☐
- 2NPS-SWG003-1..... ☐ _____/_____



ILT 22-2

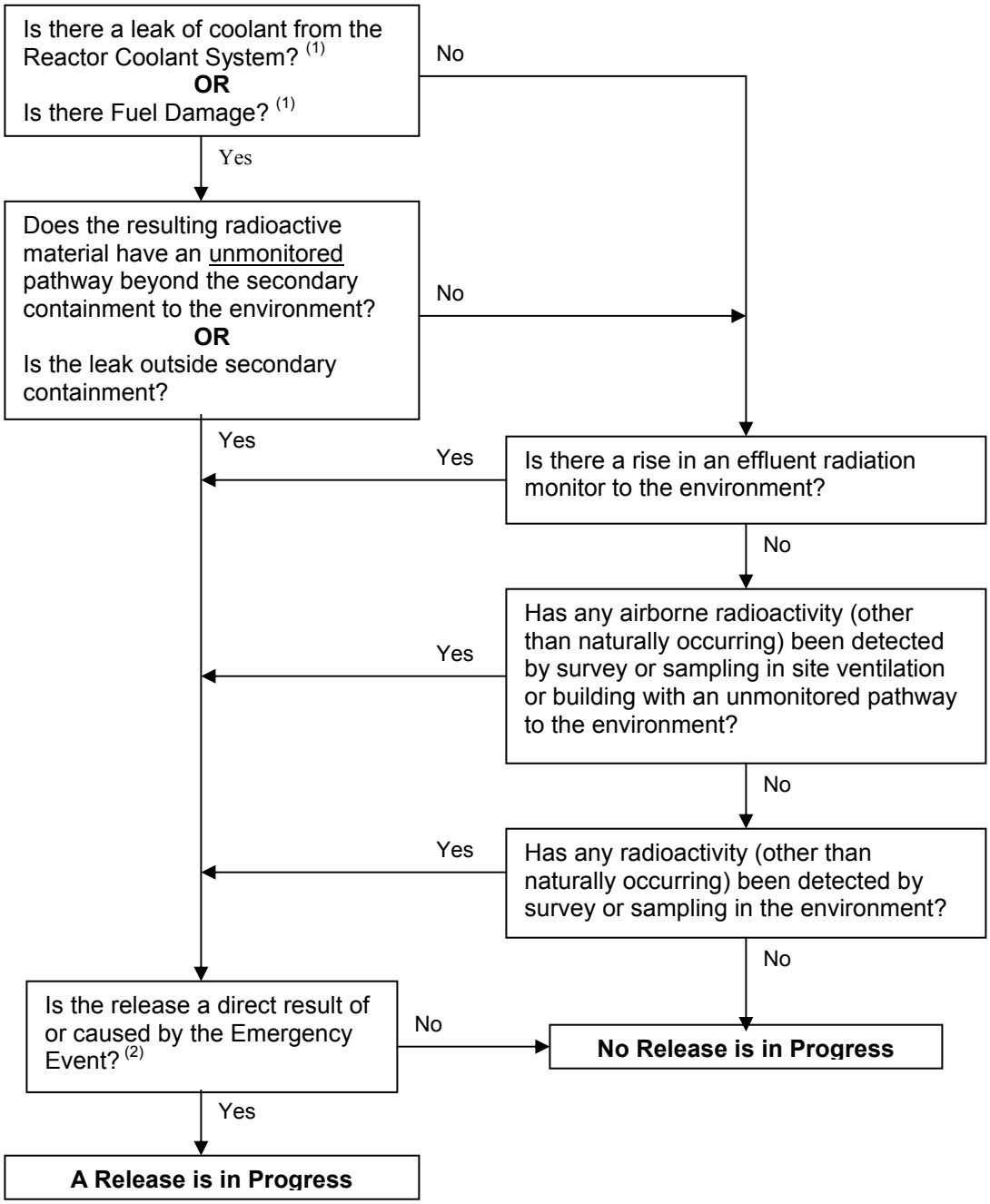
NRC Written Exam

Operator Reference Packet

BWR RELEASE IN PROGRESS DETERMINATION GUIDANCE

BWR Airborne

An **"Airborne"** release due to the event is defined as:
Any airborne radioactive release (Particulate, Iodine or Noble Gas) that is a result of, or caused by, the emergency event.

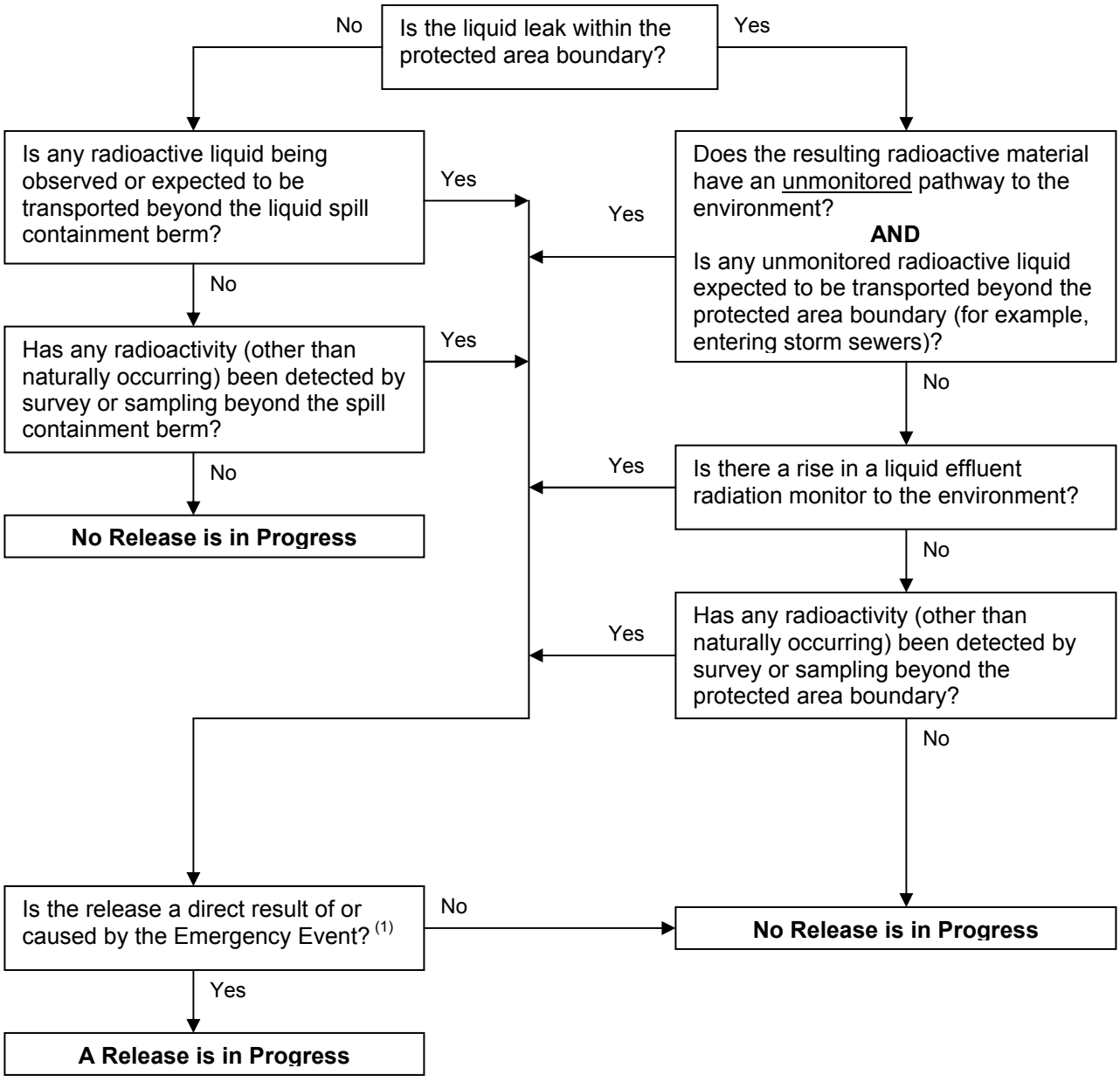


1. For release determination, a loss or leak from a barrier is NOT equivalent to the Potential Loss or Loss as determined by the EAL Fission Product Barrier matrix.
2. If a release of radioactivity is identified, then it must be a direct result of, or caused by, the classified or a concurrent emergency event and not attributed to the normal operation or shutdown of plant systems to be considered a "Release in Progress."

BWR RELEASE IN PROGRESS DETERMINATION GUIDANCE

Liquid

A **"Liquid"** release due to the event is defined as:
The movement of radioactive liquid beyond the protected area boundary or beyond a spill berm for liquid storage outside of the protected area, that is a result of, or caused by, the emergency event



1. If a release of radioactivity is identified, then it must be a direct result of, or caused by, the classified or a concurrent emergency event and not attributed to the normal operation or shutdown of plant systems to be considered a "Release in Progress."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

LCO 3.4.5 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE;

c. ≤ 25 gpm identified LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
B. Unidentified LEAKAGE not within limit. <u>OR</u> Identified LEAKAGE not within limit.	B. 1 Reduce LEAKAGE to within limits.	4 hours
C. Unidentified LEAKAGE increase not within limit.	C. 1 Reduce unidentified LEAKAGE increase to within limit. <u>OR</u> C.2 Identify source of unidentified LEAKAGE increase.	4 hours 4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify RCS unidentified and identified LEAKAGE and unidentified LEAKAGE increase are within limits.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System; and

b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

- NOTE -
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray (HPCS) System is inoperable.
- -

ACTIONS

- NOTE -
LCO 3.0.4.b is not applicable to DGs.
- -

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit. <u>AND</u>	1 hour <u>AND</u> Once per 8 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable. AND	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	A.3 Restore required offsite circuit to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program AND 24 hours from discovery of both HPCS and Low Pressure Core Spray (LPCS) Systems with no offsite power OR In accordance with the Risk Informed Completion Time Program
B. One required DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s). AND	1 hour AND Once per 8 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	B.4	Restore required DG to OPERABLE status.	72 hours from discovery of an inoperable Division 3 DG OR In accordance with the Risk Informed Completion Time Program AND 14 days OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. AND	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	C.2 Restore one required offsite circuit to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
D. One required offsite circuit inoperable. AND One required DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.8, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any division. ----------	
	D.1 Restore required offsite circuit to OPERABLE status. OR	12 hours OR In accordance with the Risk Informed Completion Time Program
	D.2 Restore required DG to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours OR 24 hours if Division 3 DG is inoperable
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	---------- NOTE ---------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ---------- Verify each required DG starts from standby conditions and achieves: a. In ≤ 10 seconds, voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG; and b. Steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program*

(continued)

* Following return to OPERABILITY of the HPCS System, the past due Surveillances will be completed by January 18, 2019.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	---------- NOTES ---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by, and immediately follow, without shutdown, a successful performance of SR 3.8.1.2. ---------- Verify each required DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3960 kW and ≤ 4400 kW for Division 1 and 2 DGs, and ≥ 2340 kW and ≤ 2600 kW for Division 3 DG.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.4	Verify each required day tank contains ≥ one hour supply of fuel oil.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.5	Check for and remove accumulated water from each required day tank.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.6	Verify each required fuel oil transfer subsystem operates to automatically transfer fuel oil from the storage tank to the day tank.	In accordance with the Surveillance Frequency Control Program*

(continued)

* Following return to OPERABILITY of the HPCS System, the past due Surveillances will be completed by January 18, 2019.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	----------NOTES---------- 1. This Surveillance shall not normally be performed in MODE 1 or 2 (not applicable to Division 3 DG). However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed within the power factor limit. However if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ----------	
	Verify each required DG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is $\leq$ 64.5 Hz for Division 1 and 2 DGs and $\leq$ 66.75 Hz for Division 3 DG.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.8 ---------- NOTES ---------- 1. This Surveillance shall not normally be performed in MODE 1 or 2 (not applicable to Division 3 DG). However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each required DG operating within the power factor limit does not trip and voltage is maintained: a. ≤ 4576 V during and following a load rejection of a load ≥ 4400 kW for Division 1 and 2 DGs; and b. ≤ 5824 V during and following a load rejection of a load ≥ 2600 kW for Division 3 DG.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.9 ---------- NOTES ---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, or 3 (not applicable to Division 3 DG). However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2 only; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 13.20 seconds, 2. energizes auto-connected shutdown loads for Division 1 and 2 DGs only, through the associated automatic load sequence time delay relays, 3. maintains steady state voltage ≥ 3950 V and ≤ 4370 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes for Division 1 and 2 DGs and supplies permanently connected shutdown loads for ≥ 5 minutes for Division 3 DG.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	---------- NOTES ---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1 or 2 (not applicable to Division 3 DG). However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each required DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start, achieves voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG; b. Achieves steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system for Divisions 1 and 2 only; and e. Emergency loads are auto-connected through the associated automatic load sequence time delay relays to the offsite power system for Divisions 1 and 2 only.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	---------- NOTE ---------- This Surveillance shall not normally be performed in MODE 1, 2, or 3 (not applicable to Division 3 DG). However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
	Verify each required DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ECCS initiation signal except: a. Engine overspeed; and b. Generator differential current.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.12 ---------- NOTES ---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2 unless the other two DGs are OPERABLE. If either of the other two DGs become inoperable, this Surveillance shall be suspended. However, this surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each required DG operating within the power factor limit operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 4620 kW and ≤ 4840 kW for Division 1 and 2 DGs, and ≥ 2730 kW and ≤ 2860 kW for Division 3 DG; and b. For the remaining hours of the test loaded ≥ 3960 kW and ≤ 4400 kW for Division 1 and 2 DGs, and ≥ 2340 kW and ≤ 2600 kW for Division 3 DG.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	---------- NOTES ---------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 3960 kW for Division 1 and 2 DGs, and ≥ 2340 kW for Division 3 DG. Momentary transients below the load limit do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. ---------- Verify each required DG starts and achieves: a. In ≤ 10 seconds, voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG; and b. Steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program*

(continued)

* Following return to OPERABILITY of the HPCS System, the past due Surveillances will be completed by January 18, 2019.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	---------- NOTE ---------- This Surveillance shall not normally be performed in MODE 1, 2, or 3 (not applicable to Division 3 DG). However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each required DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.15	---------- NOTE ---------- This Surveillance shall not normally be performed in MODE 1, 2, or 3 (not applicable to Division 3 DG). However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ECCS initiation signal overrides the test mode by: a. Returning DG to ready-to-load operation; and b. Automatically energizing the emergency load from offsite power.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	---------- NOTE ---------- This Surveillance shall not normally be performed in MODE 1, 2, or 3. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify interval between each sequenced load block, for the Division 1 and 2 DGs only, is $\geq$ 90% of the design interval for each automatic load sequence time delay relay.	In accordance with the Surveillance Frequency Control Program*

(continued)

* Following return to OPERABILITY of the HPCS System, the past due Surveillances will be completed by January 11, 2019.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.17 ---------- NOTES ---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, or 3 (not applicable to Division 3 DG). However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2 only; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. for Divisions 1 and 2, energizes auto-connected emergency loads through the associated automatic load sequence time delay relays and for Division 3, energizes auto-connected emergency loads, 3. maintains steady state voltage ≥ 3950 V and ≤ 4370 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.18	---------- NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify, when started simultaneously from standby condition, each Division 1, 2, and 3 DG achieves, in ≤ 10 seconds, voltage ≥ 3950 V for Division 1 and 2 DGs and ≥ 3820 V for Division 3 DG, and frequency ≥ 58.8 Hz for Division 1 and 2 DGs and ≥ 58.0 Hz for Division 3 DG.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 The Division 1 and Division 2 emergency uninterruptible power supply (UPS) inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One emergency UPS inverter inoperable.	A.1	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.8, "Distribution Systems – Operating" with any 120 VAC uninterruptible panel de-energized. ----------	
		Restore emergency UPS inverters to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct emergency UPS inverter voltage, frequency, and alignment to 120 VAC uninterruptible panels.	In accordance with the Surveillance Frequency Control Program

[33 FR 9704, July 4, 1968, as amended at 41 FR 18303, May 3, 1976; 45 FR 30615, May 9, 1980; 51 FR 40310, Nov. 6, 1986; 53 FR 19250, May 27, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39301, July 29, 1996; 64 FR 53614, Oct. 4, 1999; 71 FR 29246, May 22, 2006; 72 FR 49501, Aug. 28, 2007; 86 FR 43402, Aug. 9, 2021]

Effective Date Note: See 64 FR 53582, Oct. 4, 1999, for effectiveness of § 50.71(e) introductory text.

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

[Top of File]

(a) General requirements.[1] (1) Each nuclear power reactor licensee licensed under §§ 50.21(b) or 50.22 holding an operating license under this part or a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g), shall notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; [2] or

(ii) Those non-emergency events specified in paragraph (b) of this section that occurred within three years of the date of discovery.

(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 Headquarters Operations Center at the numbers specified in appendix A to part 73 of this chapter.

(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.

(4) The licensee shall activate the Emergency Response Data System (ERDS) [3] 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.

(5) When making a report under paragraph (a)(1) of this section, the licensee shall identify:

(i) The Emergency Class declared; or

(ii) Paragraph (b)(1), "One-hour reports," paragraph (b)(2), "Four-hour reports," or paragraph (b)(3), "Eight-hour reports," as the paragraph of this section requiring notification of the non-emergency event.

(b) Non-emergency events--(1) One-hour reports. If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee shall notify the NRC as soon as practical and in all cases within one hour of the occurrence of any deviation from the plant's Technical Specifications authorized pursuant to Sec. 50.54(x) of this part.

(2) Four-hour reports. If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following:

(i) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

(ii)-(iii) [Reserved]

(iv)(A) Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(B) Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(v)-(x) [Reserved]

(xi) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.

(3) Eight-hour reports. If not reported under paragraphs (a), (b)(1) or (b)(2) of this section, the licensee shall notify the NRC as soon as practical and in all cases within eight hours of the occurrence of any of the following:

(i) [Reserved]

(ii) Any event or condition that results in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or

(B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

(iii) [Reserved]

(iv)(A) Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) of this section, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(B) The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are:

(1) Reactor protection system (RPS) including: Reactor scram and reactor trip. [4]

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: High-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: High-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

(v) Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (b)(3)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (b)(3)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.

(vii)-(xi) [Reserved]

(xii) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(xiii) Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

(c) *Followup notification*. With respect to the telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, shall during the course of the event:

(1) *Immediately report* (i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.

(2) *Immediately report* (i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

1. Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular Secs. 20.1906, 20.2202, 50.36, 72.74, 72.75, and 73.1200.

2. These Emergency Classes are addressed in Appendix E of this part.

3. Requirements for ERDS are addressed in Appendix E, Section VI.

4. Actuation of the RPS when the reactor is critical is reportable under paragraph (b)(2)(iv)(B) of this section.

[48 FR 39046, Aug. 29, 1983; 48 FR 40882, Sept. 12, 1983; 55 FR 29194, July 18, 1990, as amended at 56 FR 944, Jan. 10, 1991; 56 FR 23473, May 21, 1991; 56 FR 40184, Aug. 13, 1991; 57 FR 41381, Sept. 10, 1992; 58 FR 67661, Dec. 22, 1993; 59 FR 14087, Mar. 25, 1994; 65 FR 63786, Oct. 25, 2000; 72 FR 49502, Aug. 28, 2007; 85 FR 65662, Oct. 16, 2020; 87 FR 20697, Apr. 8, 2022; 88 FR 15880, Mar. 14, 2023]

§ 50.73 Licensee event report system.

[Top of File]

(a) Reportable events. (1) The holder of an operating license under this part or a combined license under part 52 of this chapter (after the Commission has made the finding under § 52.103(g) of this chapter) for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. In the case of an invalid actuation reported under § 50.73(a)(2)(iv), other than actuation of the reactor protection system (RPS) when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Unless otherwise specified in this section, the licensee shall report an event if it occurred within 3 years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

(2) The licensee shall report:

(i)(A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any operation or condition which was prohibited by the plant's Technical Specifications except when:

(1) The Technical Specification is administrative in nature;

(2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or

(3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.

(C) Any deviation from the plant's Technical Specifications authorized pursuant to Sec. 50.54(x) of this part.

(ii) Any event or condition that resulted in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or

(B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

(iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

(iv)(A) Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, except when:

(1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or

(2) The actuation was invalid and;

(i) Occurred while the system was properly removed from service; or

(ii) Occurred after the safety function had been already completed.

(B) The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are:

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

(9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

(v) Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.

(vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(viii)(A) Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in appendix B to part 20, table 2, column 1.

(B) Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.

(ix)(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:

(1) Shut down the reactor and maintain it in a safe shutdown condition;

(2) Remove residual heat;

(3) Control the release of radioactive material; or

(4) Mitigate the consequences of an accident.

(B) Events covered in paragraph (a)(2)(ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (a)(2)(ix)(A) of this section if the event results from:

(1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or

(2) Normal and expected wear or degradation.

(x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(b) Contents. The Licensee Event Report shall contain:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2)(i) A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event.

(ii) The narrative description must include the following specific information as appropriate for the particular event:

(A) Plant operating conditions before the event.

(B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event.

(C) Dates and approximate times of occurrences.

(D) The cause of each component or system failure or personnel error, if known.

(E) The failure mode, mechanism, and effect of each failed component, if known.

(F) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER.

(1) The Energy Industry Identification System is defined in: IEEE Std 803-1983 (May 16, 1983) Recommended Practice for Unique Identification in Power Plants and Related Facilities--Principles and Definitions.

(2) IEEE Std 803-1983 has been approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51.

(3) A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ 08855-1331. IEEE Std 803-1983 is available for inspection at the NRC's Technical Library, which is located in the Two White Flint North Building, 11545 Rockville Pike, Rockville, Maryland 20852-2738; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html*.

(G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected.

(H) For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.

(I) The method of discovery of each component or system failure or procedural error.

(J) For each human performance related root cause, the licensee shall discuss the cause(s) and circumstances.

(K) Automatically and manually initiated safety system responses.

(L) The manufacturer and model number (or other identification) of each component that failed during the event.

(3) An assessment of the safety consequences and implications of the event. This assessment must include:

(i) The availability of systems or components that could have performed the same function as the components and systems that failed during the event, and

(ii) For events that occurred when the reactor was shutdown, the availability of systems or components that are needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

(4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events at the same plant that are known to the licensee.

(6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.

(c) *Supplemental information*. The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee shall submit, as specified in § 50.4, the requested information as a supplement to the initial LER.

(d) *Submission of reports*. Licensee Event Reports must be prepared on Form NRC 366 and submitted to the U.S. Nuclear Regulatory Commission, as specified in § 50.4.

(e) *Report legibility*. The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.

(f) [Reserved]

(g) *Reportable occurrences.* The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.

[48 FR 33858, July 26, 1983, as amended at 49 FR 47824, Dec. 7, 1984; 51 FR 40310, Nov. 6, 1986; 56 FR 23473, May 21, 1991; 56 FR 61352, Dec. 3, 1991; 57 FR 41381, Sept. 10, 1992; 58 FR 67661, Dec. 22, 1993; 59 FR 50689, Oct. 5, 1994; 63 FR 50480, Sept. 22, 1998; 65 FR 63787, Oct. 25, 2000; 69 FR 18803, Apr. 9, 2004; 72 FR 49502, Aug. 28, 2007]

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☑
RO Question #: 36	SRO Question #:	

Question and Comment:

ARE SHORTING LINKS INSTALLED OR ARE INITIAL CONDITIONS OF OP-101A TO BE ASSUMED?

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO	
Proctor (Signature): [signature]	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☑
RO Question #: 48	SRO Question #:	

Question and Comment:

2CES* 1PM-414 WOW NAME

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO	
Proctor (Signature):	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☑
RO Question #: 34	SRO Question #:	

Question and Comment:

CHOICE D. ARE EVALUATING RAD MONITOR RESPONSE ONLY OR WITH WHAT COULD HAVE CAUSED THE RESPONSE?

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO	
Proctor (Signature): [signature]	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:			
Applicant Level:	RO: ☐	SRO: ☑	
RO Question #: 8		SRO Question #:	

Question and Comment:

IS "LOWERING SLOWLY" CURRENT CONDITION OR DO WE EVALUATE ASSUMING CONTINUOUSLY LOWERING?

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO	
Proctor (Signature):	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☑
RO Question #: 4	SRO Question #:	

Question and Comment:

ARE WE ASSUMING A NORMAL TRANSFER OURS?

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO	
Proctor (Signature): [signature]	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☒
RO Question #: 59	SRO Question #:	

Question and Comment:

Component ID and noun name of

TSV 1D
TSV 1C
TSV 1B

BOTH GE & STONE & WEBSTER

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☑
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): GEOFFREY GIORNO

Proctor (Signature): [signature] Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☒
RO Question #: 39	SRO Question #:	

Question and Comment:

Is this question asking the impact of ADS as a DIV 1 system, or DIV 1 and DIV 2 system?

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☒
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): JUSTIN CARHART	
Proctor (Signature):	Date: 11/13/23

Question and Comment During Initial Written Exams

Applicant Name:		
Applicant Level:	RO: ☐	SRO: ☒
RO Question #: 39	SRO Question #:	

Question and Comment:

Is the question asking automatically or would we count manual action to open ADS valves via key lock

Response:	
Withheld additional guidance. Instructed applicants do the best they can, based on the information provided.	☒
Did not withhold additional guidance. Provided the following response (enter verbatim).	☐

Changes and Clarifications Made To Examination:	
Identified on white board	☐
Verbally called to applicants attention	☐
NO changes made to key	☐
Pen and ink change made to key, if required	☐

Proctor (Print): JUSTIN CARHART	
Proctor (Signature):	Date: 11/13/23

2023 Nine Mile Point Unit 2 Initial NRC Examination
Written Examination Review

In accordance with NUREG 1021, Revision 12, Section ES-4.4, an examination analysis was conducted on the 2023 Unit 2 ILT 22-2 Initial NRC Written Examination.

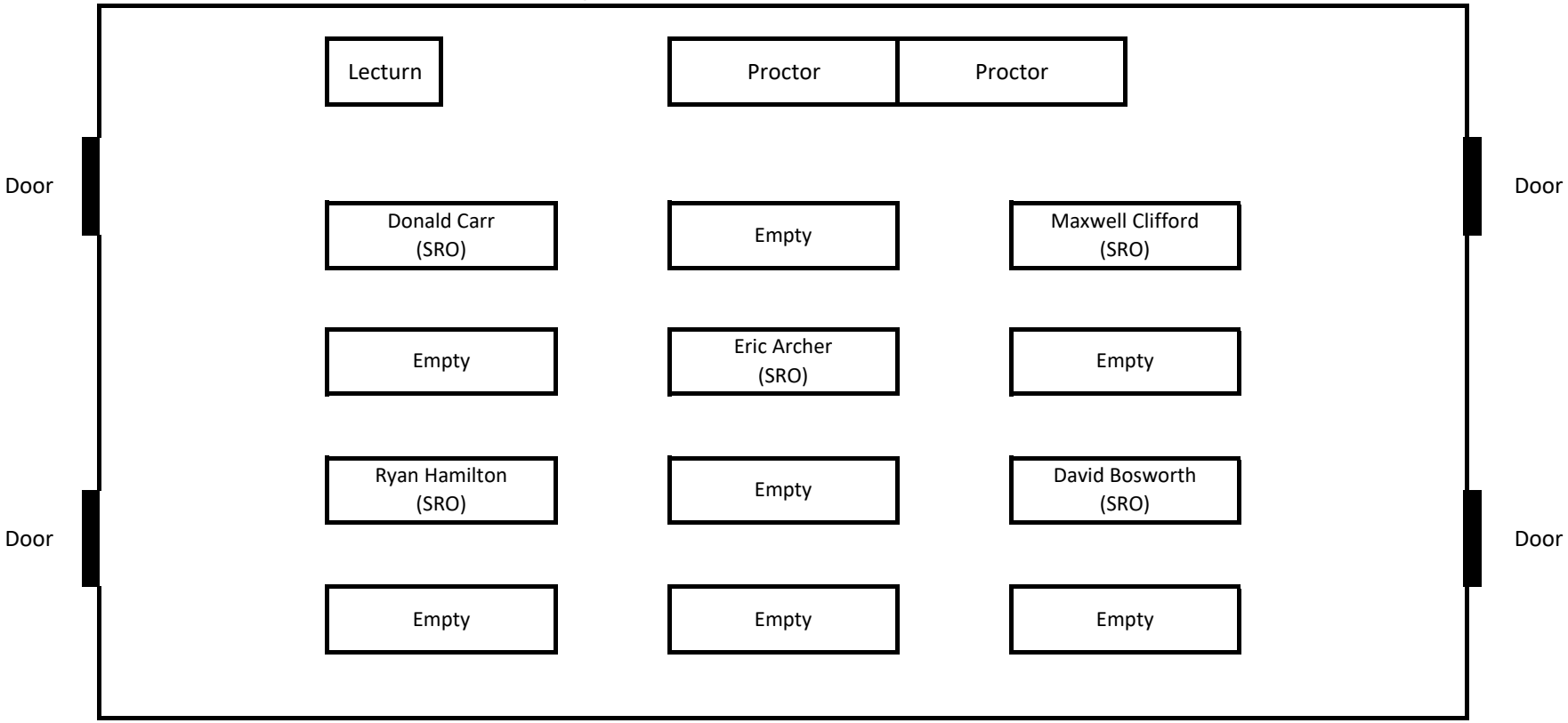
This analysis included a detailed review of each question missed by 50% or more of the applicants. No questions were identified during this analysis requiring a change in grading. The table below summarizes the findings of the high miss question review.

Unit 2 2023 NRC Exam Analysis - High Miss Questions

Question	Percent Correct	New, Modified, Bank	Higher or Lower Cognitive	System or Topic	Justification
None	N/A	N/A	N/A	N/A	N/A

Additionally, an examination review session was held to solicit applicant feedback on each question and ensure knowledge gaps were closed. This examination review session did not identify any questions requiring review for a change in grading.

NMP2 Seating Chart - November 2023 Exam NLC Classroom A1/A2



Exam Analysis

(Question Pass Rate/Exam Score)

		Score											
		(Avg)	Score #	Score %	Score #	Score %	Score #	Score %	Score #	Score %	Score #	Score %
RO		95.2%	74	98.6%	71	94.6%	73	97.3%	71	94.6%	68	90.6%
SRO		92.8%	25	100.0%	21	84.0%	25	100.0%	23	92.0%	22	88.0%
Combined		94.6%	99	99.0%	92	92.0%	98	98.0%	94	94.0%	90	90.0%
Question #	KEY	Pass Rate	RESPONSE	RESULT	RESPONSE	RESULT	RESPONSE	RESULT	RESPONSE	RESULT	RESPONSE	RESULT
RO 01	B	100%	B		B		B		B		B	
RO 02	A	100%	A		A		A		A		A	
RO 03	C	100%	C		C		C		C		C	
RO 04	C	60%	C		A		A		C		C	
RO 05	B	100%	B		B		B		B		B	
RO 06	C	100%	C		C		C		C		C	
RO 07	C	100%	C		C		C		C		C	
RO 08	B	100%	B		B		B		B		B	
RO 09	A	100%	A		A		A		A		A	
RO 10	C	100%	C		C		C		C		C	
RO 11	B	100%	B		B		B		B		B	
RO 12	C	80%	C		C		C		C		D	
RO 13	D	100%	D		D		D		D		D	
RO 14	A	100%	A		A		A		A		A	
RO 15	D	100%	D		D		D		D		D	
RO 16	B	100%	B		B		B		B		B	
RO 17	C	100%	C		C		C		C		C	
RO 18	C	60%	C		C		C		A		D	
RO 19	D	100%	D		D		D		D		D	
RO 20	B	100%	B		B		B		B		B	
RO 21	B	80%	B		B		B		B		A	
RO 22	C	100%	C		C		C		C		C	
RO 23	B	100%	B		B		B		B		B	
RO 24	C	100%	C		C		C		C		C	
RO 25	D	100%	D		D		D		D		D	
RO 26	C	100%	C		C		C		C		C	
RO 27	D	100%	D		D		D		D		D	
RO 28	C	100%	C		C		C		C		C	
RO 29	A	100%	A		A		A		A		A	
RO 30	C	100%	C		C		C		C		C	
RO 31	D	100%	D		D		D		D		D	
RO 32	D	100%	D		D		D		D		D	
RO 33	B	100%	B		B		B		B		B	
RO 34	A	100%	A		A		A		A		A	
RO 35	C	100%	C		C		C		C		C	
RO 36	B	100%	B		B		B		B		B	
RO 37	C	100%	C		C		C		C		C	
RO 38	A	100%	A		A		A		A		A	
RO 39	A	100%	A		A		A		A		A	
RO 40	B	100%	B		B		B		B		B	
RO 41	B	100%	B		B		B		B		B	
RO 42	D	80%	D		D		D		D		C	
RO 43	D	100%	D		D		D		D		D	
RO 44	C	100%	C		C		C		C		C	
RO 45	B	80%	B		A		B		B		B	
RO 46	A	100%	A		A		A		A		A	
RO 47	A	80%	A		A		A		C		A	
RO 48	A	100%	A		A		A		A		A	
RO 49	D	100%	D		D		D		D		D	
RO 50	A	100%	A		A		A		A		A	
RO 51	C	80%	C		C		D		C		C	
RO 52	D	80%	D		D		D		D		B	
RO 53	A	100%	A		A		A		A		A	
RO 54	D	100%	D		D		D		D		D	
RO 55	C	80%	C		B		C		C		C	
RO 56	D	100%	D		D		D		D		D	
RO 57	D	100%	D		D		D		D		D	
RO 58	A	100%	A		A		A		A		A	
RO 59	C	100%	C		C		C		C		C	
RO 60	B	80%	B		B		B		A		B	
RO 61	A	100%	A		A		A		A		A	
RO 62	A	100%	A		A		A		A		A	
RO 63	D	100%	D		D		D		D		D	
RO 64	D	100%	D		D		D		D		D	
RO 65	D	100%	D		D		D		D		D	
RO 66	C	60%	C		C		C		D		D	
RO 67	C	100%	C		C		C		C		C	
RO 68	B	60%	A		A		B		B		B	
RO 69	A	100%	A		A		A		A		A	
RO 70	C	80%	C		C		C		C		A	
RO 71	C	100%	C		C		C		C		C	
RO 72	C	100%	C		C		C		C		C	
RO 73	C	100%	C		C		C		C		C	
RO 74	B	100%	B		B		B		B		B	
RO 75	A	100%	A		A		A		A		A	
SRO 1 / 76	C	100%	C		C		C		C		C	
SRO 2 / 77	D	100%	D		D		D		D		D	
SRO 3 / 78	C	60%	C		B		C		C		B	
SRO 4 / 79	D	100%	D		D		D		D		D	
SRO 5 / 80	D	60%	D		B		D		B		D	
SRO 6 / 81	C	60%	C		B		C		A		C	
SRO 7 / 82	C	100%	C		C		C		C		C	
SRO 8 / 83	C	100%	C		C		C		C		C	
SRO 9 / 84	B	100%	B		B		B		B		B	
SRO 10 / 85	D	100%	D		D		D		D		D	
SRO 11 / 86	C	80%	C		C		C		C		D	
SRO 12 / 87	D	100%	D		D		D		D		D	
SRO 13 / 88	A	80%	A		A		A		A		D	
SRO 14 / 89	B	80%	B		D		B		B		B	
SRO 15 / 90	A	100%	A		A		A		A		A	
SRO 16 / 91	D	100%	D		D		D		D		D	
SRO 17 / 92	D	100%	D		D		D		D		D	
SRO 18 / 93	C	100%	C		C		C		C		C	
SRO 19 / 94	D	100%	D		D		D		D		D	
SRO 20 / 95	A	100%	A		A		A		A		A	
SRO 21 / 96	D	100%	D		D		D		D		D	
SRO 22 / 97	B	100%	B		B		B		B		B	
SRO 23 / 98	B	100%	B		B		B		B		B	
SRO 24 / 99	A	100%	A		A		A		A		A	
SRO 25 / 100	B	100%	B		B		B		B		B	