Final Submittal
(Blue Paper)

FINAL SIMULATOR SCENARIOS

BROWNS FERRY
2008-301

Browns Ferry Nuclear Plant Operations Training Group

HLT Class 0610 NRC Exam Simulator Scenarios

Facility: BFN Scenario Number: HLTS-3-1 Op-Test Number: HLT0610

Examiners:____________________ Operators: ____________________

Initial Conditions:

Unit 3 has been operating for 192 days. Unit 2 has been operating for 56 days. Unit 1 has been operating for 274 days. 3ED Diesel Generator is tagged for water jacket leakage repair. Day 2 of the LCO. Expected to be returned to service this shift. Fuel leakers on U3 are currently at RFI 60,000. Thunderstorms are passing through the region, but no watches are in effect for the immediate area. The 3C RFP was oscillating approximating 30 RPM during last shift, but is now working properly and being monitored. The 3C RFP Pump is operating in automatic in order to collect data for the next 24 hours. A trouble shooting plan is being developed.

Turnover:

Support scheduled maintenance and testing activities. Alternate Stator Cooling Water Pumps per 3-OI-35A, Sect 6.3 per scheduled OPA.

Event Number	Malfunction Number	Event Type*	Event Description
1	mrf an01b reset	N-ATC N-BOP N-SRO	The crew will alternate Stator Cooling Water Pumps using 3-OI-35A.
1	N/A	I-BOP TS-SRO	The crew will respond to a HPCI Rupture Diaphragm pressure switch PS-73-20B failure.
2	imf fw05b 100 8:00	R-ATC C-BOP R-SRO	The crew will respond to a 3B HP FW heater isolation using 3-AOI-6-1. The crew will reduce power to ~91% using a recirc flow reduction. The crew will isolate feedwater to the 3B FW heater string. The crew will further reduce power to <79% using a recirc flow reduction.
3	imf sw10a	C-BOP C-SRO TS-SRO	The crew will respond to a trip of the 3A Fuel Pool Cooling pump using 3-AOI-78-1.
4	imf fw13b	C-ATC C-BOP C-SRO	The crew will respond to a trip of the 3B Reactor Feedwater Pump (RFP) using 3-AOI-3-1 and 3-OI-3.
5	bat NRCrfpactrip	M All	The crew will respond to a total loss of feedwater and reactor scram.
6	bat HLTS04-1	M All	The crew will respond to a RCIC steam leak into secondary containment. The crew will anticipate Emergency Depressurization or perform Emergency Depressurization due to secondary containment high radiation.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: BFN Scenario Number: HLTS-3-2 Op-Test Number: HLT0610

Examiners:________________ Operators:________________

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Initial Conditions:

Unit 3 is at 79% power. 3C RHR Pump is out of service. T.S 3.5.1.A.1, 3.6.2.3, 3.6.2.4, 3.6.2.5 have been entered. Unit 3 is 6 hours into a seven day LCO. Appendix R LCO addressed and in LCO tracking. Loop II of RHR has been vented within the hour in preparation for placing Torus cooling in service. Valve 3-FCV-73-36 seal-in circuit has been disabled per step 7.6 of 3-SR-3.5.1.7

Turnover:

Continue with 3-SR-3.5.1.7 which is in progress and is complete up to Step 7.11 (HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure).
Alternate EHC Pumps per section 6.3 of 3-OI-47A. Increase reactor power to 90% using Recirc flow (3-GOI-100-12, step 5.132) at 8 Mwe per minute.

Event Number	Malfunction Number	Event Type*	Event Description
1	N/A	N-BOP	The crew will alternate EHC pumps using 3-OI-47A.
2	N/A	R-ATC R-SRO	The crew will continue with power ascension using 3-GOI-12 and 3-OI-68.
3	imf hp08	C-BOP C-SRO TS-SRO	The crew will recognize and respond to a HPCI steam line break. HPCI will fail to auto isolate and must be manually isolated. The SRO will enter and execute EOI-3.
4	imf rd01a	C-ATC	Recognize and respond to a 3A CRD pump trip using 3-AOI-85-3.
5	imf ad01g 40	C-BOP C-SRO TS-SRO	The crew will recognize and respond to a stuck open SRV using 3-AOI-1-1.
6.	bat RRPAVIB imf cr02a 75 3:00	M All	The crew will recognize and respond to a recirc pump high vibration, dual seal failure, trip, core power oscillations and scram. The crew will carry out actions using EOI-1 & 2 and 3-AOI-100-1.
7	imf th22 100 1:30	M All	The crew will recognize and respond to a MSIV Closure and LOCA using EOI-1 & 2. The crew will monitor and control primary containment until reactor water level approaches TAF. The crew will transition to EOI C-1 and perform Emergency Depressurization to enable level restoration using low pressure systems.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: BFN Scenario Number: HLTS-3-3 Op-Test Number: HLT0610

Examiners:____________________ Operators: ____________________

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Initial Conditions:

The HPCI system is tagged out for 14 hours to repair the Auxiliary Oil Pump. It is expected back in 3 hours. Flow indicator 3-78B is out of service. Instrument Mechanics are looking for a new transmitter. The Main Generator voltage regulator has been placed in Manual for PMs on the Automatic voltage regulator. The spare RBCCW pump in service to Unit 2.

Turnover:

Reduce power to 95% using recirculation flow due to low system load requirements. PMs on the voltage regulator are complete. Return the Main Generator voltage regulator to Automatic operation.

Event Number	Malfunction Number	Event Type*	Event Description
1	N/A	R-ATC R-SRO	The ATC operator will reduce reactor power to 95% using recirc flow using 3-OI-68.
1	N/A	N-BOP N-SRO	The BOP operator will return the Main Generator voltage regulator to Automatic using 3-OI-47.
2	ior zdihs7542a start	C-BOP C-SRO TS-SRO	The crew will recognize and respond to an inadvertent start of the 3D Core Spray pump. The SRO will address Tech Specs.
3	imf rd07 18-35	R-ATC C-SRO TS-SRO	The crew will recognize and respond to a control rod drifting into the core using 3-AOI-85-5. The SRO will address Tech Specs.
4	imf ed12a	C All TS-SRO	The crew will recognize and respond to a loss of 3A 480V RMOV board. The SRO will address Tech Specs.
5	bat NRC/ HLTS10-1	M All	The crew will recognize and respond to a recirc pump trip, power oscillations, scram and ATWS.
6	Timed out from batch file	M All	The crew will recognize and respond to a fuel failure during the ATWS recovery actions.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: BFN Scenario Number: HLTS-3-4 Op-Test Number: HLT0610

Examiners:____________________ Operators: ____________________

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Initial Conditions:

The unit is starting up following a refuel outage. Reactor power is at ~ 1%. "C" RFP is uncoupled for performance of turbine overspeed testing. Currently at step 5.76.8 of 3-GOI-100-1A.

Turnover:

The 3C RFP is uncoupled and the suction and discharge valves are tagged for performance of turbine overspeed. Currently at step 5.76.8 of 3-GOI-100-1A and at step 5.6.13 of 3-OI-3 for warming 3B RFP.

Event Number	Malfunction Number	Event Type*	Event Description
1	none	R-ATC N-BOP R-SRO	Crew will continue to pull rods to increase power and start warming up 2B RFP
2	imf rd14a	I-ATC I-SRO TS-SRO	Crew will respond to a RWM failure. SRO references Tech Specs.
3	imf sw02a trip 7048FTC	C-BOP C-SRO	Crew will respond to a RBCCW pump trip Crew manually closes 70-48 after fails to auto close
4	ior zdihs468a imf th23 5	C-BOP C-SRO	Crew will respond to feedwater controller malfunction which results in cold water injection
5	imf th23 5	M All	Crew responds to fuel failure after cold water injection
6	imf cu04 25 ior zdihs691 null	M All	Crew responds to a RWCU line break and scrams reactor before any area reaches max safe value.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SIMULATOR EXERCISE GUIDE

TITLE	:	SLOW LOSS OF HP FEEDWATER HEATING ON B STRING, 2A FPC PUMP TRIP, RFP TRIP, LOSS OF ALL FEEDWATER, UNISOLABLE RCIC STEAM LINE BREAK, 2 OR MORE AREA RAD LEVELS ABOVE MAX SAFE.
REVISION	:	0
DATE	:	January 2, 2008
PROGRAM	:	BFN Hot License Training

PREPARED BY:	(Operations Instructor)	1/2/08 Date
REVIEWED BY:	N/A (LOR Lead Instructor or Designee)	Date
REVIEWED BY:	E Robert Bell (Operations Training Manager or Designee)	1/5/08 Date
CONCURRED:	(Operations Superintendent or Designee) (Required for Exam Scenarios only)	Date
VALIDATION BY:	(Operations SRO) (Required for Exam Scenarios only)	1/4/08 Date
LOGGED-IN:	(Librarian)	Date
TASKS LIST UPDATED:		Date

NUCLEAR TRAINING

REVISION/USAGE LOG

REVISION NUMBER	DESCRIPTION OF CHANGES	DATE	PAGES AFFECTED	REVIEWED BY
0	Initial	01/02/2008	All	RM Spadoni

I. PROGRAM: BFN Licensed Operator Requalification Training

II. COURSE: License Requalification Training (Simulator Exercise Guide)

III. TITLE: SLOW LOSS OF HP FEEDWATER HEATING ON B STRING, FPC PUMP TRIP, RFP TRIP, LOSS OF ALL FEEDWATER, UNISOLABLE RCIC STEAM LINE BREAK, 2 OR MORE AREA RAD LEVELS ABOVE MAX SAFE.

IV. LENGHT OF LESSON: 1 ½ to 2 hours

V. Training Objectives

A. Terminal Objectives

1. Perform routine shift turnover, plant assessment and routine shift operation in accordance with BFN procedures.

2. Given uncertain or degrading conditions, the operating crew will use team skills to conduct proper diagnostics and make conservative operational decisions to remove equipment/unit from operation. (SOER 94-1)

3. Given abnormal conditions, the operating crew will place the unit in a stabilized condition per normal, annunciator, abnormal, and emergency procedures.

4. Use step text procedural compliance.

B. Enabling Objectives

1. The operating crew will recognize and respond to a high pressure heater string isolation as directed by 3-ARP-9-6A and 3-AOI-6-1A.

2. The operating crew will recognize and respond to a spurious FPC system trip and will place the 3B pump I/S in accordance with 3-ARP-94 win 1 and 3-AOI-78-1.

3. The operating crew will recognize and respond to a RFP Trip with 3-AOI-3-1.

4. The operating crew will recognize and respond to a loss of feedwater event and Rx SCRAM.

5. The operating crew will recognize and respond to unisolable RCIC steam line break, 2 or more area rad levels above max safe requiring Emergency Depressurization.

VI. References: The procedures used in the simulator are controlled copies and are used in development and performance of simulator scenarios. Scenarios are validated prior to use, and any procedure differences will be corrected using the procedure revision level present in the simulator. Any procedure differences noted during presentation will be corrected in the same manner. As such, it is expected that the references listed in this section need only contain the reference material which is not available in the simulator.

A. SOER 94-01

B. SOER 96-01

VII. Training Materials:

A. Calculator (If required)

B. Control Rod Insertion Sheet (If required)

C. Stopwatch (If required)

D. Hold Order / Caution tags (If required)

E. Annunciator window covers (If required)

F. Steam tables (If required)

VIII. Console Operator Instructions

A. Scenario File Summary

1. File: bat HLTS3-1

	MF/RF/IOR#	Description
a)	ior zlofcv712[2] on	Fails 71-2 and 71-3 open
b)	ior zlofcv713[2] on	
c)	ior zlohs712a[2] on	
d)	ior zlohs713a[2] on	
e)	ior ypovfcv712 fail_now	
f)	ior ypovfcv713 fail_now	
g)	imf rm10h (e1 :25) 30	HCU-East rad ≈ 30 mr/hr
h)	imf rm10j (e1 :25) 25	HCU-West rad ≈ 25 mr/hr
i)	Imf rm10p (e1 2:00) 50	CS/RCIC area rad 50mr/hr
j)	imf DG01D	D D/G Fails to Start
k)	imf DG02D	D D/G Trip Protective Relay Operation
l)	ior zlo0hS2110d20a[1] OFF	1816 Green Light Off
m)	mrf DG01D open	Opens logic breaker
n)	ior zdihs718a null	Fails 71-8 valve closed

2. File: bat HLTS3-1-1

	MF/RF/IOR#	Description
a)	mmf rm10p 1000 6:00	RCIC rad to max in 6 mins.
b)	mmf rm10h 1000 13:00	HCU-West to Max in 13 mins.
c)	mmf rm10j 1000 14:00	HCU- East to Max in 14 mins.
d)	imf rc09 100 7:00	RCIC steam leak
e)	imf ad01b 0	MSRV 1-19 fails closed
f)	imf ad01f 0	MSRV 1-34 fails closed
g)	Imf ad03b	MSRV 1-19 Stuck Closed
h)	Imf ad03f	MSRV 1-34 Stuck Closed

IX. Console Operator Instructions

B. Console Operators Manipulations

ELAP TIME	PFK	DESCRIPTION/ACTION
Sim. Setup	rst 28	100% power MOC
Sim. Setup	restorepref HLTS3-1	Establishes Preference Keys
Sim. Setup	setup	Verify Preference Keys
Sim. Setup	esc	Clears Popup Window
Sim. Setup	F3	trg e1 MODESW Assigns trigger
Sim. Setup	F4	bat HLTS3-1 see file summary
Sim. Setup	manual	Tag D D/G with Hold notices

ROLE PLAY: (After Stator coolant pumps alternated) As AUO, report 3-FIS-035-0065 reading 610 gpm, 3-HS-035-0040 selected for "A" Stator Coolant pump on panel 25-114. If asked, inlet pressure is 10 psig on 3-PI-35-90.

When requested to reset local Stator Coolant panel alarm then:	F5	mrf an01b reset	Allows resetting MCR alarm

ROLE PLAY: As an IM report that HPCI rupture diaphragm pressure switch PS-73-20B has failed low.

When directed from the Floor then:	F6	imf fw05b 100 8:00	'B' HP heater string isolation

ROLE PLAY: If sent to investigate which valve is open, wait 2 minutes and report 3-LCV-22B light is out(B2 high level dump)

ROLE PLAY: At ≈ 79% power, as the Reactor Engineer, recommend inserting the first group of Emergency Insert Control Rods.

If asked to reset local Cond Demin alarm	F7	mrf an01d reset	allows reset of control room alarm
After conditions stabilized or as directed by Floor Instr.	F8	imf sw10a	Trips 3A FPC pump

ROLE PLAY: (If asked) As AUO, report 3-78-506, 511, & crosstie 507 are open & 3-78-510 (B hx outlet) is closed

ROLE PLAY: (If asked) As RW UO, 3-FRC-78-24 is in manual & set to 0%

If asked to throttle 3-FCV-78-66	F9	lor zlohs7866a[2] on
If asked to close 3-FCV-78-66	F10	dor zlohs7866a[2]

ROLE PLAY: (If asked) As Rx Bldg AUO, report 3B pump discharge pressure is 140 psig (PI-78-16 on 9-25-16)

ROLE PLAY: If sent to inspect breaker on 3A FPC pump, report bkr was found tripped and will not test

↓↓↓MORE FOLLOWS↓↓↓

IX. Console Operator Instructions

G. Console Operators Manipulations (continued)

ELAP. TIME	PFK	DESCRIPTION/ACTION	
After FPC restored and as directed by Floor Instr.	F11	Imf fw13b	Trips 3B RFP on thrust brng wear

ROLE PLAY: If sent to check 3B RFP report that there is no apparent cause but you will continue to check

When directed by Lead Examiner	F12	bat rfpactrip	trip a&c RFP's
If doesn't start on low level	<shift>F1	imf rc02	Start of RCIC
After HPCI is in manual control and injecting up to -50" or directed by Lead Examiner then:	<shift>F2	imf hp07	HPCI 120V failure
After 10 minutes of RCIC operations or directed by Lead Examiner then:	<shift>F4	bat HLTS3-1-1	Max. Rad (2 areas in 13 mins.)

ROLE PLAY: If directed to close RCIC valves 71-2 & 3 locally, respond that you are waiting on RadCon to enter the Reactor building.

If decided to attempt to close valves locally:	mrf rc05k emer mrf rc05s emer	71-2 to emerg 71-3 to emerg
To return transfer switch to normal	mrf rc05k norm mrf rc05s norm	71-2 to norm 71-3 to norm

ROLE PLAY: Outside US reports that it appears to be a generic problem with RFP control oil system.

After RCIC is injecting and recovering level then:	<shift>F3	dmf fw13b	Allows Crew to inject with "B" RFP

ROLE PLAY: Call the MCR and report that "B" RFP is repaired and ready for use

Terminates the scenario when the following conditions are satisfied or upon request of the floor instructor:

1. All rods fully inserted
2. Reactor Water level normal
3. Emergency Depressurization

X. Scenario Summary

With the unit operating at 100%, the operating crew will experience a slow loss of FW HTR level control on the B high pressure heater string. Once the heater is isolated and power reduced, a trip of 3A FPC pump will require the operator to start 3B FPC pump per 3-OI-78. When plant conditions are stable the 3B Reactor Feedwater Pump will trip, the crew will respond per 3-AOI-1-3. After conditions stabilize, The crew will experience a loss of the remaining RFPs which will cause the crew to scram and utilize RCIC for level control. When RCIC is initiated it develops a steam leak which cannot be isolated forcing the crew to emergency depressurize based on 2 Area Rad Monitors above maximum safe. If HPCI is used for water level control the crew will experience a problem with the flow controller to respond in automatic.

X. Information to Floor Instructors:

A. Ensure recorders are inking and recording and ICS is active and updating.

B. Assign Crew Positions based on the required rotation.

1. SRO: Unit Supervisor
2. ATC: Board Unit Operator
3. BOP: Desk Unit Operator

C. Terminate the scenario when the following conditions are satisfied or at the direction of the Lead Examiner:

1. All rods fully inserted
2. Reactor Water level normal
3. Emergency Depressurize on 2 RADS above max safe

XI. Simulator Event Guide

Event 1: NOR. OPS. & HPCI PRESSURE SWITCH FAILURE

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC/BOP	Alternate Stator Cooling Pumps IAW OI-35A, sect. 6.3. -start standby pumps -stops running pump -coordinates local verification of system flow and pressure -coordinates local positioning of selector switch	
	Responds to Report by IMs of HPCI rupture diaphragm pressure switch failure (3-PS-73-20B), by relaying information to SRO.	______
SRO	Consults Tech Spec 3.3.6.1 determines only three pressure switches required.	______

XI. Simulator Event Guide

Event 2: Slow Loss of HP Feedwater Heating on B string

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
Crew	Announces "BYPASS VALVE TO CONDENSER NOT CLOSED"	
ATC/BOP	Dispatch AUO to JB 32-42 to determine which bypass valve is open per ARP	
	Selects ICS screen FWHL	
	Announces "HEATER B2 LEVEL HIGH"	
	Dispatches personnel to Heater Level Controls	
	Verifies 3-FCV-6-95 open	
	Checks B2 heater shell pressure, drain flow	
	Announces B1 and B2 HP htr. Extraction isolation	
SRO	Enters 3-AOI-6-1: • Contacts Reactor Engineer	
ATC/BOP	Reduces power to 91% rated with recirc flow (if above)	
	Verifies 3B1 & 3B2 extraction valves closed	
	Verifies 3B1 & 3B2 MS Dr. Pump suction valves closed	
	Identifies heater level still rising	
SRO	Directs isolating FW to B HP heater string	
	Directs power reduction to < 79% power (Mid-power runback)	
	Enters 3-GOI-100-12, Power Maneuvering	
	Notifies Rx Eng. of Feedwater Heater isolation and power reduction	

XI. Simulator Event Guide

Event 2: Slow Loss of HP Feedwater Heating on B string (Continued)

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC/BOP	Isolates FW to B HP heater string by closing 3-FCV-3-31 and 76	______
	Reduces Power to < 79% with Recirc. Flow	______
	Monitors MT thrust bearing temps. (3-AOI-6-1A)	______
	Closes 3-FCV-6-95	______
SRO	Notifies ODS of reason for power reduction	______
ATC/BOP	Notifies Chemistry & RACON	______
Crew	Recognizes HTR level lowers as a result of isolating the Condensate side of 3B HP HTR string (i.e. tube leak)	______

XI. Simulator Event Guide

Event 3: Trip of 3A FPC pump

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
Crew	Recognizes 3A FPC pump trip, responds per the ARP	__________
ATC/BOP	Performs the following:	
	Responds to alarm FPC system abnormal 3-ARP-9-4C win 1	__________
	Enters 3-AOI-78-1 for start of a FPC pump	__________
	Coordinates with Rx Bldg AUO and Radwaste UO to start 3B FPC pump	__________
	Starts 3B FPC pump	__________
	Verifies discharge pressure >120 psig with AUO	__________
	Directs RW UO and Rx Bldg AUO place demin in service	__________
SRO/BOP	Dispatch AUO/EMs to check breaker for 3A FPC pump	__________
SRO	-Directs restoration of system after cause is determined	__________
SRO	Evaluate Tech. Spec. (TRM 3.9.2/3.9.3)	__________

XI. Simulator Event Guide

Event 4: 3B RFPT Trip

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC/BOP	Announces "RFPT B Abnormal" alarm and trip of RFPT 'B'.	
	Refers to ARP, 3-AOI-3-1 and 3-OI-3 and take required action	
SRO	Dispatches AUO to RFP to determine cause of trip	
ATC/BOP	Verifies that unit stable	
	Verifies Rx Thermal limits	
SRO	Contacts maintenance to check reason for RFPT trip	

NOTE: LEAD EXAMINER notify Console Instructor when ready to trip the next RFP (i.e. next event)

XI. Simulator Event Guide

Event 5: 3A and 3C RFP Trip

POSITION	EXPECTED ACTION	SAT/UNSAT/NOTES
ATC/BOP	Recognizes 3A RFP trip and need for reactor scram	
SRO	Directs Reactor scram	
ATC	Manually scrams the reactor -mode switch in S/D -checks power lowering -reports all rods in -recognizes trip of 3C RFP and informs SRO all RFP's are tripped	
SRO	Enters 3-EOI-1 on low reactor water level Directs level be controlled by: -RCIC -CRD -HPCI -Enter AOI-100-1	
ATC/BOP	Utilizes RCIC for reactor water level control	
Crew	Recognizes radiation alarms associated with RCIC operation	
ATC/BOP	Evacuates Reactor. Bldg.	
SRO	Enters EOI-3	

XI. Simulator Event Guide

Event 6: RCIC STEAM LEAK

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC/BOP	If HPCI is used, recognizes auto control failure and places HPCI controller in manual	
ATC/BOP	Places 3B RFP I/S after notified able to reset	
Crew	Monitors area radiation levels	
ATC/BOP	Recognizes and reports area radiation alarm for RCIC room	
	Recognizes and reports high area temperature for RCIC room	
	Recognizes RCIC failure to isolate and attempts to manually isolate it	
SRO	Directs RCIC be isolated locally	
	Determines has two area radiation levels above max safe IAW EOI-3 and directs emergency depressurization by opening 6 ADS valves (C2)	
ATC/BOP	Opens 6 ADS valves and recognizes 2 valves failed to open and opens 2 additional valves	
	Verifies RFP discharge valves closed	

XI. Simulator Event Guide

Event 6: RCIC STEAM LEAK

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
SRO	Directs RPV level be maintained between +2" and +51" with one or more of the following: (After emergency depress.) -LPCI -Core Spray -Condensate	
ATC/BOP	After emergency depressurization maintains RPV water level TAF and restores level +2" to +51" with one or more of the following: -LPCI -Core Spray -Condensate	
SRO	After EOI-2 entered on high SP water level or temperature directs the following: - H_2O_2 analyzers placed in service	
ATC/BOP	Places H_2O_2 analyzers in service	
SRO	Directs all available Suppression Pool cooling be placed into service due to Suppression Pool water temperature	
ATC/BOP	Places all available Suppression Pool cooling into service	

XII. Crew Critical Tasks (If an evaluated scenario)

	Task	SAT/UNSAT
1.	Maintains reactor water level above TAF	______________
2.	Anticipates Emergency depressurize and rapidly depressurizes using BPV's to main condenser and/or Emergency depressurize based on 2 areas radiation above maximum safe with a primary system discharging to secondary containment (within 5 minutes)	______________

XIII. Scenario Verification Data

EVENT	TASK#	K/A	RO	SRO	CONTROL MANIPULATION
1. Loss of HP Feedwater Heating		295014	3.7	3.9	B17
	U-068-NO-10				
	U-006-AB-01				
	S-006-AB-01				
	T-000-AD-17				
2. 3A FPC pump trip					
3. RFPs Trip		295001A2.01	3.7	3.7	B3
	U-003-AB-01				
	S-003-AB-01				
	U-003-NO-08	295001A4.02	3.9	3.7	
		295009G12	3.8	4.4	
	T-000-AD-17				
4. HPCI Pressure Switch Failure	U-073-AL-19	206000A2.09	3.5	3.7	B5
	S-000-AD-27	2.1.12	2.9	4.0	
5. RCIC Leak/MSL Leak	U-000-EM-10	295033	3.6	3.9	A7,B14,A12,
	U-000-EM-11				B15,I4,I20
	S-000-EM-10	295032	3.5	3.6	
	S-000-EM-12				
	U-000-EM-01		3.8	4.4	
	U-000-EM-02				
	U-000-EM-03		3.6	4.2	
	S-000-EM-01		3.5	4.1	
	S-000-EM-02		3.9	4.5	
	S-000-EM-03		3.9	4.5	
	U-000-EM-14	2.4.38	2.2	4.0	
	S-000-EM-15	295026	3.6	3.8	
	S-000-EM-24				
	T-000-AD-04				
	T-000-EM-09				
	T-000-EM-11				
	T-000-EM-16				

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER HLTS 3-1

5 Total Malfunctions Inserted; List: (4-8)
1) High Pressure Heater Isolation
2) 3A FPC pump trip
3) RFPT trip
4) RCIC steam leak,
5) RCIC failure to isolate (auto or manual),

2 Malfunctions That Occur After EOI Entry; List: (1-4)
1) RCIC steam leak
2) RCIC isolation failure (auto or manual)

3 Abnormal Events; List (1-3)
1) HP Heater Isol. (ARPs)
2) 3A FPC pump trip. (AOI & ARP)
3) RFPTs trip (ARP, AOI)

1 Major Transients; List: (1-2)
1) RCIC Line Break

3 EOIs used; List: (1-3)
1) EOI-1
2) EOI-2
3) EOI-3

1 EOI Contingencies Used; List: (0-3)
1) C2

90 Run Time (minutes)

29 EOI Run Time (minutes); 30 % of Scenario EOI Run Time

2 Crew Critical Tasks

yes Technical Specifications Exercised (yes/no)

XIV. SHIFT TURNOVER INFORMATION

Equipment out of service/LCOs: Unit 3 has been operating for 193 days, Unit 2 has been operating for 56 days. Unit 1 has been operating for 290 days.

3ED Diesel Generator tagged for water jacket leakage repair Day 2 of LCO, will be returned to service this shift.

Operation/Maintenance for the Shift: Support scheduled maintenance and testing activities

Alternate Stator Cooling Water Pumps per 3-OI-35A, Sect 6.3 per scheduled OPA.

Unusual Conditions/Problem Areas: Fuel leakers on U3 are currently @ RFI 60,000.

Storms passing through the region, No Watches in effect fo r the immediate area.

3C RFW Pump was oscillating approximating 30 RPM during last shift, but currently working properly and being monitored. Pump is operating in automatic to collect data for next 24 hours. Trouble shooting plan being developed.

SIMULATOR EVALUATION GUIDE

TITLE	:	HPCI STEAMLINE BREAK, SRV FAILURE, RECIRC PUMP TRIP, DRYWELL LEAK , EMERGENCY DEPRESSURIZATION ON LEVEL (C1)
REVISION	:	0
DATE	:	January 2, 2008
PROGRAM	:	BFN Operator Training - Hot License

PREPARED BY:	(Operations Instructor)	1/2/08 Date
REVIEWED BY:	N/A (LOR Lead Instructor or Designee)	Date
REVIEWED BY:	(Operations Training Manager or Designee)	1/5/08 Date
CONCURRED:	(Operations Superintendent or Designee) (Required for Exam Scenarios only)	Date
VALIDATION BY:	(Operations SRO) (Required for Exam Scenarios only)	1/4/08 Date
LOGGED-IN:	(Librarian)	Date
TASKS LIST UPDATED:		Date

NUCLEAR TRAINING

REVISION/USAGE LOG

REVISION NUMBER	DESCRIPTION OF REVISION	DATE	PAGES AFFECTED	REVIEWED BY
0	INITIAL	4/6/07	All	

I. Program: BFN Operator Training

II. Course: Hot License Training

III. Title: HPCI STEAMLINE BREAK, SRV FAILURE, RECIRC PUMP TRIP, DRYWELL LEAK , EMERGENCY DEPRESSURIZATION ON LEVEL (C1)

IV. Length of Scenario: ≈1 to 1 ½ hours

V. Examination Objectives:

A. Terminal Objective

1. Perform routine shift turnover, plant assessment and routine shift operation in accordance with BFN procedures.
2. Given abnormal conditions, the operating crew will place the unit in a stabilized condition per normal, abnormal, annunciator and emergency procedures.

B. Enabling Objectives:

1. The operating crew will alternate EHC pumps.
2. The operating crew will continue power ascension from ≈ 79% power.
3. The operating crew will experience a HPCI steam line break during performance of 3-SR-3.5.1.7 , HPCI Flow Rate, with a failure of HPCI to auto isolate.
4. The operating crew will recognize and respond to a safety-relief valve failed open.
5. The operating crew will recognize and respond to a high vibration and trip of 3A Recirc pump.
6. The operating crew will recognize and respond to reactor power oscillations by scramming the reactor.
7. The operating crew will recognize and respond to a high drywell pressure condition.
8. The operating crew will Emergency De-pressurize when in C1 before reactor water level reaches -190”.

VI. References: The procedures used in the simulator are controlled copies and are used in development and performance of simulator scenarios. Scenarios are validated prior to use, and any procedure differences will be corrected using the procedure revision level present in the simulator. Any procedure differences noted during presentation will be corrected in the same manner. As such, it is expected that the references listed in this section need only contain the reference material which is not available in the simulator.

VII. Training Materials:

A. Calculator

B. Control Rod Insertion Sheet

C. Stopwatch

D. Hold Order/Caution tags

E. Annunciator window covers

F. Steam tables

VIII. Console Operators Instructions

A. Scenario File Summary

1. File: bat HLTS3-2

MF/RF/1OR#	Description
a) trg e1 MODESW	Sets trigger
b) trg e2 adssrv1-22	Sets trigger
c) ior zlohs7416a[1] off	Tag Out 3C RHR
d) imf rh01c	
e) ior zdihs7416a null	
f) mrf hw01 fast	Advances all charts
g) imf th33b (e1 0) 1 2:00	B MSL break in DW
h) imf th21 (e1 5:00) 1 10:00	Recirc. line break
i) imf rd01a (e1 10:00)	3A CRDP trip
j) imf rd01b	3B CRDP trip
k) imf hp09	Failure of HPCI to auto isolate
l) ior zdihs718a close	Fails RCIC
m) ior ypovfcv718 fail_power	Keeps the 8 valve closed
n) imf rp11 (e1 1:00)	MSIV logic fuse failure
o) ior zdihs261a null	Prevents Fire pump A from starting
p) ior zdihs262a null	B
q) ior zdihs263a null	C

2. File: bat torhrc

MF/RF/1OR#	Description
1) ior zlohs7416a[1] off	RHR C Tagout
2) imf rh01c	
3) ior zdihs7416a null	

3. File: bat RRPAVIB

MF/RF/1OR#	Description
1) imf th12a	Inserts Vibration Alarm
2) imf th10a (none 1:)	Fails Recirc Pump A Inboard Seal
3) imf th11a (none 2:)	Fails Recirc Pump A Outboard Seal)
4) ior zdihs681 open	Prevents Recirc Pump A Suction Valve Closure

B. Console Operators Manipulations

ELAP. TIME	PFK#	DESCRIPTION/ACTION
Simulator setup	rst 28	≈ 78 %Power MOC, **use mid-power runback push button**
Simulator setup	restorepref HLTS3-2	Establishes Function Keys
Simulator setup	setup	Verify Function Keys
Simulator setup	esc	Clears Function Key Popup
Simulator setup	F3	bat HLTS3-2 See Scenario File Summary
Simulator setup	manual	Place suppression pool cooling in service (Loop II)
Simulator setup	manual	Place HO tags on '3C' RHR pump
Simulator setup	manual	Place TESTING/MAINT frames on Panel 9-3F, Windows 5, 11, 26 for HPCI 3-SR-3.5.1.7 complete up to step 7.11

ROLE PLAY: If asked, state that the anti-rotation collar markings are aligned.

When HPCI is at rated pressure and flow	F4	imf hp08	Steam leak into HPCI room

ROLE PLAY: AUO at HPCI quad. Reports a large steam leak on HPCI and present location is elev. 565 Rx.Bldg.

When requested, wait 2 min. then:	F5	imf rd01a	trips 2A CRDP
When directed by Lead Instructor	F6	imf ad01g 40	Fails SRV-1-4 open
When RO cycles SRV then:	F7	dmf ad01g	SRV-1-4 closes
When directed by Lead Instructor	F8	bat RRPAVIB	Recirc Pump A high vibration, seal failure, suction valve fails to close and power oscillations.

When dispatched to check 2A Recirc Vibration, wait 2 minutes and report back swinging 10 to 14 mils

When 'A' Recirc trips	F9	dmf th12a	Deletes vibration high alarm
4 min. after 2A recirc. pump trip then:	F10 and F11	imf cr02a 75 3:00 imf th22 (none 1:30) 100	Core power oscillations Bottom head leak
When requested, wait 3 minutes	F12	bat app16fg	Defeats RHR injection valve timers

Terminate the scenario when the following conditions are satisfied are at the direction of the Lead Examiner.

1. RPV water level +2" to +51"
2. Drywell sprayed
3. Emergency Depressurizarion completed

IX. Scenario Summary

Given Unit 2 at 79% power, the crew will alternate EHC pumps and resume power ascension to 100%. As 3-SR-3.5.1.7, HPCI Flow Rate, is continued the crew will experience a ruptured HPCI steam line with a failure of HPCI to automatically isolate. Manual HPCI isolation will be possible. As power ascension is continued, an SRV fails open but can be closed as steps of 3-AOI-1-1 are performed. The crew experiences high vibration with a subsequent trip and seal leakage on the 3A Recirc Pump resulting in high drywell pressure. When the diesel generators automatically start the 3ED diesel generator fails to auto start but can be manually started. Finally, the crew will Emergency Depressurizes before reactor water level reaches -190".

X

Information to Floor Instructors:

A. Ensure recorders are inking and recording and ICS is active and updating.

B. Assign Crew Positions based on the required rotation.

1. SRO: Unit Supervisor
2. ATC: Board Unit Operator
3. BOP: Desk Unit Operator

C. Conduct a shift turnover with the Unit Supervisor.

D. Direct the shift crew to review the control board and take note of present conditions, alarms, etc.

E. Terminate the scenario when the following conditions are satisfied are at the request of the floor/lead instructor/evaluator.

1. RPV water level +2" to +51"
2. Emergency Depressurizarion completed

XI. Simulator Event Guide

Event 1: Alternate EHC Pumps

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
ATC/BOP	Receive crew briefing and walk boards down	______
SRO	Directs BOP to alternate EHC pumps	______
BOP	Alternates EHC Pumps in accordance with 3-OI-47A	______
	• Starts 3B EHC Pump	______
	• Verifies EHC header pressure 1550 to 1650 psig	______
	• Verifies 3B EHC motor amps <140	______
	• Stops 3A EHC Pump	______

XI. Simulator Event Guide (Continued)

Event 2: Power Ascension continued

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
SRO	Directs power ascension per 3-GOI-100-12 and 3-OI-68	______________
ATC	Raises reactor power at 8 Mwe/minute in accordance with 3-GOI-100-12 and 3-OI-68	______________
BOP	Performs as peer checker for recirc flow changes	______________

XI. Simulator Event Guide (Continued)

Event 3: HPCI Steam Line Break

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
SRO	Directs BOP to continue with 3-SR-3.5.1.7 at step 7.11	
BOP	Makes plant announcement HPCI is to be started	
	Responds to Reactor Bldg Hi Rad alarm per the ARP	
SRO	Enters EOI-3 on High Rad. / High Temp.	
BOP	Determines HPCI area source of hi rad	
	Responds to HPCI Leak Detection Temp Hi alarm per the ARP	
	Recognizes HPCI not isolated when isolation lights are illuminated	
SRO	Directs HPCI manually isolated	
BOP	Manually isolates HPCI steam supply	
	Evacuates HPCI area	
SRO	Receives EOI-3 entry on flood level in HPCI room	
BOP	Notifies Rad Con and Fire Protection	
	Monitors for lowering temperature and radiation levels in HPCI area	
SRO	Directs entry into 3-AOI-64-2B	
	Directs FCV-1-55 and FCV-1-56 Open	
BOP	Opens FCV-1-55 and FCV-1-56 Open	
SRO	Sends personnel to investigate	
	Determines unit in 72 hour LCO (TS 3.5.1.D - HPCI and C RHR Inop)	
	Tech. Specs. 3.6.1.3, on FCV 73-2 or 73-3 when tagged (1 hour)	

XI. Simulator Event Guide (Continued)

Event 4: SRV-1-22 Fails Open

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
CREW	Recognizes SRV open • Main Steam Relief Valve Open alarm • lowering generator output	______
SRO	Directs response per AOI-1-1	______
BOP	Determines SRV-1-22 from acoustic monitor	______
BOP	Places SRV-1-22 control switch from close to open to close several times	______
BOP	Cycles relief valve and reports SRV closed	______
SRO	Evaluates Tech Spec operability of ADS valve. Determines valve operable, but requests Eng. evaluation (Functional evaluation)	______

XI. Simulator Event Guide (Continued)

Event 5: Recirc Vibration, Seal Leakage, Power Oscillations and Scram

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC	Announces Recirc "3A" high vibration alarm	
	Consults ARP for Panel 9-4	
	Directs AUO to Local Panel to check vibration	
	Monitors Recirc Pump Temperatures	
SRO	Contacts Reactor Engineer	
	Directs BUO to reduce speed of 3A RRP to reduce vibration	
ATC	Reduces 3A RRP speed with peer check to clear vibration alarm	
	Announces Recirc A Seal Leakage Alarm	
	Identifies Seal Failure via Instrumentation	
	Recognizes lowering pressure on Recirc Pump A #1 seal	
SRO	Directs crew to watch for signs of increased leakage	
ATC	Acknowledges Recirc Pump A seal leakoff high alarm; informs SRO; consults ARP	
	Recognizes lowering pressure on Recirc Pump A outboard seal; informs SRO	
	Monitors drywell parameters; notes pressure and temperature increasing; informs SRO	
SRO	When vibration report received or dual seal failure is reported, directs 'A' Recirc Pump tripped	
ATC	Trips Recirc A and closes the discharge valve	
SRO	Directs actions per 3-AOI-68-1	

XI. Simulator Event Guide (Continued)

Event 5: Recirc Vibration, Seal Leakage, Power Oscillations and Scram (Continued)

POSITION	EXPECTED ACTIONS	SAT/UNSAT/NOTES
ATC	Directs AUO to Recirc MG Set to monitor oil temp.	______
SRO	Directs 'A' Recirc Isolated	______
ATC	Notes that Recirc Pump A suction isolation valve will not close; informs SRO	______
	Directs AUO to close Recirc Pump suction valve locally at Board.	______
	Checks Power to flow map to verify in region 1	______
	Checks APRMs and LPRMs for indication of power oscillations	______
	Informs SRO of Power Oscillations	______
SRO	Directs inserting emergency shove sheet control rods	______
BOP	Keeps SRO informed as drywell pressure approaches 2.45 psig	______
SRO	Directs venting per 3-OI-64-1	______
BOP	Vents per 3-OI-64-1	______
	Directs Logs person to monitor release rates	______
SRO	Directs manual reactor scram prior to reaching 2.45psig DW pressure	______
ATC	Scrams the reactor	______
SRO	Directs 3-AOI-100-1	______
ATC	Carry out actions of 3-AOI-100-1	______
SRO	Enters EOI- 1 & 2 at 2.45 psig drywell pressure	______
SRO	Directs venting per Appendix 12	______

XI. Simulator Event Guide (Continued)

EVENT 6: MSIV CLOSURE/LOCA

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
SRO	- RPV pressure controlled 800 to 1000 psig with one or more of the following: - MSRV's (App 11A) - RCIC (App 11B) - RPV level be maintained between +2" to +51" with one or more of the following: -RCIC -CRD	
BOP	Controls pressure 800 to 1000 psig with one or more of the following: - MSRV's (App 11A) - RCIC (App 11B) Recognizes MSIV closures and reports to SRO.	
SRO	Directs determining the cause of the isolation Directs App 8G, App 12, and H_2O_2 Analyzers in service	
BOP	Performs App 8G, App 12, and Places H_2O_2 Analyzers in service	

EVENT 6: MSIV CLOSURE/LOCA (continued)

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
ATC/BOP	Attempts to maintain RPV water level +2" to +51" with one or more of the following:	
	-RCIC (App 5C) -CRD (App 5B) 3-BYV-85-551 -SLC (App7B)	
SRO	Directs SP cooling be placed in service	
BOP	Places SP cooling in service	
SRO	Directs App 8G be performed	
BOP	Performs App 8G	
	Monitors containment parameters	
SRO	Enters EOI-2 on DW pressure and re-enters EOI-1 and directs the following:	
	- Verify all available DW coolers in service - Venting per App 12 - H_2O_2 analyzers placed in service	
SRO	Directs cooldown	
ATC/BOP	Verify all available DW coolers in service	
ATC/BOP	Commences a cooldown as directed	
SRO	Determines cannot maintain SC pressure less than 12 psig and directs SC sprayed	
BOP	Sprays suppression chamber per App 17C	

EVENT 6: MSIV CLOSURE/LOCA (continued)

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
SRO	When SC pressure exceeds 12 psig or if SRO determines cannot maintain DW temp. <280 then directs the following:	
	- Ensures Recirc. pumps shutdown	
	- DW blowers secured	
	- DW sprayed per App 17B	
ATC	Trips Recirc. pumps	
	Secures DW blowers	
	Requests 16F & 16G be performed	
	Sprays the DW using RHR	
SRO	Directs DW sprays/SC sprays be stopped when that area reaches 0 psig	
BOP	Stops DW/SC sprays when that area reaches 0 psig	
SRO	Directs CRD inject per App 5B	
ATC	Performs App 5B Reports 3B CRDP tripped	
	Monitors containment parameters	
SRO	Monitors RPV water level, determines level is lowering. Re-enters EOI-1 at +2" RPV level - Directs performance of App 5B (CRD) - Directs performance of App 7B (SLC)	
Crew	Monitors Drywell / PSC / and RPV water level	
SRO	Enters C1 at ≈ -100" to - 122"	
	Directs ADS inhibited	
ATC	Closes RFP discharge valves	
	Reports 3A CRDP tripped	

EVENT 6: MSIV CLOSURE/LOCA (continued)

POSITION	EXPECTED ACTION(S)	SAT/UNSAT/NOTES
BOP	Inhibits ADS	______
SRO	After entering C1 align all available injection systems for injection. -Containment sprays terminated When water level reaches TAF (-162") and before -190 directs the following: Enters C2 - Six ADS valves opened - RPV level returned +2" to +51"	______ ______ ______ ______
BOP	When directed by US terminates Containment Sprays and lines up RHR for LPCI	______
BOP	Opens and verifies open 6 ADS valves	______
ATC/BOP	Restores RPV water level +2" to +51" using: -RHR -Core Spray -Condensate	______ ______ ______
SRO	Classifies event as Site Area Emergency (1.1-S1)	______

XII. Crew Critical Tasks

	Task	SAT/UNSAT
1)	Manually isolate HPCI before 2 areas exceed Maximum Safe Radiation or Temperature levels.	
2)	Prevents ADS actuation when Rx level reaches -120".	
3)	Emergency depressurizes RPV based upon not being able to maintain reactor water level above -162, but before reaching -190"	
4)	Restores / maintains water level above TAF	

XIII. SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER HLTS-13

6 Total Malfunctions Inserted; List: (4-8)

1) HPCI steam line break
2) RBCCW 3A pump trips
3) 3A Recirc. high vibration
4) 3A Recirc pump suction valve fails open and will not close
5) Failure of ADS/SRV 1-22
6) Drywell Leak

3 Malfunctions That Occur After EOI Entry; List: (1-4)
1) CRD pump 3B fails to start
2) CRD pump 3A trips
3) RCIC 71-8 fails to open

1 Abnormal Events; List: (1-3)
1) SRV fails open

2 Major Transients; List: (1-2)
1) Loss of all high pressure makeup
2) Drywell Leak

3 EOIs used; List: (1-3)
1) EOI-1
2) EOI-2
3) EOI-3

2 EOI Contingencies Used; List: (0-3)
1) C1
2) C2

90 Run Time (minutes)

45 EOI Run Time (minutes); 50 % of Scenario EOI Run Time

3 Crew Critical Tasks (2-5)

Yes Technical Specifications Exercised (yes/no) - Technical Requirements Manual

XIV. Shift Turnover Information

Equipment out of service/LCOs: 3C RHR Pump is out of service. T.S 3.5.1.A.1, 3.6.2.3, 3.6.2.4, 3.6.2.5 have been entered. Unit 2 is 6 hours into a seven day LCO. Appendix R LCO addressed and in LCO tracking.

Operation/Maintenance for the Shift: Unit 3 is at 79% power, Alternate EHC Pumps per section 6.3 of OI 47A. Increase reactor power to 90% using Recirc flow (GOI-100-12,step 5.132) at 8 Mwe. per minute. Continue with 3-SR-3.5.1.7 which is in progress and is complete up to Step 7.11 (HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure). Loop II RHR has been vented within the hour in preparation for placing Torus cooling in service.

Unusual Conditions/Problem Areas: 3-FCV-73-36 seal-in circuit has been disabled per step 7.6 of 3-SR-3.5.1.7

TVA

Browns Ferry Nuclear Plant

Unit 3

Surveillance Procedure

3-SR-3.5.1.7

HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure

Revision 0044

Quality Related

Level of Use: Continuous Use

Effective Date: 12-17-2007

Responsible Organization: OPS, Operations

Prepared By: MICHAEL S. RICE x6934

Approved By: John T. Kulisek

Current Revision Description

Pages Affected: 18, 26, 33, 34, 35, 36, 71, 72, 73, 75, 96, 104

Type of Change: Revision Tracking Number: 048

PCRs: 07003919

Revised the stroke time criteria for the 3-FCV-73-18 valve to have a normal stroke time range of 0.8 to 2.2 seconds and a maximum allowable stroke time range of 3.0 seconds. This change in accepance criteria was evaluated and approved for use per 0-TI-383 Evaluation 07-1-IST-073-337.
Added instruction to restroke 3-FCV-73-18 if initial stroke time is less than maximum allowable but outside the normal range. (PCR 07003919)

Added instruction to contact Duty Maintenance Manager if 3-FCV-73-18 was restroked and to record the time. Per the OM Code, the restroked valve has to be evaluated within 96 hours.

Added new Illustration 1, Process for Stroke Timing Valves Per the ASME OM Code.

Added SR key number to Attachment 1 for scheduling.

Added instruction in Attachment 5 to contact OPS immediately if any evaluation results are found to be NOT Acceptable.

Added new Attachment 10, ASME OM Code Restroke TIme Record Form. (PCR 07003919)

Table of Contents

Table of Contents (continued)

1.0 INTRODUCTION

1.1 Purpose

A. This procedure verifies the following High Pressure Coolant Injection (HPCI) System Technical Specification (Tech Spec) surveillance requirements (SR):

1. The HPCI main and booster pump set must be capable of pumping 5,000 gpm against a simulated system head corresponding to reactor pressure in order to satisfy SR 3.5.1.7.

2. HPCI discharge piping must be vented to meet SR 3.5.1.1 in lieu of performing 3-SR-3.5.1.1 (HPCI) for the HPCI System if deemed necessary by the Unit Supervisor (US).

3. This surveillance performs ASME OM Code Inservice Test (IST) Program testing of HPCI pumps and valves in order to satisfy Tech Spec 5.5.6 program requirements.

4. This surveillance provides overlap testing of the HPCI minimum flow valve open and close functions to demonstrate compliance with SR 3.3.5.1.2 for Table 3.3.5.1-1 Function 3f and SR 3.3.5.1.6.

B. This procedure also verifies the following additional licensing, INPO, and Fire Protection Report (FPR) testing requirements:

1. Time-to-rated-flow testing is performed once an operating cycle or whenever HPCI governor control system (GCS) corrective maintenance is performed in order to satisfy a unit startup licensing commitment. This testing is **NOT** specifically required by TS or IST Program requirements.

2. This procedure also accomplishes overspeed trip tappet trip valve assembly testing recommended by INPO to ensure that the trip mechanism is **NOT** binding. This testing is accomplished when the HPCI turbine is cold and then when it is warm.

3. This surveillance is utilized to verify BFN FPR testing requirements which demonstrate HPCI function operability.

1.2 Scope

A. This surveillance verifies the HPCI turbine, main and booster pump set, and supporting equipment (e.g., gland seal condenser) are capable of delivering 5,000 gpm against a simulated system head corresponding to reactor pressure.

1. This testing is accomplished by using startup test data which conservatively approximates the required discharge head needed to overcome system piping resistance and produce 5,000 gpm flow. This pressure is added to the reactor steam dome pressure and is used as the minimum discharge head required to satisfactorily meet SR 3.5.1.7.

2. The HPCI turbine is started and system flow is throttled back to a condensate storage tank until a 5,000 gpm flow rate is attained while verifying that the minimum, required discharge pressure can be obtained.

B. The same venting methodology utilized in 3-SR-3.5.1.1 (HPCI) is also used in this surveillance to provide an alternate means of venting HPCI discharge piping to comply with SR-3.5.1.1 for the HPCI System. This venting is performed at the discretion of the US in lieu of performing 3-SR-3.5.1.1 (HPCI).

C. This surveillance in conjunction with SRs/SIs listed as being ASME type in Surveillance Program Matrix fully implements the ASME OM Code IST Program required by Tech Spec 5.5.6.

Satisfactory completion of this surveillance verifies Tech Spec 5.5.6 compliance for the following valves:

Valve	Test Description
ISV-73-23	HPCI turbine discharge pressure monitored to ensure that valve is sufficiently open to perform its intended function.
CKV-73-603	HPCI turbine discharge pressure monitored to ensure that valve is sufficiently open to perform its intended function.
CKV-73-559	The minimum flow valve is opened and the discharge pressure drop is monitored to verify that this check valve is opening to pass bypass flow.
FCV-73-18	This fast-acting valve's closure time is monitored.

1.2 Scope (continued)

D. When Tech Spec and IST Program testing is accomplished, the turbine is shutdown with the system throttled to simulate a system head corresponding to reactor pressure.

1. If time-to-rated-flow testing is required, the turbine lube oil system and turbine casing are given sufficient time to drain and cool to ambient temperature, respectively. Once these two conditions are met, the system is configured for a cold, quick turbine start and manual HPCI initiation performed.

2. The time to reach 5,000 gpm flow against a simulated system head corresponding to reactor pressure is verified to be ≤ 30 seconds.

E. This surveillance in conjunction with 3-SR-3.3.5.1.6 performs the following BFN FPR, Volume 1, Appendix R Safe Shutdown Program (Section V - Testing and Monitoring) testing to verify that:

1. FCV-73-18 automatically opens and remains open during turbine startup and operation,

2. FCV-73-18 closes when a manual turbine trip is initiated from the main control room,

3. FCV-73-30 automatically closes when HPCI flow is greater than approximately 1250 gpm,

4. FCV-73-30 automatically opens when HPCI flow is less than approximately 700 gpm, and

5. HPCI turbine and pump set operate per design during manual turbine startup and operation.

1.2 Scope (continued)

F. This surveillance satisfies SR 3.3.5.1.5 for Table 3.3.5.1-1 Function 3f by functionally verifying that HPCI MIN FLOW VALVE FCV-73-30 closes when HPCI pump set is operating above approximately 1250 gpm flow.

This testing is accomplished during turbine startup. FCV-73-30 is initially opened to provide a minimum bypass flow path. When the turbine is started, HPCI flow rises to the recirculation flow path back to the condensate storage tank. The rising flow closes FCV-73-30 when an approximately 1250 gpm flow rate is reached.

This surveillance also functionally verifies that FCV-73-30 will open when HPCI flow is reduced below approximately 700 gpm flow.

This testing is accomplished when the ASME OM Code IST Program testing is almost completed. A jumper is installed to allow FCV-73-30 to open when a low flow signal is present. HPCI turbine speed is reduced with the flow indicating controller in manual and HPCI pump set flow is reduced by throttling FCV-73-35 in the close direction until HPCI flow drops below approximately 700 gpm.

1.3 Frequency

A. This surveillance shall be performed once every 92 days when required by plant conditions or whenever GCS corrective maintenance is performed which could affect the GCS function. This SR shall be performed as required to satisfy BFN GL 89-10 Program requirements.

B. [NRC/C] [NER/C] This surveillance is to be used for post-maintenance testing to verify HPCI operability if the Governor Control System components require corrective maintenance. [LER 296/85003] [INPO SOER 81-013]

1.4 Applicability

The surveillance requirements of this procedure are applicable in Mode 1. Modes 2 and 3 are also applicable except when RPV steam dome pressure ≤ 150 psig.

2.0 REFERENCES

2.1 Technical Specifications

Section 3.5.1, ECCS - Operating

2.2 Updated Final Safety Analysis Report

Section 6.3, Summary Description - Core Standby Cooling Systems

Section 6.4.1, High Pressure Coolant Injection System Description

Section 6.6, Inspection and Testing

Section 7.4, Core Standby Cooling System and Instrumentation

2.3 Plant Instructions

0-OI-65, Standby Gas Treatment System

3-OI-73, High Pressure Coolant Injection System

3-SI-3.1.5, HPCI Pump Performance

3-SI-3.1.12, HPCI System Pump Baseline Data Evaluation

3-SI-3.2.1, ASME Section XI Valve Performance

3-SR-3.3.5.1.5(F), High Pressure Coolant Injection System Pump Minimum Bypass Flow Indicating Switch Calibration

3-SR-3.6.2.1.1, Suppression Chamber Water Temperature Check

3-SR-3.5.1.1 (HPCI), Maintenance of Filled HPCI Discharge Piping

0-TI-230, Predictive Monitoring Program.

0-TI-280, Calculations of Flow Transmitter Output for Use With ASME Section XI

SPP-8.1, Conduct of Testing

SPP-10.3, Verification Program

2.4 Plant Drawings

3-47E812-1 and -2, HPCI System Flow Diagram

3-47E610-73-1 and -2, HPCI System Mechanical Control Diagram

2.4 Plant Drawings (continued)

3-45E714-1 through -4, 250V DC RMOV Bd Schematic Diagram

3-45N3675-2, Panel 9-39 Wiring Diagram

3-45N3635-19, Local Instrument Panels Connection Diagram

3-730E928-1 through 5, -7 and -8, HPCI System Elementary Diagram

2.5 Vendor Manuals

BFN-VTM-G080-9270, BFN Unit 3 - Terry Model CCS HPCI Turbine Operation and Maintenance Manual

BFN-VTM-B580-0010, Byron Jackson Technical Instructions High Pressure Coolant Injection Pumps

2.6 Other Documents

NRC Inspection Report 82-13

Licensee Event Report 296/85003, Inoperability of HPCI System

Licensee Event Report 259/8232, Operator Notification

STI-15, HPCI Startup Test Instruction

Browns Ferry Nuclear Plant Fire Protection Report, Volume 1, Appendix R Safe Shutdown Program

INPO SOER 89-001, Testing of Steam Turbine/Pump Overspeed Trip

GE SIL No. 336 R1, Surveillance Testing Recommendations for HPCI and RCIC Systems

TVA Program Plan Implementation of NRC Generic Letter 89-10

Memorandum from D. Baker, GENE Power Ascension Operations Manager, to M. Bajestani, BFN Technical Support Manager, dated June 25, 1991 (RIMS R40 910716 805)

PGC-007-073-0, HPCI Operation Time Required to Raise Suppression Pool Temperature 1 deg F (R40 910629 984)

INPO SOER 81-013, Concurrent Loss of High Pressure Core Cooling Systems

NRC Information Notice 91-50, A Review of Water Hammer Events After 1985

2.6 Other Documents (continued)

GE SIL No. 106 R2, Suppression Pool Temperature Monitoring and Control

GE SIL No. 392 R1, Improved HPCI Turbine Mechanical-Hydraulic Trip Design

NRC Information Notice 93-67, Bursting of High Pressure Coolant Injection Steam Line Rupture Discs Injures Plant Personnel

SEOPR 96-0-073-2, HPCI Turbine Administrative Vibration Limits

NEDC-32751P, Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 and 3 (RIMS R08-980316-888)

TVA-BFN-TS-384, Technical Specification (TS) Change TS-384 Request for License Amendment for Power Uprate Operation (RIMS R08-980316-888)

GE-NE-B13-01866-39, Summary of System Evaluations and Proposed Changes to Design Criteria Documents (RIMS W79-980427-005)

3.0 PRECAUTIONS AND LIMITATIONS

A. [NRC/C] LCO 3.5.1 requires the HPCI System to be OPERABLE in Mode 1 and Modes 2 and 3 except when RPV steam dome pressure ≤ 150 psig.

1. Entry into associated LCO 3.5.1 CONDITIONS AND REQUIRED ACTIONS is **NOT** initially required provided the HPCI function is demonstrated operable no later than 12 hours after reactor steam dome pressure reaches rated pressure from startup.

2. However, this surveillance removes the HPCI function from service (e.g., trip turbine using 3-HS-73-18A) for short duration's while performing surveillance testing.

3. Consequently, entry into LCO 3.5.1 is administratively controlled within this surveillance by declaring the HPCI function temporarily inoperable during testing and verifying that LCO 3.5.1 CONDITIONS AND REQUIRED ACTIONS have been met including tracking HPCI function inoperability in Narrative Logs. [NCO 89-0216-002]

B. If maintenance other than what is provided in this surveillance procedure becomes necessary, a work order should be generated.

C. Consult Attachment 9 for Panel 3-9-3 annunciators which will alarm during performance of this surveillance.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

D. HPCI turbine operation below 2,400 rpm for extended periods except during turbine startup and shutdown can result in inadequate oil pressure from the turbine driven oil pump, higher system vibration, excessive exhaust line check valve wear, or overheating of turbine driven oil pump when operating at low rpm with auxiliary oil pump (AOP) running.

E. Suppression pool temperature will rise approximately 1°F every three minutes during testing of HPCI System.

The temperature must **NOT** be allowed to exceed 105°F and must be returned to ≤ 95°F within 24 hours after securing the HPCI Turbine as required by LCO 3.6.2.1. The suppression pool temperature shall be monitored every 5 minutes and recorded in accordance with 3-SR-3.6.2.1.1, Suppression Chamber Water Temperature Check.

F. Pressure suppression chamber (PSC) water shall **NOT** be used as the HPCI water supply to perform this test because of its lower quality and the potential water hammer risk when PSC water level is **NOT** high enough to swap HPCI suction to the PSC.

G. The suppression pool shall be maintained at -5.5 to -2 inches as indicated by 3-LI-64-54A or 3-LI-64-66 on Panel 3-9-3.

H. Personnel stay time in HPCI Room during HPCI System operation should be minimized if excessive exposure to noise, heat, or radiation is anticipated.

I. HPCI AOP operation should be minimized when HPCI System is in standby readiness or following HPCI System shutdown. When AOP is operating, turbine stop valve is held full open. If HPCI System is then manually or automatically initiated, a HPCI turbine overspeed trip or high steam line flow isolation may occur.

J. A radiation work permit (RWP) may be required for all personnel located in the HPCI Room participating in the performance of this SR. RADCON shall be consulted prior to turbine roll in order to determine the appropriate RWP requirements.

K. Corrective Action shall be dispositioned in accordance with SPP-8.1, Conduct of Testing.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

L. HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 must be verified in AUTO mode of operation with flow setpoint at 5,000 gpm if an automatic HPCI initiation occurs during performance of this surveillance.

The HPCI flow controller 3-FIC-73-33 is a "FLOW X10" controller, 5000 gpm on the controller digital display will read 500. The steps in this procedure which list a flow value will be displayed as follows " flow as read on the digital display followed by the actual flow in gpm" i.e. a flow of 1250 gpm is shown as "125 (1250 gpm)" a flow of 5000gpm is shown as "500 (5000 gpm).

M. The risk of steam emission to the surrounding area rises if a rupture disk breaks during initial startup of turbine. Therefore, the number of personnel in HPCI Room should be minimized until stable operation is achieved.

N. The identification number and calibration date for new test equipment, along with step numbers for which it was used, shall be noted in the remarks Section of Surveillance Procedure Review Form if during performance of this surveillance it becomes necessary to change test equipment.

O. The HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 will **NOT** open automatically when low system flow is sensed unless a HPCI initiation signal is present.

P. HPCI pump and bearing temperatures should be monitored periodically using HPCI/RCIC/RFW TEMPERATURES 3-TR-73-54 on Panel 3-9-47 or ICS to ensure that temperatures are stable or **NOT** rising rapidly. Turbine shutdown should be initiated if any oil temperature reading exceeds 155°F or any other unsatisfactory oil condition is observed by personnel located in the HPCI Room.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. [NRC/C] The HPCI System will be placed in configurations that make it susceptible to overspeed tripping and motive steam loss should an initiation signal occur when one of the following conditions is present:

1. When HPCI TURBINE STOP VALVE 3-FCV-73-18 is cycled open, the GCS ramp generator will time out in approximately 12-13 seconds. If HPCI System receives an automatic initiation signal after the ramp generator times out and is reset by closure of turbine stop valve, a turbine overspeed or a high steam line flow isolation may occur.

2. Manipulations of mechanical overspeed trip assembly (e.g., verifying freedom of movement) may result in closure of turbine stop valve at a time when it is required to be open for turbine operation.

3. Placing HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 in manual mode or changing its flow setpoint may **NOT** permit HPCI System to automatically achieve design flow in the event of an automatic initiation.

4. Manual initiation of a turbine trip using 3-HS-73-18A will prevent turbine stop valve from opening while trip push-button is depressed.

Since the above conditions may lead to a HPCI overspeed trip or loss of steam supply if a HPCI initiation should occur during surveillance testing, the HPCI System will be administratively removed from operable service to ensure that RPV injection capability is maintained at all times when surveillance testing could result in an overspeed condition of the HPCI turbine. [NCO 89-0216-002]

R. HPCI TURBINE STOP VALVE 3-FCV-73-18 operation should be observed for visual and/or audible signs of a fast opening/closing transient during turbine startup. Site Engineering and/or Mechanical Maintenance must be notified if this type of transient occurs in order to evaluate the need for balance chamber adjustments.

S. [II/F] Prior to initiating HPCI System and adding heat energy to suppression chamber, the Unit Supervisor will evaluate need of placing Residual Heat Removal System in suppression pool cooling mode to avoid the possibility of thermal stagnation during sustained heat additions. [II-B-91-129]

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. The BFN ASME OM Code Ten Year Program for monitoring pump flow and total developed head requires the use of measuring instruments capable of $\pm$ 2% accuracy at full scale (FS) and having a maximum range which does **NOT** exceed three times the maximum, expected process value. This accuracy requirement is implemented for HPCI flow measurements by directly measuring output of HPCI flow transmitter using the Integrated Computer System (ICS). Existing local HPCI pump set suction and discharge pressure gages satisfy ASME OM Code accuracy and range requirements and do **NOT** require substitution with more accurate instrumentation.

Turbine speed indication (SI-73-51) on Panel 9-3 exceeds the 2% FS accuracy requirement based on a review of two, as-found calibration checks performed over a three year period. However, HPCI tachometer drift problems have made it necessary to utilize local, hand held M&TE instrumentation (e.g., stroboscope) to ensure that accurate turbine speed settings are established for ASME OM Code purposes.

U. The ASME OM Code data recorded by this surveillance should be reviewed and recorded in accordance with 3-SI-3.1.5 within 96 hours of completion of this surveillance.

V. ASME OM Code data collection requires that HPCI pump set be operated at a predetermined flow rate and speed when discharge pressure readings are taken. While the flow rate may be adjusted anywhere within the allowable range specified (e.g., 4950 to 5050 gpm), UO must attempt to maintain the flow rate as close as possible to midrange. This ensures that discharge pressure readings do **NOT** vary significantly due to operating point changes from performance to performance of this surveillance unless an actual deficiency exists. UO must also ensure that turbine speed is adjusted as close as possible to ASME OM Code test value of 3,800 rpm within the range 3790 to 3810 rpm. Averaging techniques are acceptable.

W. Any control room ICS console may be utilized for collecting ICS data specified by this surveillance. If ICS console originally selected fails to operate properly during surveillance performance, another ICS console(s) may be used for completion of test activities provided failure is isolated to console in use. If an alternate ICS console(s) is used, then change(s) shall be noted in post-test remarks Section of Attachment 1.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

X. HPCI System ICS startup transient data may be displayed and printed as follows:

1. **PRESS** CANC key on ICS console keyboard.
2. **SELECT** GROUP.
3. **SELECT** MODIFY NON SYSTEMS GROUP.
4. **SELECT** test group to be modified (e.g., Test 15).
5. **ENTER** the following HPCI data points using F6 key to select fields:

FIELD	POINT ID
03	73-31
06	73-33
09	73-51
12	DIG027

6. **DELETE** remaining data points from group by selecting Field 15 and repeatedly pressing ENTER key until remaining data points are removed.
7. **PRESS** F3 to save group redefinition.
8. **PRESS** F1 to display group.
9. **SELECT** OTHER GROUP FNCTS.
10. **SELECT** GROUP GRAPH 4 PTS ON 1 PLOT (Selection #4).
11. **PRESS** F2 to continue.
12. **SETUP** desired start data and time using F3 key.
13. **SETUP** time axis resolution of 1 minute per graticule.
14. **PRESS** Print Screen key to print plot of transient startup data.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Y. Pressure switch 3-PS-73-47A controls AOP operation and resets at approximately 35 psig lowering oil pressure.

AOP Operation causes 3-CKV-73-708 to close and the main, gear-driven oil pump discharge pressure rises to its dead head value. This action causes 3-PS-73-47A to initiate at approximately 92 psig rising oil pressure which in turn stops the AOP. The AOP will continue cycling on and off in this manner until the main, gear-driven oil pump slows sufficiently to prevent initiation of 3-PS-73-47A.

Z. 3-FCV-73-6A and 3-FCV-73-6B close during HPCI turbine operation disabling the drain path for inlet condensing pot (3-MCP-73-5).

Isolation of the drain path will eventually result in filling the inlet condensing pot and HPCI TURBINE INLET DRAIN POT LEVEL HIGH 3-LA-73-5 (3-XA-55-3F, window 26) will alarm. This is an expected condition and will **NOT** result in turbine damage because steam flow into the turbine will prevent any excessive accumulation of condensate in the inlet piping.

AA. 3-FCV-73-18 should be monitored for one continuous smooth action from full closed to full open position.

The monitoring may be performed by either local visual line of sight, video camera or video recorder, to ensure that once the 3-FCV-73-16 valve is opened and Auxiliary Oil pump starts, the valve does **NOT** behave erratically (i.e., suddenly opening then closing and finally ramping open). (BFNPER 99-04221)

BB. During Starting, shutdown and tripping of the HPCI Turbine a second operator should be utilized to assist in monitoring alarms and parameters for abnormal conditions.

CC. Local vibration readings of the HPCI turbine and pump bearings (using portable M&TE) may be obtained during each performance of this SR.

DD. Caution tags are available as prerequisites and are placed in Attachment 3 to ensure that plant personnel do **NOT** operate these components prior to completion of time-to-rated-flow testing.

EE. The Critical Steps warning represents a step or series of steps for an activity which requires additional focus, attention, and increased awareness. The Operator performing these steps for the activity needs to ensure the Unit Supervisor and other Control Room staff are aware of the evolution. PEER checks are required for this activity and short briefs need to be made prior to performing the evolution. Included in the briefs are worst case scenario and contingencies.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

FF. Step 7.0[21] serves to verify that the mechanical overspeed trip tappet assembly is functioning properly and **NOT** binding when the HPCI turbine is at its nominal, design operating temperature, and the overspeed trip automatic reset time is approximately 4-6 seconds based on available turbine rpm coastdown data and GE SIL 392 R1 recommendations.

GG. The discharge flow verification can be affected by how much air has been introduced into the system and the fact that the discharge line is vented for 1 minute through a closed drain prior to the discharge flow verification. The most opportune time for this check is when the vent valve is opened when the initial flow can be seen due to the turbulence initially created with the sightglass empty.

Sight glass flow indication can be verified by any of the following: (Flashlight should be used to assist in determination.)

1. Initial turbulence or bubbles seen through the sightglass when the 3-HS-73-63 push-button is depressed, followed by the sight glass filling and the bubbles dissipating.
2. This occurs very fast therefore the operator must be monitoring prior to depressing 3-HS-73-63.
3. Flowing water seen in sightglass
4. Lowering temperature gradient over the Ten minute period as seen by the performance of Attachment 2 Step 1.0[23], if 3-FCV-73-45 is determined to be seated.
5. Rising temperature over the Ten minute period as seen by the performance of Attachment 2 Step 1.0[23], if 3-FCV-73-45 is determined to have leakage.

HH. When timing the 3-FCV-73-18 valve, the 3.0 second requirement is such a tight tolerance that using a stopwatch does **NOT** leave room for any errors. The use of a Yokogawa recorder may be used as desired by System Engineering. The Yokogawa Recorder can be connected in Panel 3-9-3 or Panel 3-9-39. Only one location is required for testing, but may be connected to both for additional data as required. System Engineering should determine the location to be used. The Preferred location is Panel 3-9-3 for consistency and communication. System Engineering will determine the location to be used.

II. When using the Yokogawa Recorder for measuring 3-FCV-73-18, communications and countdown methods are to be established to ensure recorders are on prior to operating the valve.

Date ____________

NOTES
1) Section 4.0 through Step 7.0[7], sets up the HPCI Surveillance for the Dynamic Run. These steps may be performed up to 24 hours prior to the actual HPCI turbine operation. Care should be given to all LCO entries. The latest revision of this surveillance should be re-verified.
2) If this test continues for more than one shift, a Pre-job brief will be required for all new personnel involved.

4.0 PREREQUISITES

[1] **VERIFY** this copy of 3-SR-3.5.1.7 is the most current revision. ________

[2] **VERIFY** the HPCI System is in a standby readiness configuration in accordance with 3-OI-73, High Pressure Coolant Injection System. ________

[3] **VERIFY** the Reactor steam dome pressure is ≥ 950 psig and ≤ 1040 psig. ________

[4] **VERIFY** at least 2 turbine bypass valves full open (N/A if Main Turbine is on-line). ________

[5] **IF** ICS will be utilized to collect HPCI flow rate data, **THEN**

CHECK that no gross instrument channel failure is present by noting that HPCI flow rate on the ICS-displayed (single value display (SVD 73-33) or the HPCI System mimic.), is within 100 gpm of flow rate indicated on HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33. ________

[6] **VERIFY** the following Operations personnel as a minimum are available to perform this procedure. (This does **NOT** include IV's or multiple shift performance or Peer Checking requirements.)

UO: 2
AUO: 4 ________

Date ____________

4.0 PREREQUISITES (continued)

[7] **NOTIFY** each organization listed below.

AND

REQUEST the number of Qualified personnel from each organization to be available to support and perform their associated activity: (If possible give a possible time reference when personnel will be required.)

A. RADCON (1) will be available to

- Determine RWP requirements
- Will be available to monitor for airborne contamination and radiation levels in the HPCI Room during the startup and operation of the HPCI Turbine. ________

B. Electrical Maintenance (3 EM's) will be available to

- Turbine/main pump speed readings using hand held instrumentation locally in the HPCI Room during the Startup and Operation of the HPCI Turbine.
- Install jumper for Min Flow Valve.
- Installation of Yokogawa Recorder for 3-FCV-73-18 if used. ________

C. Mechanical Maintenance (2 MM's) will be available to:

- Obtain a HPCI lube oil sample after turbine shutdown prior to securing the Aux Oil Pump
- Adjust 3-PCV-073-0501
- Perform MPI-0-073-TRB001 if required. ________

Date ____________

4.0 PREREQUISITES (continued)

[8] **CONTACT** System Engineer or Duty Maintenance Manager to determine if the following are to be performed:

[8.1] **RECORD** below if local vibration readings of HPCI pump bearings using portable M&TE (Step 7.0[15.10]) are required:

YES / NO (Circle one) ________

[8.2] **RECORD** below if verification of Time To Achieve Rated Flow And Pressure is required: (Normally performed once every two years after the refueling outage.)

YES / NO (Circle one) ________

[9] **IF** local vibration readings of HPCI pump bearings using portable M&TE are **NOT** required, **THEN**

N/A Step 7.0[15.10]; (Otherwise **N/A** this step). ________

[10] **IF** verification of time to achieve rated flow is required, **THEN**

VERIFY that a Caution Order and associated tags have been prepared to control operation of HPCI AUXILIARY OIL PUMP 3-PMP-73-47 and HPCI PUMP CST TEST VLV 3-FCV-73-35. (Otherwise **N/A**) ________

[11] **CHECK** that a control room ICS console display is available to monitor HPCI discharge pressure, flow, and manual initiation status as a function of time; (**REFER TO** Step 3.0W and 3.0X). (**N/A** if ICS is **NOT** available) ________

[12] **PLACE** "TESTING/MAINTENANCE" alarm window frame(s) around the alarm windows listed in Attachment 9. ________

[13] **CONTACT** System Engineering to determine the following:

[13.1] **CHECK** the Timing Method to be used for 3-FCV-73-18.

STOPWATCH ☐ YOKOGAWA RECORDER ☐ ________

Date ____________

4.0 PREREQUISITES (continued)

[14] **CHECK** the Location(s) were the Yokogawa Recorder are to be installed. (**N/A** if Yokogawa Recorders are **NOT** used.)

Panel 3-9-3 ☐ Panel 3-9-39 ☐ ________

[15] **IF** Yokogawa Recorders are to be used, **THEN**

NOTIFY Electrical Maintenance to install the Recorders per Section 1.0 of Attachment 8 in the location(s) determined by System Engineering. (Otherwise **N/A**) ________

5.0 SPECIAL TOOLS AND EQUIPMENT RECOMMENDED

5.1 Recommended Tools

- (1) Banana jack jumper
- Carpenter's ruler or similar tool for measuring HPCI lube oil tank and booster pump oil levels
- Tape
- Screwdriver for lifting leads.
- Crescent wrench for adjusting lube oil pressures.

Date ____________

5.2 Recommended Measuring and Test Equipment (M&TE)

NOTE

The equipment data listed below may **NOT** be available at the time this procedure starts. The data listed below may be collected at a later time but prior to using the equipment to ensure calibrations requirements are met.

[1] **ENTER** information where required. Vibration M&TE accuracy and frequency response range are controlled by the BFN Vibration Program and have been verified to meet the listed requirements. **VERIFY** required range and accuracy for remaining M&TE by reviewing calibration sheets.

Parameter Measured	Recommended Instrument (or equivalent instrument)	Required Range	Required Accuracy	Frequency Response Range	Calibration Due Date	M&TE ID
Speed	MC Instruments Digital Probe Tachometer (Model 112) OR CSI Stroboscope (Model No. 444)	5428.6 rpm minimum	± 2% of calibrated range	N/A		
Vibration	CSI Model 2100 series vibration meter or equal	N/A	± 5% of calibrated range	21.11-1000 Hz minimum		
Time	(Local) digital or analog stopwatch	N/A	± 1 second	N/A	N/A	
	(MCR) digital or analog stopwatch	N/A	± 1 second	N/A	N/A	
	Yokogawa Recorder (Panel 3-9-3 if used for 3-FCV-73-18)					
	Yokogawa Recorder (Panel 3-9-39 if used for 3-FCV-73-18)					
Temperature	Omega Model HH22 digital thermometer (surface and area air temps are required)	50°F to 300°F minimum	± 1 °F	N/A		

6.0 ACCEPTANCE CRITERIA

A. Responses which fail to meet the acceptance criteria stated below shall constitute unsatisfactory surveillance procedure results and require immediate notification of Unit Supervisor (US) at time of failure.

The following acceptance criteria shall be demonstrated as required by this surveillance:

1. HPCI System is vented from high POINT VENT by observing continuous water flow from vent when venting is performed at US discretion in lieu of 3-SR-3.5.1.1 (HPCI).

2. HPCI pump set delivers 5,000 gpm flow at a minimum discharge pressure 110 psi above reactor pressure.

3. [NRC/C] The HPCI System achieves 5,000 gpm flow at a minimum discharge pressure 110 psi above reactor pressure within 30 seconds from a cold, non-oil-primed, turbine quick start. (Only required following a refueling outage or anytime maintenance affects Governor Control System operation.) [LER 296/85003]

4. The differential pressure developed by the HPCI pump set shall be $\geq$ 1034 psid and $\leq$ 1201 psid when HPCI pump set is operating at 4950-5050 gpm flow and 3790-3810 rpm main pump speed.

5. HPCI PUMP MIN FLOW VALVE FCV-73-30 shall open when HPCI flow rate lowers.

6.0 ACCEPTANCE CRITERIA (continued)

6. The following valves shall comply with ASME OM Code Inservice Test (IST) acceptance criteria stipulated below:

Valve	Acceptance Criteria
3-FCV-73-18	Valve shall close within 3.0 seconds or less when a close signal is present.
3-ISV-73-23	Valve shall open sufficiently to perform its intended function by noting that turbine exhaust pressure does **NOT** exceed 40 psig when turbine is operating at or near rated conditions.
3-CKV-73-559	Valve shall open sufficiently to perform its intended function by noting at least a 70 psi drop in the HPCI pump set discharge pressure when HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 is opened while HPCI pump set is operating at or near rated conditions.
3-CKV-73-603	Valve shall open sufficiently to perform its intended function by noting that turbine exhaust pressure does **NOT** exceed 40 psig when turbine is operating at or near rated conditions.

B. Steps which determine the above criteria are designated by (AC) next to initial blank.

Date ____________

7.0 PROCEDURE STEPS

[1] **CHECK** that the following initial conditions are satisfied:

A. Precautions and limitations in Section 3.0 have been reviewed. ________

B. Prerequisites listed in Section 4.0 are met. ________

C. The following annunciators are RESET:

- HPCI TURBINE INLET DRAIN POT LEVEL HIGH 3-LA-73-5 (3-XA-55-3F, window 26) ________
- HPCI TURBINE TRIPPED 3-ZA-73-18 (3-XA-55-3F, window 11) ________
- HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5) ________
- HPCI TURB EXH DRAIN POT LEVEL HIGH 3-LA-73-8A (3-XA-55-3F, window 33) ________

D. The following indicating lights are EXTINGUISHED:

- HPCI AUTO INIT 3-IL-73-59 ________
- HPCI AUTO ISOL LOGIC A 3-IL-73-58A ________
- HPCI AUTO ISOL LOGIC B 3-IL-73-58B ________
- HPCI TURBINE TRIP RX LVL HIGH 3-IL-73-18B ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

This surveillance will make HPCI INOP.

[2] **PERFORM** the following:

[2.1] The US and Unit Operator (UO) have been provided with copies of this SR. ________

[2.2] UO has reviewed surveillance test scope including wire lifts and jumper placements. ________

[2.3] **OBTAIN** permission from US to perform this surveillance. ________ US

[2.4] [NRC/C] **NOTIFY** UO that this surveillance is commencing. [RPT 82-16, LER 259/8232] ________

[3] **RECORD** date and time started, reason for test, and plant conditions on Attachment 1, Surveillance Procedure Review Form. ________

[4] **VERIFY** that suitable means of communication (e.g., hand held radios, plant telephone system) will be available between Main Control Room, HPCI Room, and HPCI vent station. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE
Step 7.0[7.5] may be performed in parallel with remaining surveillance steps up to Step 7.0[11] at the discretion and direction of US.

[5] **VENT** the HPCI discharge piping.

OR

VERIFY the HPCI discharge piping has been vented within the last 24 hours by one of the following:

[5.1] **VERIFY** 3-SR-3.5.1.1(HPCI) has been performed within the last 24 hours. (**N/A** if 3-SR-3.5.1.1(HPCI) has **NOT** been performed.) ________

[5.2] **VENT** the HPCI discharge piping by performing Attachment 2. (**N/A** if 3-SR-3.5.1.1(HPCI) has been performed.) ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES
1) Three AUOs will be required in HPCI Room for performance of lube oil checks and one Mechanical Maintenance available to adjust 3-PCV-073-501.
2) A crescent wrench may be required to position lube oil system stopcock valves.

[6] **PERFORM** the following lube oil system and control/stop valve checks and adjustments:

[6.1] **CHECK** the following oil levels locally:

- HPCI turbine lube oil reservoir level is per Attachment 6. ________
- Oil level in HPCI booster pump inboard and outboard bearing oil sight glasses is per Attachment 6. ________

[6.2] **PERFORM** the following:

- [NRC/C] **REVIEW** Step 3.0Q for additional background information regarding HPCI System removal from operable service. [NCO 89-0216-002] ________
- **IF** Yokogawa Recorder(s) will be used to time 3-FCV-73-18, **THEN**

  VERIFY installation of recorders in the location(s) specified in Step 4.0[13] per Attachment 8. (Otherwise **N/A**) ________

Date ________

7.0 PROCEDURE STEPS (continued)

[6.3] **IF** HPCI System is Operable, **THEN**

PERFORM the following: (Otherwise **N/A**)

[6.3.1] **VERIFY** HPCI System may be removed from operable service.

________ US

[6.3.2] **DECLARE** HPCI System inoperable.

________ US

[6.3.3] **ENTER** appropriate LCO information into Narrative log.

________ US

NOTES

1) The TEST push-button 3-HS-73-47B is located in the HPCI Room at a local control station on the south wall near the AOP.
2) Initial timing of 3-FCV-73-18 must be performed during the FIRST start of the Aux Oil Pump with the oil system cold and de-pressurized.
3) Coordination between the operator starting the aux oil pump and the operator timing the 3-FCV-73-18 valve must be performed to ensure proper timing.
4) Step 7.0[6.4] and Step 7.0[6.5] should be reviewed prior to starting the HPCI Aux Oil Pump. These steps may be signed off after completion of Step 7.0[6.5].

[6.4] **SIMULTANEOUSLY PERFORM** the following:

- **DEPRESS** and **HOLD** the HPCI AUX OIL PUMP 3-HS-073-0047B TEST push-button until Step 7.0[6.15].

  AND

- **START** the local STOPWATCH. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[6.5] **WHEN** Initial movement of local stem is observed on HPCI TURBINE STOP VALVE 3-FCV-73-18, **THEN**

STOP the stop watch, and **RECORD** the time below:

__________ seconds ________

NOTE
A time exceeding 13 seconds for turbine stop valve to begin opening may indicate a problem with function of stop valve or lube oil system.

[6.6] **VERIFY** initial movement for turbine stop valve to begin opening is less than 13 seconds.

<u>OR</u>

IF recorded time is greater than 13 seconds, **THEN**

CONTACT Systems Engineering to determine if diagnostic maintenance activities are required prior to proceeding with testing. ________

[6.7] **CHECK** that HPCI TURBINE STOP VALVE 3-FCV-73-18 indicates OPEN by observing 3-ZI-73-18 position indicating lights. ________

[6.8] **CHECK** that HPCI TURBINE CONTROL VALVE 3-FCV-73-19 indicates OPEN by observing 3-ZI-73-19 position indicating lights. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

[NER/C] Step 7.0[6.9] verifies the turbine stop valve closing function using a Yokogawa Recorder. Starting the recorder should be prior to operating the valve. Therefore a countdown method or other method should be established between the Recorder Operator and the Operator at Panel 3-9-3. must be ready to measure closure time of 3-FCV-73-18 since this is a fast acting valve. These steps may be signed off after completion of Step 7.0[6.9.4]. [INPO SOER 89-001]

[6.9] **IF** a Yokogawa Recorder is to be used to measure 3-FCV-73-18, **THEN**

MEASURE closure time of HPCI TURBINE STOP VALVE 3-FCV-73-18 by performing following: (Otherwise **N/A**)

[6.9.1] **NOTIFY** the Recorder Operator to start the Yokogawa Recorder on the desired point of the countdown. _________

[6.9.2] **DEPRESS** and **HOLD** HPCI TURBINE TRIP 3-HS-73-18A until Step 7.0[6.9.4]. _________

[6.9.3] [NRC/C] **WHEN** HPCI TURBINE STOP VALVE 3-FCV-73-18 is CLOSED as indicated by 3-ZI-73-18 position indicating lights, **THEN**

STOP the Recorder and **RECORD** closure time below: [Appendix R]

3-FCV-73-18 CLOSURE TIME (SEC)		
NORMAL	MEASURED	MAXIMUM
0.8 - 2.2		3.0

____

A. **VERIFY** the stroke time recorded is less than or equal to the maximum value listed. ______(AC)

[6.9.4] **RELEASE** HPCI TURBINE TRIP, 3-HS-73-18A. _________

Date ____________

7.0 PROCEDURE STEPS (continued)

[6.9.5] **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 indicates OPEN after a short time delay by observing 3-ZI-73-18 position indicating lights. ________

[6.9.6] **IF** the stroke time measured in step 7.0[6.9.3] is less than or equal to the maximum listed but outside the normal range, **THEN**

PERFORM the following: (Otherwise N/A)

A. **RECORD** on Attachment 10 the initial measured stroke time from step 7.0[6.9.3] above. ________

B. **RESTROKE** and **TIME** 3-FCV-073-0018 and **RECORD** the restroke time on Attachment 10. ________

C. **VERIFY** the restroke time recorded on Attachment 10 is less than or equal to the maximum value listed. ______(AC)

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

[NER/C] Step 7.0[6.10] verifies the turbine stop valve closing function using a stopwatch. The stopwatch must be ready to measure closure time of 3-FCV-73-18 since this is a fast acting valve. These steps may be signed off after completion of Step 7.0[6.10.3]. [INPO SOER 89-001]

[6.10] **IF** a stopwatch is to be used to measure 3-FCV-73-18, **THEN**

MEASURE closure time of HPCI TURBINE STOP VALVE 3-FCV-73-18 by performing the following: (Otherwise **N/A**)

[6.10.1] **MEASURE** closure time of HPCI TURBINE STOP VALVE 3-FCV-73-18 by performing the following substeps simultaneously:

- **DEPRESS** and **HOLD** HPCI TURBINE TRIP 3-HS-73-18A until Step 7.0[6.10.3].

  AND

- **START** stopwatch at same time trip push-button is depressed. _________

[6.10.2] [NRC/C] **WHEN** HPCI TURBINE STOP VALVE 3-FCV-73-18 is CLOSED as indicated by 3-ZI-73-18 position indicating lights, **THEN**

STOP stopwatch and **RECORD** closure time below: [Appendix R]

3-FCV-73-18 CLOSURE TIME (SEC)		
NORMAL	MEASURED	MAXIMUM
0.8 - 2.2		3.0

____

A. **VERIFY** the stroke time recorded is less than or equal to the maximum value listed. ______(AC)

[6.10.3] **RELEASE** HPCI TURBINE TRIP 3-HS-73-18A. _________

Date ____________

7.0 PROCEDURE STEPS (continued)

[6.10.4] **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 indicates OPEN after a short time delay by observing 3-ZI-73-18 position indicating lights. ________

[6.10.5] **IF** the stroke time measured in step 7.0[6.10.2] is less than or equal to the maximum listed but outside the normal range, **THEN**

PERFORM the following: (Otherwise N/A)

A. **RECORD** on Attachment 10 the initial measured stroke time from step 7.0[6.10.2] above. ________

B. **RESTROKE** and **TIME** 3-FCV-073-0018 and **RECORD** the restroke time on Attachment 10. ________

C. **VERIFY** the restroke time recorded on Attachment 10 is less than or equal to the maximum value listed. ______(AC)

NOTE
The removal of the Yokogawa Recorders may be performed in conjunction with the remainder of the procedure.

[6.11] **IF** a Yokogawa Recorder was used to measure 3-FCV-73-18, **THEN**

NOTIFY Electrical Maintenance to REMOVE the Yokogawa Recorders per Section 2.0 of Attachment 8: (Otherwise **N/A**) ________

Date ___________

7.0 PROCEDURE STEPS (continued)

NOTE

The next section verifies free movement of turbine overspeed trip tappet trip valve assembly prior to turbine operation. The trip knob reset action occurs automatically after a variable time delay with no operator action required.

[6.12] **VERIFY** free movement of turbine overspeed trip tappet trip valve assembly as follows:

[6.12.1] **LIFT** and **HOLD** HPCI TURBINE MECH TRIP VLV 3-XCV-073-0018 trip knob until Step 7.0[6.12.3]. ________

[6.12.2] **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 closes by observing 3-ZI-73-18 position indicating lights. ________

[6.12.3] **RELEASE** HPCI TURBINE MECH TRIP VLV 3-XCV-073-0018 trip knob. ________

[6.12.4] **CHECK** HPCI TURBINE MECH TRIP VLV 3-XCV-073-0018 is reset by observing 3-ZI-73-18 position indicating lights and noting that HPCI TURBINE STOP VALVE 3-FCV-73-18 reopens. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

The Target values and ranges in the table below are for information only. If the target is **NOT** met, then surveillance testing may proceed with concurrence from Systems Engineering and Unit Supervisor. 3-PCV-073-0501 may be adjusted by Mechanical Maintenance to ensure target values are met. Adjustments should be documented in the Post-Test remarks.

[6.13] **CHECK** that HPCI lube oil pressures listed below are within the desired range:

Parameter/Indicator	Indicated Value	Target
HPCI TURB THRUST BRG & EGR PRESS INDR 3-PI-073-0506	psig	≥ 15 psig
HPCI TURB OUTBD JOURNAL BRG SUPPLY 3-PI-073-0508	psig	≥ 10 psig
HPCI TURB INBD BRG SUPPLY PRESS INDR 3-PI-073-0510	psig	≥ 10 psig
HPCI MAIN PUMP BRG & SPEED REDUCER SPLY 3-PI-073-0509	psig	≥ 20 psig
OIL SUPPLY PRESSURE 3-PI-073-0501A	psig	36-40 psig
HPCI OIL FILTER INLET PRESS IND. 3-PI-073-0053A	psig	85-90 psig

____

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

The trip knob reset action occurs automatically after a variable time delay with no operator action required.

[6.14] **TEST** the HPCI Turbine Mechanical Trip Valve as follows:

[6.14.1] **VERIFY** HPCI GOVERNOR CONTROL VALVE CLOSURE BOOSTER VALVE 3-SHV-73-0707 one-half (1/2) turn open.

________ 1st

________ 2nd

[6.14.2] **LIFT** and **HOLD** HPCI TURBINE MECH TRIP VLV 3-XCV-073-0018 trip knob until Step 7.0[6.14.4]. ________

[6.14.3] **ADJUST** the HPCI TURB OIL INLET THR VLV FOR 3-PCV-073-0018C, 3-THV-73-714 as required to obtain:

18-20 psig as indicated on HPCI MECH TRIP VLV INLET PRESS 3-PI-073-0018B. ________

[6.14.4] **RELEASE** HPCI TURBINE MECH TRIP VLV 3-XCV-073-0018 trip knob. ________

[6.14.5] **CHECK** that HPCI TURBINE STOP VALVE 3-FCV-73-18 is OPEN by observing 3-ZI-73-18 position indicating lights. ________

[6.14.6] **CHECK** that HPCI TURBINE CONTROL VALVE 3-FCV-73-19 is OPEN by observing 3-ZI-73-19 position indicating lights. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

After releasing the AUX OIL PUMP 3-HS-073-0047B TEST push-button in Step 7.0[6.15], **ALLOW** at least one minute for oil to drain back to oil tank before performing Step 7.0[6.18].

[6.15] **RELEASE** the HPCI AUX OIL PUMP 3-HS-073-0047B TEST push-button. _________

[6.16] **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 closes by observing 3-ZI-73-18 position indicating lights. _________

[6.17] **CHECK** HPCI TURBINE CONTROL VALVE 3-FCV-73-19 closes by observing 3-ZI-73-19 position indicating lights. _________

NOTE

During the performance of Step 7.0[6.18], close coordination will be required. **REVIEW** Step 7.0[6.18] though Step 7.0[6.21] for clear understanding of the operation of 3-FCV-73-18 and 3-FCV-73-19 upon Aux Oil Pump start.

[6.18] **AFTER** at least one minute from performing Step 7.0[6.15], **DEPRESS** and **HOLD** the HPCI AUX OIL PUMP 3-HS-073-0047B TEST push-button until Step 7.0[6.21]. _________

[6.19] **VISUALLY CHECK** that turbine control valve 3-FCV-73-19 approaches or reaches the full open position while the turbine stop valve 3-FCV-73-18 is closed. _________

[6.20] **VISUALLY CHECK** that when the turbine stop valve 3-FCV-73-18 begins to open, the turbine control valve 3-FCV-73-19 is initially driven in the closed direction, then reverses and proceeds to the full open position again. _________

[6.21] **RELEASE** the HPCI AUX OIL PUMP 3-HS-073-0047B TEST push-button. _________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) Suppression pool temperature shall be monitored every 5 minutes in accordance with 3-SR-3.6.2.1.1, Suppression Chamber Water Temperature Check, when heat is being added to suppression pool.
2) Step 7.0[7] thru Step 7.0[12] is performed in preparation of the HPCI Turbine start. A pre-job brief should be considered at this time.

[7] **CALCULATE** the Minimum HPCI Main Pump Discharge Pressure as follows:

[7.1] **RECORD** pretest suction pressure and thrust bearing temperature below:

Parameter/Indicator	Indicated Value	Acceptable Range
HPCI PUMP SUCT PRESS 3-PI-073-0028B (3-25-50)	psig	≥ 10 psig
HPCI TURB THR BRG TEMP 3-TE-73-54F (ICS) OR 3-TE-73-54F (3-9-47)	°F	N/A

____

[7.2] **RECORD** reactor pressure indicated by REACTOR WIDE RANGE PRESS A 3-PI-3-54 on Panel 3-9-5 below and in Step 7.0[7.3]:

Indicated Value	Acceptance Criteria
psig	≥ 950 psig ≤ 1040 psig

____

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

The basis for determining the minimum, HPCI main pump discharge pressure is derived from startup testing performed by STI-15. Specifically, discharge pressure was measured at 100 psig above reactor pressure for successful injection at rated flow. A 10 psig margin has been added to this measured value based on engineering judgment to arrive at 110 psig value utilized by this SR.

[7.3] **CALCULATE** minimum HPCI main pump discharge pressure required as indicated below:

Reactor Pressure = ____________________ psig
(Step 7.0[7.2])

\+ ________110________ psig
(See Note)

Min Disch Press = ____________________ psig ________

Date ___________

7.0 PROCEDURE STEPS (continued)

[7.4] **RECORD** minimum HPCI main pump discharge pressure calculated in Step 7.0[7.3] in the following steps: ________

Step 7.0[13.1]
Step 7.0[14.2]
Step 7.0[17.1]

Attachment 3 (if required)
Step 1.0[17]
Step 1.0[19.2]

[7.5] **VERIFY** calculation performed in Step 7.0[7.3] is correct AND pressure value obtained has been correctly recorded in steps specified by Step 7.0[7.4]. ________
IV

NOTE

Starting the HPCI turbine with HWC in service and flow is **NOT** at a reduced rate may result in a higher than Normal Radiation Levels.

[8] **VERIFY** HWC Flow is at the Desired Setpoint or removed from service as required by Radcon. ________

[9] **PERFORM** the following

- **VERIFY** the M&TE equipment is available and ready to support HPCI operation. ________
- **VERIFY 3-**SR-3.6.2.1.1, Suppression Chamber Water Temperature Check has been commenced. ________
- **VERIFY** RHR is in Suppression Pool Cooling per 3-OI-74 as determined by the Unit Supervisor. ________

[10] **START** Standby Gas Treatment System (SGTS) in accordance with 0-OI-65, Standby Gas Treatment System. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[11] **ALIGN** HPCI System for a manual start by performing the following steps:

[11.1] **CHECK** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 is in AUTO and **SET** to control at 500 (5,000 gpm).

IF required, **THEN**

DEPRESS AUTO operation mode transfer switch and **ADJUST** setpoint using Setpoint up/down keys. ________

[11.2] **PLACE** HPCI STEAM PACKING EXHAUSTER by placing 3-HS-73-10A to START. ________

[11.3] **VERIFY** 3-FCV-73-36, using HPCI/RCIC CST TEST VLV 3-HS-73-36A, is OPEN. ________

[11.4] **OPEN** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A. ________

WARNING

[NER] Failure of both HPCI steam exhaust piping rupture discs during turbine startup and operation will result in a process steam release into HPCI Room. This release raises the risk of personnel injury until steam line isolation occurs. Therefore, personnel in HPCI Room should minimize stay time in close proximity to rupture disc cage assembly. [IE 93-67]

Start of Critical Step(s)

[12] **START** the HPCI turbine by performing the following:

[12.1] [NER] **VERIFY** communication is established with Operations personnel in HPCI Room. [IE 93-67] ________

[12.2] [NER] **REQUEST** Operations personnel in HPCI Room, to ensure that all unnecessary personnel have exited HPCI Room. [IE 93-67] ________

[12.3] [NER] **ANNOUNCE** HPCI turbine startup over plant public address system. [IE 93-67] ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[12.4] **PLACE** HPCI AUXILIARY OIL PUMP 3-HS-73-47A to START. ________

[12.5] **OPEN** 3-FCV-73-30, using HPCI PUMP MIN FLOW VALVE, 3-HS-73-30A. ________

NOTES

1) Personnel Monitoring the 3-FCV-73-18 valve for smooth operation must pay close attention to valve travel from the time 3-FCV-73-16 is opened until 3-FCV-73-18 is full open and stable.

2) Smooth operation for 3-FCV-73-18 is a continuous operation from full close to full open without erratic movement. Sound can be used to assist in determining operation of valve. (i.e., The Valve slams open suddenly and then closed and then ramps open is **NOT** smooth operation.)

[12.6] **ENSURE** personnel are ready to monitor 3-FCV-73-18 for smooth operation.

AND

NOTIFY the personnel monitoring that the next step will open 3-FCV-73-18. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) During the startup of the HPCI Turbine a second operator should be utilized to monitor for abnormal conditions and alarms.
2) The HPCI Turbine parameters should be monitored during HPCI startup. This will ensure proper response of the control systems. If HPCI pumps suction pressure causes an auto swap of the HPCI suction valves from CST to the torus, then the HPCI Turbine should be tripped.
3) **REVIEW** Step 7.0[12.8] to ensure actions occur when 3-FCV-73-16 opens.

CAUTIONS

1) If HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 fails to fully open, then the governor control system ramp generator will time out and HPCI turbine speed, discharge pressure, or flow will be lower than expected.

   DO NOT RE-ATTEMPT to open HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 unless HPCI TURBINE STOP VALVE 3-FCV-73-18 is closed using HPCI TURBINE TRIP 3-HS-73-18A. Failure to observe this caution will result in a turbine overspeed trip if 3-FCV-73-16 is opened with the ramp generator timed out.
2) During the startup of the HPCI Turbine, the flow indication will remain high during the transient until the Governor Control System stabilizes the HPCI Flow to the desired setpoint.
   - The response time of the Governor Control System is slow. Therefore flow should **NOT** be adjusted until the system has stabilized. During this time the operator should monitor the speed indication for proper operation of the Governor Control.
   - The Ramp Generator will cause the Turbine Speed to rise at a steady rate until the Signal Converter circuit takes control and lowers the speed to stabilize the flow at the desired setpoint.
3) Starting the HPCI turbine with HWC in service and without the flow being at a reduced rate may result in higher than Normal Radiation Levels.

[12.7] **OPEN** 3-FCV-73-16, using HPCI TURBINE STEAM SUPPLY VLV, 3-HS-73-16A. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[12.8] **OBSERVE** that the following actions occurs:

- HPCI AUXILIARY OIL PUMP starts. ________
- [NRC/C] HPCI TURBINE STOP VALVE 3-FCV-73-18 opens by observing 3-ZI-73-18 position indicating lights. [Appendix R] ________
- HPCI TURBINE CONTROL VALVE 3-FCV-73-19 partially or fully opens by observing 3-ZI-73-19 position indicating lights. ________
- [NRC/C] HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 closes when HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 indicates approximately ≥ 125 (≥ 1250 gpm) flow. [Appendix R] ________
- HPCI turbine speed rises to greater than 2400 rpm as indicated on HPCI TURBINE SPEED 3-SI-73-51. ________
- HPCI STM LINE CNDS INBD/OUTBD DR VLVS 3-FCV-73-6A and 3-FCV-73-6B close by observing 3-ZI-73-6A and 3-ZI-73-6B position indicating lights. ________
- HPCI AUXILIARY OIL PUMP stops as turbine speed rises. ________

[12.9] **VERIFY** Smooth operation of 3-FCV-73-18 and mark results below.

Yes ☐ No ☐ ________

- **IF** the Answer above is “NO”, **THEN**

  NOTIFY System Engineer to initiate a WO and proceed with test. (Otherwise **N/A**) ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[12.10] **VERIFY RESET** the following annunciators:

- HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5)
- HPCI TURBINE TRIPPED 3-ZA-73-18 (3-XA-55-3F, window 11)
- HPCI TURBINE GLAND SEAL DRAIN PRESSURE HIGH 3-PA-73-46 (3-XA-55-3F, window 14)
- HPCI TURBINE BEARING OIL PRESSURE LOW 3-PA-73-47 (3-XA-55-3F, window 19) ________

[12.11] **VERIFY** system flow, discharge pressure, and turbine speed are stable prior to performing the next step. ________

End of Critical Step(s)

Date ___________

7.0 PROCEDURE STEPS (continued)

NOTES
1) **PAUSE** periodically as HPCI discharge pressure approaches the desired test pressure to allow HPCI system flow, discharge pressure, and turbine speed to stabilize. BFPER 00-003572-000
2) Due to discharge pressure indicator failures, speed should be monitored **NOT** to exceed 4200 rpm to minimize exceeding design discharge pressure.

[13] **WHILE** maintaining HPCI Turbine Speed less than 4200 rpm, **ADJUST** HPCI Pump Discharge Pressure as follows:

[13.1] [NRC/C] **SLOWLY THROTTLE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A, as necessary, until the following are achieved:

- HPCI PUMP DISCH PRESS as indicated on 3-PI-73-31A is

  ≥ ______________ psig
  (Step 7.0[7.3])

- Discharge flow steadies at or above 500 (5,000 gpm) as indicated by HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33. [Appendix R] ________

[13.2] [NRC/C] **CHECK** HPCI Room for evidence of steam, oil, and gland seal condenser leaks. ________

[13.3] **REQUEST** RADCON to monitor radiation and contamination levels to ensure either has **NOT** risen significantly. [RPT-82-13] ________

[13.4] **VERIFY** system flow, discharge pressure, and turbine speed are stable prior to performing the next step. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE
Steps 7.0[14.1], 7.0[14.2] and 7.0[14.3] may be performed concurrently.

CAUTION
HPCI main/booster pump bearing temperatures shall **NOT** be allowed to exceed 155°F.

[14] **MONITOR** and **OBTAIN** the following data:

[14.1] **MONITOR** the following HPCI turbine and pump set temperatures using HPCI/RCIC/RFW TEMPERATURES 3-TR-73-54 on Panel 3-9-47 or ICS to verify temperatures are **NOT** rising rapidly.

AND

CHECK that no temperature exceeds 155°F:

PARAMETER INST CHANNEL
HPCI OIL COOLER DISCH 3-TE-73-54A
HPCI TURB HP BRG OIL 3-TE-73-54D (Gov End)
HPCI TURB LP BRG OIL 3-TE-73-54E (Cplg End)
HPCI TURB THRUST BRG 3-TE-73-54F
HPCI PUMP INBOARD BRG 3-TE-73-54G
HPCI PUMP OUTBOARD BRG 3-TE-73-54H
HPCI SPEED INCREASER 3-TE-73-54J

____

Date ____________

7.0 PROCEDURE STEPS (continued)

[14.2] **RECORD** following data:

Parameter/Indicator	Indicated Value	Acceptable Range
HPCI SYSTEM FLOW 3-FIC-73-33 or ICS	gpm	≥ 5,000 gpm
HPCI PUMP DISCH PRESS 3-PI-73-31A	psig	≥ ________ (Step 7.0[7.3])
HPCI TURBINE SPEED 3-SI-73-51	rpm	≥ 2,400 rpm
HPCI TURB EXH PRESS 3-PI-73-21A	psig	≤ 40 psig
REACTOR WIDE RANGE PRESS A 3-PI-3-54	psig	≥ 950 psig ≤ 1040 psig

______(AC)

[14.3] **RECORD** following data:

Parameter/Indicator	Indicated Value	Acceptable Range
HPCI PUMP SUCT Press 3-PI-73-28A	psig	≥ 10 psig

____

Date ____________

7.0 PROCEDURE STEPS (continued)

[15] **OBTAIN** ASME OM Code data for HPCI main and booster pump set as follows:

[15.1] **PLACE** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 in MANUAL as follows:

DEPRESS the MANUAL operation mode transfer switch on 3-FIC-73-33. ________

[15.2] **ADJUST** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 MANUAL operation lever, until approximately 3800 rpm on HPCI TURBINE SPEED 3-SI-73-51. ________

[15.3] **ADJUST** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 MANUAL operation lever, to achieve 3790 to 3810 rpm turbine speed, using hand held tachometer.

AND

RECORD final turbine speed below:

HPCI Turbine Speed (M&TE) ___________ rpm ________

[15.4] **VERIFY**HPCI test condition flow rate as follows:

[15.4.1] **IF** ICS is utilized to obtain HPCI flow rate data, **THEN**

CHECK that no gross instrument channel failures have occurred by noting that ICS-displayed HPCI flow rate is within 100 gpm of flow rate indicated on HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33. (Otherwise **N/A**) ________

[15.4.2] **THROTTLE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A to obtain either of the following:

- An ICS display reading of 4950 to 5050 gpm.

  OR

- 495 to 505 (4950 to 5050 gpm) as indicated on HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[15.5] **ALLOW** HPCI pump set to operate until steady-state conditions are achieved, **THEN**

VERIFY conditions of Steps 7.0[15.3] and 7.0[15.4] are met. ________

> **NOTE**
>
> The indicator on 3-PI-73-31B may oscillate due to pump generated pressure pulses. Should this condition exist, an average between the predominate high and low readings should be recorded as the indicated value.

[15.6] **OBTAIN** the HPCI pump data as follows:

[15.6.1] On Panel 3-LPNL-25-0050

PERFORM the following:

A. **OBSERVE** 3-PI-73-31B, while performing the following to verify unobstructed instrumentation.

CLOSE and **OPEN** PANEL ISOL VLV TO 3-PI-73-31B, 3-PISV-73-9013 several times.

B. **IF** required to stabilize 3-PI-73-31B indicator, **THEN**

THROTTLE PANEL ISOL VLV TO 3-PI-73-31B, 3-PISV-73-9013, as required to stabilize 3-PI-73-31B. (Otherwise **N/A**) ________

Date ____________

7.0 PROCEDURE STEPS (continued)

CAUTIONS

1) The HPCI pump set differential pressure is very sensitive to minor turbine speed and pump set flow adjustments. Therefore, it is anticipated that the UO will be required to make minor speed and flow rate adjustments in order to properly establish the ASME OM Code operating point.
2) HPCI pump discharge pressure has no required range because it is a function of pump speed and flow.

[15.6.2] **COMPLETE** following table entries stipulated below:

Parameter/Indicator	Indicated Value	Required Value
HPCI SYSTEM FLOW 3-FIC-73-33 or ICS	gpm	4,950-5,050 gpm
HPCI MAIN PUMP DISCH PRESS 3-PI-073-0031B (HPCI RM)	psig	SEE CAUTION ABOVE
HPCI TURBINE SPEED HAND-HELD TACHOMETER	rpm	3790-3810 rpm
HPCI TURB EXH PRESS 3-PI-73-21A	psig	≤ 40 psig
REACTOR WIDE RANGE PRESS A 3-PI-3-54	psig	≥ 950 psig ≤ 1040 psig
HPCI PUMP SUCTION PRESS 3-PI-073-0028B (HPCI RM)	psig	≥ 10 psig

______(AC)

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

Calculation independent verification (IV) consists of verifying arithmetic for accuracy and arithmetic inputs have been properly transferred between the steps within the surveillance. IV is **NOT** required to verify pressure data recorded at local instrument rack is correct.

[15.7] **CALCULATE** HPCI pump set differential pressure as follows:

[15.7.1] Using applicable data recorded in Step 7.0[15.6.2], **CALCULATE** HPCI pump set differential pressure:

Discharge Pressure (3-PI-73-31B)		_________	psig
Suction Pressure (3-PI-73-28B)	-	_________	psig
Differential Pressure	=	_________	psid

____

[15.7.2] **VERIFY** that the differential pressure calculated is ≥ 1034 and ≤ 1201 psid. ______(AC)

[15.7.3] **INDEPENDENTLY VERIFY** HPCI pump set differential pressure calculation is correct. _________ IV

[15.7.4] **IF** acceptance criteria is **NOT** met at Step 7.0[15.7.2], **THEN**

NOTIFY the Unit Supervisor that the Unit 3 HPCI pump is INOPERABLE due to low or high differential pressure (**N/A** otherwise). _________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) The HPCI lube oil system oil filter inlet pressure minus outlet pressure shall **NOT** be greater than 12 psi.
2) The target values and ranges in the table below are for information only. If the target is **NOT** met, the out of range readings should be observed periodically to ensure that the readings are **NOT** changing at a rate that could result in loss of oil pressure.
3) The data gathered in the following steps may be obtained concurrently with the vibration data at Step 7.0[15.10].

[15.8] **RECORD** the following process data values obtained locally at HPCI turbine:

Parameter/Indicator	Indicated Value	Target Value
HPCI TURB THRUST BRG & EGR SUPPLY PRESS INDR 3-PI-73-506	psig	≥ 13 psig
HPCI TURB OUTBD JOURNAL BRG SUPPLY 3-PI-73-508	psig	≥ 8 psig
HPCI TURB INBD BRG SUPPLY PRESS INDR 3-PI-73-510	psig	≥ 8 psig
HPCI MAIN PUMP BRG & SPEED REDUCER SPLY 3-PI-73-509	psig	≥ 18 psig
HPCI MAIN OIL PUMP DISCH PRESS INDR 3-PI-73-505	psig	105-110 psig
HPCI OIL FILTER INLET PRESS INDR 3-PI-73-53A	psig	SEE NOTE 1
HPCI OIL FILTER OUTLET PRESS INDR 3-PI-73-53B	psig	SEE NOTE 1
HPCI OIL SUPPLY PRESS INDICATOR 3-PI-73-501A	psig	36-40 psig

____

Date ___________

7.0 PROCEDURE STEPS (continued)

[15.9] At the HPCI turbine skid:, **CHECK** Local HPCI Oil Temperature 3-TI-073-1152 indication does **NOT** exceed 155°F.

AND

RECORD the HPCI Oil Temperature in the table below.

Parameter/Inst Channel	Indicated Value
HPCI LUBE OIL TEMP 3-TI-073-1152	°F

____

[15.10] **OBTAIN** HPCI turbine and pump set vibration levels and **RECORD** data in table below: (**N/A** if **NOT** required)

VIBS Point	Measured Value
CH	in/sec
CV	in/sec
CA	in/sec
DH	in/sec
DV	in/sec
DA	in/sec
EH	in/sec
EV	in/sec
FV	in/sec
GH	in/sec
GV	in/sec
HH	in/sec
HV	in/sec
HA	in/sec

____
EM

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) Steps 7.0[16.1] through 7.0[16.6] may be repeated to obtain accurate pressure drop readings.
2) Two personnel that will be used to install jumpers on Panel 3-9-39 may be dispatched at this time.
3) Opening 3-FCV-73-30 while HPCI pump set is operating at design flow will result in a considerable rise in HPCI Room noise as flow in minimum flow line rises. This is an expected condition which should be noted.

[16] **PERFORM** the following minimum flow function testing:

[16.1] **RECORD** below HPCI pump discharge pressure measured in HPCI Room by 3-PI-73-31B on Instrument Rack 3-25-50:

HPCI Pump Disch Press ____________ psig ________

[16.2] **NOTIFY** Operations personnel in HPCI Room to monitor HPCI pump discharge pressure measured by 3-PI-73-31B on Instrument Rack 3-25-50 when HPCI MIN FLOW VALVE 3-FCV-73-30 reaches open position. ________

[16.3] **OPEN** 3-FCV-73-30 as follows:

MOMENTARILY PLACE HPCI PUMP MIN FLOW VALVE, 3-HS-73-30A in the OPEN position. ________

[16.4] **RECORD** below the lowest HPCI pump discharge pressure measured in HPCI Room by 3-PI-73-31B on Instrument Rack 3-25-50.

HPCI Pump Disch Press ____________ psig ________

[16.5] **CHECK** HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 has re-closed after stroking full open. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

Calculation independent verification (IV) consists of verifying arithmetic for accuracy and arithmetic inputs have been properly transferred between the steps within the surveillance. IV is **NOT** required to verify pressure data recorded at local instrument rack is correct.

[16.6] **PERFORM** the following to calculate the change in HPCI pump set discharge pressure

[16.6.1] **CALCULATE** change in HPCI pump set discharge pressure as stipulated below:

Initial Discharge Pressure		____________ (Step 7.0[16.1])	psig	
Lowest Discharge Pressure	-	____________ (Step 7.0[16.4])	psig	
Discharge Pressure Change	=	____________	psig	________

[16.6.2] **INDEPENDENTLY VERIFY** pressure drop calculation performed in above is correct. ________ IV

NOTE

Verification that discharge pressure change meets the acceptance criteria stipulated in following step provides positive confirmation that HPCI PUMP MIN FLOW CHECK VALVE 3-CKV-73-559 has opened sufficiently to perform its intended design function.

[16.7] **CHECK** that discharge pressure change recorded in Step 7.0[16.6] is $\geq$ 70 psig. ______(AC)

[16.8] **ADJUST** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 MANUAL operation lever, until a turbine speed of approximately 3000 rpm is indicated by HPCI TURBINE SPEED 3-SI-73-51. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTE

Installation of the following jumper simulates the presence of a HPCI initiation signal which allows the minimum flow valve to open on low flow.

[16.9] **PLACE** jumper across 3-RLY-073-23A-K24 Contacts 11-12 in Panel 3-9-39. **REFER TO** Attachment 7.

_________ 1st

_________ 2nd

CAUTION

Throttling HPCI flow in the following step will result in the minimum flow valve opening. This will cause rapid filling of the torus. Therefore, UO should ensure that jumper is removed and minimum flow valve closed as quickly as possible to minimize torus filling.

[16.10] **THROTTLE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A until HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 indicates approximately 70 (700 gpm). _________

[16.11] [NRC/C] **CHECK** that HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 is OPEN. [Appendix R] ______(AC)

[16.12] **REMOVE** jumper placed across 3-RLY-073-23A-K24 Contacts 11-12 in Panel 3-9-39.

_________ 1st

_________ 2nd

[16.13] **THROTTLE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A until HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33, indicates between 400 and 500 (4000 and 5000 gpm). _________

[16.14] **VERIFY CLOSED** 3-FCV-73-30 using HPCI PUMP MIN FLOW VALVE, 3-HS-73-30A. _________

Date ____________

7.0 PROCEDURE STEPS (continued)

[16.15] **VERIFY** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 setpoint flow is within 50 gpm of indicated flow utilizing setpoint up/down key adjustments. ________

[16.16] **DEPRESS** AUTO operation mode transfer switch on HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33.

AND

ADJUST HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 setpoint using Setpoint up/down keys to 500 (5,000 gpm). ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) Time To Achieve Rated Flow And Pressure will be performed by Attachment 3 following the HPCI Turbine Trip if required. The following steps will adjust the 3-FCV-73-35 to the required position and will have a Caution Order placed to control the desired position.
2) Adjustments made in Step 7.0[17] should allow time for the system to stabilize prior to making further adjustments. This may require several attempts to ensure both conditions in Step 7.0[17] are met.
3) Due to discharge pressure indicator failures, speed should be monitored **NOT** to exceed 4200 rpm to minimize exceeding design discharge pressure.

[17] **IF** Time To Achieve Rated Flow And Pressure is to be performed (**REFER TO** Step 4.0[8]), **THEN**

PERFORM the following: (Otherwise **N/A**)

[17.1] **WHILE** maintaining HPCI Turbine Speed less than 4200 rpm, **THROTTLE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A, as necessary, until the following conditions are met:

- HPCI PUMP DISCH PRESS 3-PI-73-31A reads

  ≥ ______________ psig
  (Step 7.0[7.3])

  AND

- HPCI discharge flow steadies at or above 500 (5,000 gpm) as indicated by HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33. ________

Date ___________

7.0 PROCEDURE STEPS (continued)

NOTES

1) [NRC/C] **CONSULT** Step 3.0Q for additional background information regarding HPCI System removal from operable service. [NCO 89-0216-002]
2) The intent of Step 7.0[18] is to depress and hold the trip push-button for thirty seconds, verify the alarms, close 3-FCV-73-16, observe the aux. oil pump starts, then release the push-button.
3) During the HPCI Turbine trip a second operator should be utilized to monitor for abnormal conditions and alarms.
4) HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5) needs to be verified prior to 3-FCV-73-16 becoming full close.

Start of Critical Step(s)

[18] **PERFORM** the following steps to shutdown HPCI turbine:

[18.1] **VERIFY** HPCI System has been declared inoperable and **ENTER** appropriate LCO information into Narrative log as required. _________ US

[18.2] **DEPRESS** and **HOLD** HPCI TURBINE TRIP 3-HS-73-18A until Step 7.0[18.8] is performed. _________

[18.3] **WAIT** 30 seconds and **OBSERVE** following annunciators are in ALARM:

- [NRC/C] HPCI TURBINE TRIPPED 3-ZA-73-18 (3-XA-55-3F, window 11). [Appendix R] _________
- HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5). _________

[18.4] **CLOSE** 3-FCV-73-16, using HPCI TURBINE STEAM SUPPLY VLV, 3-HS-73-16A. _________

[18.5] **OBSERVE** HPCI AUXILIARY OIL PUMP starts as turbine slows. _________

[18.6] **OBSERVE** HPCI TURBINE SPEED 3-SI-73-51, reading lowers to approximately zero. _________

Date ____________

7.0 PROCEDURE STEPS (continued)

[18.7] **VERIFY** HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 is closed. ________

[18.8] **RELEASE** HPCI TURBINE TRIP, 3-HS-73-18A. ________

[18.9] **RECORD** below and on Attachment 3 (if required) the time of HPCI Turbine shutdown:

Time ____________ ________

[18.10] On Panel 3-LPNL-25-0050, **OPEN** or **VERIFY OPEN** PANEL ISOL VLV TO 3-PI-73-31B, 3-PISV-73-9013. ________

End of Critical Step(s)

[19] **VERIFY RESET** the following annunciators:

- HPCI TURBINE TRIPPED (3-XA-55-3F, window 11) ________
- HPCI PUMP DISCH FLOW LOW (3-XA-55-3F, window 5) ________

[20] **VERIFY** HPCI STM LINE CNDS INBD/OUTBD DR VLVS 3-FCV-73-6A and 3-FCV-73-6B are OPEN by observing 3-ZI-73-6A and 3-ZI-73-6B position indicating lights. ________

NOTES

1) Step 7.0[21] should be reviewed prior to performance to ensure proper operation of system.
2) Two people are needed to perform the 3-FCV-73-18 time delay test.

[21] **PERFORM** the following at HPCI turbine:

[21.1] Using the 3-FCV-73-18 valve stem position, **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 is OPEN. ________

[21.2] **LIFT** and **IMMEDIATELY RELEASE** HPCI TURBINE MECH TRIP VLV 3-XCV-73-18 trip knob. ________

[21.3] Using the 3-FCV-73-18 valve stem position, **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 closes. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[21.4] **START** the stop watch when HPCI TURBINE STOP VALVE, 3-FCV-73-18 is full closed. ________

[21.5] **STOP** the stop watch when the HPCI TURBINE STOP VALVE 3-FCV-73-18 begins to open

AND

RECORD time delay.

Time ___________ seconds ________

[21.6] **IF** the 3-FCV-73-18 time delay in Step 7.0[21.5] is **NOT** within 4-6 seconds, **THEN**

PERFORM Attachment 4, 3-FCV-73-18 TIME DELAY ADJUSTMENT. (Otherwise **N/A**.) ________

[22] **IF** time to achieve rated flow and pressure is to be verified (**REFER TO** Step 4.0[8]), **THEN**

PERFORM the following: (Otherwise **N/A** this section)

[22.1] **PERFORM** Attachment 3. ________

[22.2] **WHEN** Attachment 3 is completed, **THEN**

CONTINUE in this procedure. ________

[23] **CLOSE** 3-FCV-73-35 using HPCI PUMP CST TEST VLV, 3-HS-73-35A. ________

[24] **CLOSE** 3-FCV-73-36, using HPCI/RCIC CST TEST VLV, 3-HS-73-36A. ________

[25] **CHECK** HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 is CLOSED. ________

[26] **CHECK** HPCI PUMP INJECTION VLV 3-FCV-73-44 is CLOSED. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[27] **VERIFY** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 is in AUTO position.

IF required, **THEN**

DEPRESS AUTO operation mode transfer switch. ________

[28] **VERIFY** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 is set to control at 500 (5,000 gpm).

IF required, **THEN**

ADJUST setpoint using Setpoint up/down keys. ________

NOTES

1) Care must be exercised to ensure that HPCI OIL TANK DRAIN 3-DRV-073-0703 is cleaned with a clean rag and solvent to remove any impurities/contaminants that could make their way into oil sample.
2) Pipe dope/sealant shall **NOT** be utilized for reinstallation of HPCI OIL TANK DRAIN 3-DRV-073-0703 pipe plug. This material is **NOT** required and serves only to contaminate oil samples.
3) Site Engineering will review and evaluate oil sample analysis as required per CI-130 to determine if a Work Order is required to correct an oil quality deficiency.

[29] **OBTAIN** an Oil Sample with the Aux Oil Pump still running to ensure thorough mixing as follows.

[29.1] **OBTAIN** two, one liter sample bottles from Chemistry Lab for obtaining HPCI lube oil sample. ________ MM

[29.2] **REMOVE** pipe plug from HPCI OIL TANK DRAIN 3-DRV-073-0703. ________ MM

[29.3] **OPEN** HPCI OIL TANK DRAIN 3-DRV-073-0703 and **REMOVE** two, one liter HPCI lube oil samples. ________ MM

Date ___________

7.0 PROCEDURE STEPS (continued)

[29.4] **LABEL** the first bottle of lube oil as **FLUSH** and **LABEL** the second bottle of lube oil as SAMPLE.

_______ MM

NOTES

1) Care must be exercised to ensure that HPCI OIL TANK DRAIN 3-DRV-073-0703 is cleaned with a clean rag and solvent to remove any impurities/contaminants that could make their way into oil sample.
2) Pipe dope/sealant shall **NOT** be utilized for reinstallation of HPCI OIL TANK DRAIN 3-DRV-073-0703 pipe plug. This material is **NOT** required and serves only to contaminate oil samples.
3) Site Engineering will review and evaluate oil sample analysis as required per CI-130 to determine if a Work Order is required to correct an oil quality deficiency.

[29.5] **CLOSE** HPCI OIL TANK DRAIN 3-DRV-073-0703 and **REINSTALL** pipe plug in end of valve housing.

_______ 1st MM

_______ 2nd MM

[29.6] **DELIVER** HPCI lube oil bottle labeled **SAMPLE** to Chemistry Lab for analysis and HPCI lube oil bottle labeled **FLUSH** for disposal and **RECORD** delivery time below:

Date __________ Time __________

_______ MM

Date ____________

7.0 PROCEDURE STEPS (continued)

CAUTION

HPCI TURBINE SPEED 3-SI-73-51 could indicate zero rpm while turbine shaft is still rotating. The auxiliary oil pump should **NOT** be stopped until visual confirmation is made by personnel that turbine speed is zero.

[30] **PERFORM** the following after allowing approximately 15 minutes to pass after turbine shutdown:

[30.1] **STOP** HPCI AUXILIARY OIL PUMP and **RETURN** 3-HS-73-47A to AUTO position. ________

[30.2] **STOP** HPCI STEAM PACKING EXHAUSTER and **RETURN** 3-HS-73-10A to AUTO position. ________

[30.3] **CHECK** HPCI TURBINE STOP VALVE 3-FCV-73-18 is CLOSED by observing 3-ZI-73-18 position indicating lights. ________

[30.4] **CHECK** HPCI TURBINE CONTROL VALVE 3-FCV-73-19 is CLOSED by observing 3-ZI-73-19 position indicating lights. ________

[31] **EXIT** HPCI System LCO by updating Narrative log. ________
US

[32] **IF** SGTS is no longer required, **THEN**

SHUT DOWN SGTS. **REFER TO** 0-OI-65, Standby Gas Treatment System. (Otherwise **N/A**) ________

Date ____________

7.0 PROCEDURE STEPS (continued)

NOTES

1) The following independent verifications are performed to ensure compliance with SPP-10.3. First party verifications have already been performed previous to this step and, therefore, have **NOT** been duplicated.
2) The independent verifications of the following step may be performed in any order.
3) If a deficiency(s) is identified during performance of the independent verifications in the following step, the independent verifier shall stop and notify the Unit Supervisor immediately for further instructions prior to correcting the deficient condition(s).
4) Successful completion of the following IVs returns HPCI System to its standby readiness configuration.

[33] **INDEPENDENTLY VERIFY** on Panel 3-9-3:

[33.1] **VERIFY** HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 is CLOSED.

________ IV

[33.2] **VERIFY** HPCI PUMP CST TEST VLV 3-FCV-73-35 is CLOSED.

________ IV

[33.3] **VERIFY** HPCI/RCIC CST TEST VLV 3-FCV-73-36 is CLOSED.

________ IV

[33.4] **VERIFY** HPCI PUMP MIN FLOW VALVE 3-FCV-73-30 is CLOSED.

________ IV

[33.5] **VERIFY** HPCI PUMP INJECTION VALVE 3-FCV-73-44 is CLOSED.

________ IV

[33.6] **VERIFY** HPCI STEAM LINE INBD DRAIN VLV 3-FCV-73-6A is OPEN by observing 3-ZI-73-6A position indicating lights.

________ IV

Date ____________

7.0 PROCEDURE STEPS (continued)

[33.7] **VERIFY** HPCI STEAM LINE OUTBD DRAIN VLV 3-FCV-73-6B is OPEN by observing 3-ZI-73-6B position indicating lights. ________ IV

[33.8] **VERIFY** HPCI TURBINE STOP VALVE 3-FCV-73-18 is CLOSED by observing 3-ZI-73-18 position indicating lights. ________ IV

[33.9] **VERIFY** HPCI TURBINE CONTROL VALVE 3-FCV-73-19 is CLOSED by observing 3-ZI-73-19 position indicating lights. ________ IV

[33.10] **VERIFY** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 is in AUTO and set to control at 500 (5,000 gpm). ________ IV

[33.11] **VERIFY** HPCI STEAM PACKING EXHAUSTER 3-HS-73-10A is in AUTO. ________ IV

[33.12] **VERIFY** HPCI AUXILIARY OIL PUMP 3-HS-73-47A is in AUTO. ________ IV

[34] **INDEPENDENTLY VERIFY** in the Auxiliary Instrument Room:

[34.1] **IF** Attachment 3 was performed, **THEN**

VERIFY no jumper is installed across 3-RLY-073-23A-K47 Contacts 1-2 in Panel 3-9-39. (Otherwise **N/A**.) ________ IV

[34.2] **VERIFY** no jumper is installed across 3-RLY-073-23A-K24 Contacts 11-12 in Panel 3-9-39. ________ IV

[35] At Panel 3-LPNL-25-50 in the HPCI room, **INDEPENDENTLY VERIFY** PANEL ISOL VLV TO 3-PI-73-31B, 3-PISV-73-9013, is OPEN. ________ IV

Date ___________

7.0 PROCEDURE STEPS (continued)

NOTE
ALLOW approximately one hour to pass before checking HPCI lube oil skid reservoir level to ensure that oil in system has drained sufficiently to provide an accurate level reading.

[36] **AFTER** approximately one hour from HPCI Turbine shutdown:

PERFORM the following inspections:

- **VERIFY** HPCI lube oil skid reservoir level is per Attachment 6. ________
- **CHECK** oil level in HPCI booster pump inboard and outboard bearing oil sight glasses is per Attachment 6. ________

[37] **IF** a Yokogawa Recorder was used to measure 3-FCV-73-18, **THEN**

PERFORM the following: (Otherwise **N/A**)

- **VERIFY** Attachment 8 is completed. ________
- **ATTACH** the Chart Paper used for Timing the 3-FCV-73-18 to this procedure. ________

[38] **IF** restroking of 3-FCV-073-0018 was required during this surveillance performance, (i.e., restroke time was recorded on Attachment 10), **THEN**

PERFORM the following: (Otherwise N/A this step)

[38.1] **NOTIFY** Duty Maintenance Manager to:

- **OBTAIN** a copy of Attachment 10 for delivery to the ASME IST Program owner.

  AND

- **CONTACT** the Duty System Engineer to notify ASME IST Program owner for evaluation of test results. ________

Date ____________

7.0 PROCEDURE STEPS (continued)

[38.2] **RECORD** time below

Time: ____________ ________

[39] **COMPLETE** Attachment 1, Surveillance Procedure Review Form, up to Unit Supervisor Review. ________

[40] **NOTIFY** UO that this surveillance is complete. ________

[41] **NOTIFY** US that this surveillance is complete. ________

8.0 ILLUSTRATIONS/ATTACHMENTS

Illustration 1 - Process for Stroke Timing Valves Per ASME OM Code

Attachment 1: Surveillance Procedure Review Form

Attachment 2: HPCI Venting

Attachment 3: HPCI Cold Quick Start

Attachment 4: 3-FCV-73-18 Time Delay Adjustment

Attachment 5: ASME OM Code Inservice Testing Review Form

Attachment 6: HPCI Lube Oil Skid and Booster Pump Oil Level Settings

Attachment 7: HFA Relay Contact Layout

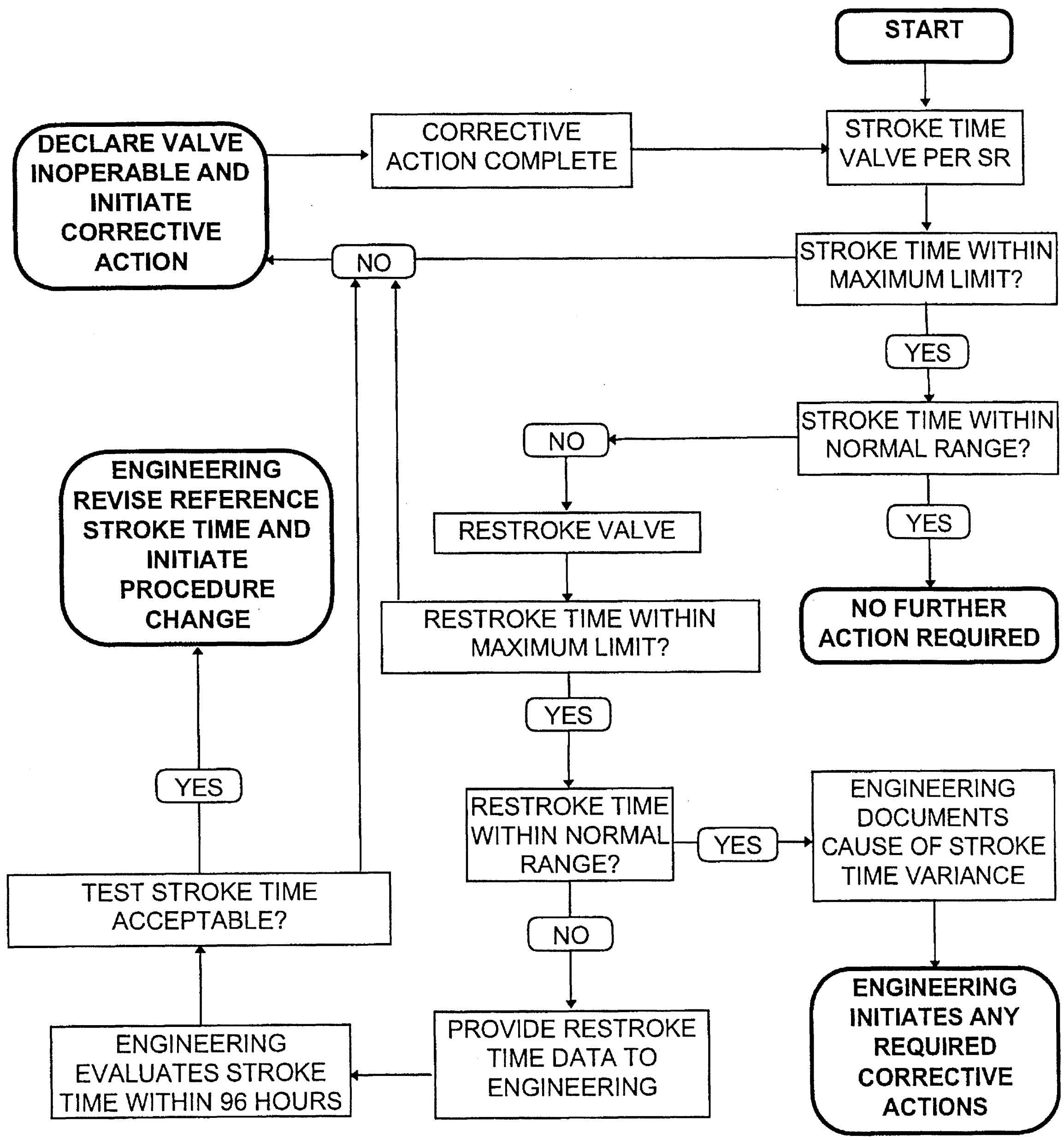
Attachment 8: Installation and Removal of Yokogawa Recorders For 3-FCV-73-18

Attachment 9 - Annunciators Affected By Surveillance Procedure Performance

Attachment 10 - ASME OM Code Restroke Time Record Form

Illustration 1
(Page 1 of 1)

Process for Stroke Timing Valves Per ASME OM Code



Attachment 1
(Page 1 of 2)

Surveillance Procedure Review Form

REASON FOR TEST:
☐ Scheduled Surveillance
☐ System Inoperable (Explain in Remarks)
☐ Maintenance (WO No. ________________)
☐ Other (Explain in Remarks)

DATE/TIME STARTED ____________
DATE/TIME COMPLETED ____________
PLANT CONDITIONS ____________

PRE-TEST REMARKS: ____

____

____

____

PERFORMED BY:

Initials	Name (Print)	Name (Signature)
	(Test Dir/Lead Perf)	
	(Test Dir/Lead Perf)	

Delays or Problems (If yes, explain in POST-TEST REMARKS)? ☐ Yes ☐ No
Acceptance Criteria Satisfied? ☐ Yes ☐ No
If the above answer is no, the Unit Supervisor shall determine if an LCO exists. LCO ☐ Yes ☐ No

UNIT SUPERVISOR ________________________________ Date __________

INDEPENDENT REVIEWER (OPS) ________________________ Date __________

SCHEDULING COORDINATOR ___________________________ Date __________

POST-TEST REMARKS: ____

____

____

____

____

____

____

____

____

Attachment 1
(Page 2 of 2)

Surveillance Procedure Review Form

Continuation Page

PERFORMED BY:

Initials	Name (Print)	Name (Signature)

POST-TEST REMARKS (Continued): ______________________________

Attachment 2
(Page 1 of 7)

HPCI Venting

Date ____________

1.0 HPCI VENTING INSTRUCTIONS

NOTES

1) The HPCI vent station is located on Elevation 565' of the Reactor Building near column lines at R16-N.
2) This attachment requires IV of valves at the vent station.
3) A digital thermometer or equivalent device may be obtained from Hot Tool Room.

WARNING

The HPCI vent line piping may contain hot feedwater. Care shall be taken when working around this potentially hot piping due to the possibility of a burn hazard existing.

[1] **VERIFY** the following valve positions from 3-PNL-9-3:

- HPCI PUMP INJECTION VALVE 3-FCV-73-44 is CLOSED. ________
- HPCI SUPPR POOL OUTBD SUCT VLV 3-FCV-73-27 is CLOSED. ________
- HPCI SUPPR POOL INBD SUCT VLV 3-FCV-73-26 is CLOSED. ________
- HPCI CST SUCTION VALVE 3-FCV-73-40 is OPEN. ________
- HPCI PUMP DISCHARGE VALVE 3-FCV-73-34 is OPEN. ________

[2] **VERIFY** CNDS SPLY TO SAFETY SYSTEMS 3-SHV-002-705 is LOCKED OPEN, locally at Elevation 541.5' of the NE quadrant of the Reactor Building. ________

Attachment 2
(Page 2 of 7)

HPCI Venting

Date ____________

1.0 HPCI VENTING INSTRUCTIONS (continued)

[3] **VERIFY** the following valve positions from 3-PNL-9-6:

- CNDS DISCHARGE NORMAL HDR VLV 3-FCV-2-167 is OPEN by noting 3-ZI-2-167 position indicating lights on Panel 3-9-6. ________
- CNDS DISCHARGE EMERGENCY HDR VLV 3-FCV-2-166 is OPEN by noting 3-ZI-2-166 position indicating lights on Panel 3-9-6. ________

[4] **THROTTLE** HPCI HIGH POINT TELL-TALE VENT SOV, 3-SHV-073-0552, approximately four turns open. ________

[5] **PLACE** thermometer probe on unpainted portion of vent line piping near HPCI HIGH POINT TELL-TALE VENT, 3-FSV-073-0062. ________

[6] **DEPRESS** and **HOLD** HPCI HIGH POINT VENT PUMP DISCH, 3-HS-073-0062, until Step 1.0[8]. ________

[7] **AFTER** 60 seconds, **THEN**

MONITOR surface temperature of vent line piping near 3-FSV-73-62 and **RECORD** temperature below.

Temperature ___________ °F ________

[8] **RELEASE** HPCI HIGH POINT VENT PUMP DISCH, 3-HS-073-0062. ________

[9] **REMOVE** thermometer probe. ________

[10] **CHECK** the surface temperature recorded in Step 1.0[7] of this attachment is less than 255°F. ________

[11] **CLOSE** HPCI HIGH POINT TELL-TALE VENT SOV, 3-SHV-073-0552. ________

[12] **OPEN** 3-FCV-73-36, using HPCI/RCIC CST TEST VLV, 3-HS-73-36A. ________

Attachment 2
(Page 3 of 7)

HPCI Venting

Date ____________

1.0 HPCI VENTING INSTRUCTIONS (continued)

[13] **OPEN** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A. ________

CAUTION

While opening HPCI PUMP INJECTION VALVE 3-FCV-73-44, HPCI discharge piping pressure must be monitored using 3-PI-73-31A on Panel 3-9-3. If discharge pressure readings equal or exceed a nominal value of 55 psig, HPCI PUMP INJECTION VALVE 3-FCV-73-44 shall be promptly closed and the Unit Supervisor contacted for additional instructions prior to proceeding with venting since this condition may indicate a gross failure of HPCI TESTABLE CHECK VLV 3-FCV-73-45.

[14] **OPEN** 3-FCV-73-44, using HPCI PUMP INJECTION VALVE, 3-HS-73-44A. ________

[15] **MONITOR** HPCI PUMP DISCH PRESS, 3-PI-73-31A on Panel 3-9-3. ________

[16] **IF** HPCI PUMP DISCH PRESS, 3-PI-73-31A, exceeds 55 psig, **THEN**

PERFORM the following: (**N/A** this section if 55 psig is **NOT** exceeded.)

[16.1] **CLOSE** the HPCI PUMP INJECTION VALVE, 3-FCV-73-44. ________

[16.2] **NOTIFY** the Unit Supervisor contacted for additional instructions prior to proceeding with venting. ________

HPCI Venting

Date ____________

1.0 HPCI VENTING INSTRUCTIONS (continued)

CAUTION

A sight glass blowout potential exists while performing the next steps. Stand clear of the flow sight glass when first depressing 3-HS-73-63. If sight glass blows out or minimal flow cannot be observed, this may indicate HPCI TESTABLE CHECK VLV 3-FCV-73-45 leakage.

[17] **OPEN** HPCI HIGH POINT TELL-TALE VENT SOV, 3-SHV-073-0551. _________

[18] **PLACE** thermometer probe on unpainted portion of vent line piping near HPCI HIGH POINT TELL-TALE VENT, 3-FSV-073-0063. _________

[19] **STATION** personnel near the HIGH POINT VENT TELL-TALE SIGHT GLASS, 3-FG-073-0513. (**REFER TO** Step 3.0GG and the caution above.) _________

[20] **VERIFY** personnel involved in the venting have reviewed and understands the indications and response that can be used by Step 3.0GG, during the performance of Step 1.0[22]. _________

Attachment 2
(Page 5 of 7)
HPCI Venting

Date ____________

1.0 HPCI VENTING INSTRUCTIONS (continued)

NOTES

1) Steps 1.0[21] thru Step 1.0[23] should be performed concurrently to ensure flow is observed in the sightglass.
2) If a steady flow of water cannot be observed from tell-tale sight flow indicator, HPCI System must be declared inoperable.

CAUTIONS

1) [NRC/C] If vent line surface temperature is > 240°F, STOP and CONTACT the Unit Supervisor for additional instructions prior to proceeding in the procedure.
2) A high surface temperature of > 240°F may indicate excessive feedwater leakage past HPCI TESTABLE CHECK VLV 3-FCV-73-45. [NRC Information Notice 89-080]

[21] **DEPRESS** and **HOLD** HPCI HIGH POINT VENT TELL TALE, 3-HS-073-0063, until Step 1.0[24] of this attachment. ________

[22] **CHECK** that HPCI System is properly vented by observing a steady flow of water in the HIGH POINT VENT TELL-TALE SIGHT GLASS, 3-FG-073-0513. ______(AC)

Attachment 2
(Page 6 of 7)
HPCI Venting

Date ________

1.0 HPCI VENTING INSTRUCTIONS (continued)

[23] **MEASURE** vent line surface temperature near HPCI HIGH POINT TELL-TALE VENT, 3-FSV-073-0063 and **RECORD** below surface temperature at the time intervals indicated:

Time	Temp (°F)	Time	Temp (°F)
1 min		6 min	
2 min		7 min	
3 min		8 min	
4 min		9 min	
5 min		10 min	

____

[24] **RELEASE** HPCI HIGH POINT VENT TELL TALE, 3-HS-073-0063. ________

[25] **REMOVE** thermometer probe. ________

[26] **CHECK** that Step 1.0[23] of this attachment, peak vent line surface temperature is less than 240°F. ________

[27] **CLOSE** HPCI HIGH POINT TELL-TALE VENT SOV, 3-SHV-073-0551. ________

[28] **CLOSE** 3-FCV-73-44, using HPCI PUMP INJECTION VALVE, 3-HS-73-44A.

____
1st

____
2nd

[29] **CLOSE** 3-FCV-73-35, using HPCI PUMP CST TEST VLV, 3-HS-73-35A. ________

Attachment 2
(Page 7 of 7)

HPCI Venting

Date ______________

1.0 HPCI VENTING INSTRUCTIONS (continued)

[30] **INDEPENDENTLY VERIFY** at Reactor Building Elevation 565':

- **VERIFY** HPCI HIGH POINT TELL-TALE VENT SOV 3-SHV-073-0552 is CLOSED. __________ IV

- **VERIFY** HPCI HIGH POINT TELL-TALE VENT SOV 3-SHV-073-0551 is CLOSED. __________ IV

Attachment 3
(Page 1 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS

CAUTIONS

1) HPCI TURBINE SPEED, 3-SI-73-51, may indicate zero rpm while turbine shaft is still rotating. The auxiliary oil pump should **NOT** be stopped until visual confirmation is made locally by personnel that turbine speed is zero.

2) **ALLOW** approximately 15 minutes to elapse following HPCI turbine shutdown before stopping HPCI PACKING EXHAUSTER to ensure removal of noncondensibles from the HPCI turbine.

[1] **RECORD** time and date of turbine shutdown from Step 7.0[18.9]:

Date ____________ Time ____________ ________

[2] **PERFORM** the following after allowing approximately 15 minutes to pass after turbine shutdown:

[2.1] **VERIFY** turbine speed is zero. ________

[2.2] **PLACE** HPCI AUXILIARY OIL PUMP, 3-HS-73-47A, to STOP and **RETURN TO** AUTO position. ________

[2.3] **STOP** HPCI STEAM PACKING EXHAUSTER by placing 3-HS-73-10A to STOP and **RETURN TO** AUTO position. ________

[3] **EXIT** HPCI System LCO by updating Narrative log. ________ US

[4] **IF** SGTS is no longer required, **THEN**

STOP SGTS in accordance with 0-OI-65, Standby Gas Treatment System. (Otherwise **N/A**) ________

Attachment 3
(Page 2 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

NOTES
1) The purpose of placing a caution tag on 3-FCV-73-35 and AOP is to alert the operator that if AOP is run after turbine shutdown period has begun, turbine shutdown period will have to begin again after AOP is stopped. This is to ensure that a non-oil primed, cold, quick start time-to-rated flow test is performed.
2) If 3-FCV-73-35 is moved from position obtained during Step 7.0[17], this surveillance may have to be reperformed. The position of 3-FCV-73-35 simulates reactor pressure during the non-oil-primed, cold, quick start time-to-rated flow test.
3) The automatic and manual functions of 3-FCV-73-35 and AOP are **NOT** affected by placement of caution tags.

[5] **CLOSE** 3-FCV-73-36, using HPCI/RCIC CST TEST VLV 3-HS-73-36A. ________

[6] **PLACE** caution tags on HPCI PUMP CST TEST VLV, 3-FCV-73-35 and HPCI AUXILIARY OIL PUMP control room and local hand-switches to restrict manual operation of these components.

<u>AND</u>

RECORD Caution Order number below:

Caution Order No: ______________________________ ________

Attachment 3
(Page 3 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

NOTES

1) The 16 hour wait period is based upon empirical data obtained by GE-San Jose and provides sufficient time for the HPCI lube oil system to completely drain back to the lube oil skid sump.

2) Empirical data obtained by GE - San Jose has demonstrated that a HPCI turbine temperature which is within 25°F of ambient will show no observable variation in its start time from a completely cold turbine and may be considered cold.

3) Ambient, HPCI Room temperature may be obtained using either an analog or digital temperature gage. HPCI TURB THRUST BRG temperature is recorded by HPCI/RCIC/FW MISC TEMPERATURE 3-TR-73-54 on Panel 3-9-47 as Point TE-73-54F.

4) Based upon temperature data from previous surveillance performances, time for HPCI turbine to reach a cold condition is approximately 36-48 hours.

[7] **VERIFY** that at a minimum of 16 hours have elapsed since time and date recorded in Step 1.0[1]. ________

[8] **CHECK** that HPCI TURB THRUST BRG temperature has returned to within 25°F of the ambient, HPCI Room temperature. ________

[9] **RECORD** below time, date and temperatures present when performance of this surveillance was resumed.

Date ___________ Time ___________

HPCI Turb Thrust Brg Temperature	**HPCI Room Ambient Temperature**
___________°F (3-TE-73-54F)	___________°F (Portable M&TE)

____

Attachment 3
(Page 4 of 10)

HPCI Cold Quick Start

Date ______

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

[10] **REMOVE** caution tags placed on HPCI PUMP CST TEST VLV 3-FCV-73-35 and HPCI AUXILIARY OIL PUMP hand-switches in Step 1.0[6]. ______

[11] **OPEN** 3-FCV-73-36 using HPCI/RCIC CST TEST VLV 3-HS-73-36A. ______

NOTES

1) Placing a jumper across Contacts 1-2 of 3-RLY-073-23A-K47 allows for immediate start of HPCI AUXILIARY OIL PUMP when 3-HS-73-47A is placed in START. This is necessary to simulate an immediate start of the HPCI AUXILIARY OIL PUMP that occurs during an actual HPCI initiation on high drywell pressure or low-low RPV water level.

2) 3-RLY-073-23A-K47 is located on Panel 3-9-39. Opening back of Panel and facing backs of relays, this relay is located on third row of relays from bottom and is third relay from right.

[12] **PLACE** jumper across 3-RLY-073-23A-K47 Contacts 1-2 in Panel 3-9-39. **REFER TO** Attachment 7.

______
1st

______
2nd

[13] **START** or **VERIFY** started SGTS in accordance with 0-OI-65, Standby Gas Treatment System. ______

[14] **START** HPCI STEAM PACKING EXHAUSTER by placing 3-HS-73-10A to START. ______

Attachment 3
(Page 5 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

NOTES

1) During the startup of the HPCI Turbine a second operator should be utilized to monitor for abnormal conditions and alarms.

2) The HPCI Turbine parameters should be monitored during HPCI startup. This will ensure proper response of the control systems. If HPCI pumps suction pressure causes an auto swap of the HPCI suction valves from CST to the torus, then the HPCI Turbine should be tripped.

WARNING

[NER] Failure of both HPCI steam exhaust piping rupture discs during turbine startup and operation will result in a process steam release into the HPCI Room. This release raises the risk of personnel injury until steam line isolation occurs. Therefore, personnel in the HPCI Room should minimize stay time in close proximity to the rupture disc cage assembly. [IE 93-67]

[15] **PERFORM** the following prior to HPCI turbine startup:

[15.1] [NER] **VERIFY** communication is established with Operations personnel in HPCI Room. [IE 93-67] ________

[15.2] [NER] **REQUEST** Operations personnel in HPCI Room ensure that all unnecessary personnel have exited HPCI Room. [IE 93-67] ________

[15.3] [NER] **ANNOUNCE** HPCI turbine startup over plant public address system. [IE 93-67] ________

Attachment 3
(Page 6 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

NOTE

Step 1.0[16] may be signed off after the completion of Step 1.0[17].

CAUTIONS

1) If HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 fails to fully open, then the governor control system ramp generator will time out and HPCI turbine speed, discharge pressure, or flow will be lower than expected.

2) **DO NOT REATTEMPT** to open HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 unless HPCI TURBINE STOP VALVE 3-FCV-73-18 is closed using HPCI TURBINE TRIP push-button 3-HS-73-18A. Failure to observe this caution will result in a turbine overspeed trip if 3-FCV-73-16 is opened with the ramp generator timed out.

Start of Critical Step(s)

[16] [NER/C] **SIMULTANEOUSLY PERFORM** the following sub-steps in order to accomplish a cold, non-oil-primed, quick start of the HPCI turbine: [INPO SOER 81-013] [GE SIL 336 R1]:

[16.1] **PLACE** HPCI AUXILIARY OIL PUMP, 3-HS-73-47A to START. ________

[16.2] **OPEN** 3-FCV-73-16, using HPCI TURBINE STEAM SUPPLY VLV, 3-HS-73-16A. ________

[16.3] **START** stopwatch. ________

End of Critical Step(s)

Attachment 3
(Page 7 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

[17] **WHEN** HPCI SYSTEM FLOW/CONTROL 3-FIC-73-33 indicates ≥ 500 (≥ 5,000 gpm) discharge flow and HPCI PUMP DISCH PRESS 3-PI-73-31A indicates a pump discharge pressure

≥ ______________ psig, **THEN**
(Step 7.0[7.3])

STOP the stopwatch. ________

NOTE
Steps 1.0[18] thru 1.0[20] should be performed in parallel with remaining surveillance steps to allow for turbine shutdown in order to limit heat addition to the suppression pool.

[18] **RECORD** below time taken to reach rated flow and pressure measured in Step 1.0[17]:

___________ seconds ________

Attachment 3
(Page 8 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

[19] **IF** ICS transient startup data is available, **THEN**

PERFORM the following. (Otherwise **N/A** the following substeps)

[19.1] **REVIEW** ICS group tabular trend display data obtained for HPCI discharge pressure, flow, and manual initiation status. ________

[19.2] **RECORD** below time span from HPCI manual initiation to when HPCI flow was 5,000 gpm with a discharge pressure:

≥ ______________ psig
(Step 7.0[7.3])

___________ seconds ________

[19.3] **CHECK** that time recorded is less than or equal to 30 seconds. _____(AC)

[20] **IF** ICS transient startup data is **NOT** available, **THEN**

CHECK that time recorded in Step 1.0[18] is less than or equal to 30 seconds. (Otherwise **N/A**) _____(AC)

Attachment 3
(Page 9 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

NOTES

1) [NRC/C] Consult Step 3.0Q for additional background information regarding HPCI System removal from operable service. [NCO 89-0216-002]
2) The intent of Steps 1.0[21] through 1.0[28] is to depress and hold the trip push-button for thirty seconds, verify the alarms, close 3-FCV-73-16, observe the aux. oil pump starts, then release the push-button.
3) During the HPCI Turbine trip a second operator should be utilized to monitor for abnormal conditions and alarms.
4) HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5) needs to be verified prior to 3-FCV-73-16 becoming full close.

[21] **VERIFY** HPCI System has been declared inoperable and **ENTER** appropriate LCO information into Narrative log as required. ________ US

Start of Critical Step(s)

[22] **DEPRESS** and **HOLD** HPCI TURBINE TRIP 3-HS-73-18A until Step 1.0[28]. ________

End of Critical Step(s)

[23] **WAIT** 30 seconds and **OBSERVE** the following annunciators are in ALARM:

- HPCI TURBINE TRIPPED 3-ZA-73-18 (3-XA-55-3F, window 11) ________
- HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5) ________

[24] **CLOSE** 3-FCV-73-16, using HPCI TURBINE STEAM SUPPLY VLV, 3-HS-73-16A. ________

Attachment 3
(Page 10 of 10)

HPCI Cold Quick Start

Date ____________

1.0 HPCI COLD QUICK START INSTRUCTIONS (continued)

[25] **OBSERVE** HPCI AUXILIARY OIL PUMP starts as turbine slows. ________

[26] **OBSERVE** HPCI TURBINE SPEED 3-SI-73-51 reading lowers to approximately zero. ________

[27] **VERIFY** HPCI TURBINE STEAM SUPPLY VLV 3-FCV-73-16 is closed. ________

[28] **RELEASE** HPCI TURBINE TRIP 3-HS-73-18A. ________

[29] **RESET** the following annunciators:

- HPCI TURBINE TRIPPED 3-ZA-73-18 (3-XA-55-3F, window 11) ________
- HPCI PUMP DISCH FLOW LOW 3-FA-73-33 (3-XA-55-3F, window 5) ________

[30] **VERIFY** HPCI STM LINE CNDS INBD/OUTBD DR VLVS 3-FCV-73-6A and 3-FCV-73-6B are OPEN by observing 3-ZI-73-6A and 3-ZI-73-6B position indicating lights. ________

[31] **REMOVE** jumper across 3-RLY-073-23A-K47 Contacts 1-2 in Panel 3-9-39. (Otherwise **N/A**)

____
1st

____
2nd

[32] **RETURN TO** Step 7.0[22.2] in the procedure. ________

Attachment 4
(Page 1 of 3)

3-FCV-73-18 Time Delay Adjustment

Date ____________

1.0 3-FCV-73-18 TIME DELAY ADJUSTMENT INSTRUCTIONS

NOTES

1) The following steps record the initial and final position of 3-SHV-73-712 to track the adjustments performed.
2) Turning the valve closed causes a slower reset time and opening the valve causes a faster reset time.
3) Step 1.0[3] may be performed multiple times to achieve a 4-6 second reset time for HPCI TURBINE STOP VALVE 3-FCV-73-18.
4) Two people are needed to perform the 3-FCV-73-18 time delay test.

[1] **DETERMINE** the as-found position of 3-SHV-73-712 as follows:

CLOSE 3-SHV-73-712 and **RECORD** the number of turns valve was opened.

As-Found ___________ Turns Open ________

[2] **RETURN** 3-SHV-73-712 to its original position as follows:

OPEN 3-SHV-73-712 the number of turns open recorded in Step 1.0[1] of this attachment. ________

Attachment 4
(Page 2 of 3)

3-FCV-73-18 Time Delay Adjustment

Date ____________

1.0 3-FCV-73-18 TIME DELAY ADJUSTMENT INSTRUCTIONS (continued)

NOTE

The following steps should be reviewed prior to performance of Step 1.0[3.3] to ensure proper operation of system.

[3] **PERFORM** the following until a 4-6 second reset time is achieved for HPCI TURBINE STOP VALVE 3-FCV-73-18.

[3.1] **ADJUST** 3-SHV-73-712 to try and achieve a 4-6 second reset time. (**REFER TO** Note above.) ________

[3.2] Using the 3-FCV-73-18 valve stem position, **CHECK** that the HPCI TURBINE STOP VALVE 3-FCV-73-18 is OPEN. ________

[3.3] **LIFT** and **IMMEDIATELY RELEASE** HPCI TURBINE MECH TRIP VLV 3-XCV-73-18 trip knob. ________

[3.4] **OBSERVE** the 3-FCV-73-18 valve stem position to CHECK that the HPCI TURBINE STOP VALVE 3-FCV-73-18 closes. ________

[3.5] **START** the stop watch when the HPCI TURBINE STOP VALVE 3-FCV-73-18 is full closed. ________

[3.6] **STOP** the stop watch when the HPCI TURBINE STOP VALVE 3-FCV-73-18 begins to open. ________

[4] **VERIFY** 3-FCV-73-18 time delay is within 4-6 seconds.

<u>OR</u>

RE-PERFORM Step 1.0[3] of this attachment. ________

[5] **RECORD** the final 3-FCV-73-18 time delay.

Time ___________ seconds ________

Attachment 4
(Page 3 of 3)

3-FCV-73-18 Time Delay Adjustment

Date ____________

1.0 3-FCV-73-18 TIME DELAY ADJUSTMENT INSTRUCTIONS (continued)

[6] **RECORD** the Final number of turns open for 3-SHV-73-712, by adding or subtracting the adjustments made in Step 1.0[3] to the initial position recorded in Step 1.0[1] of this attachment.

Number of turns open ___________ ________

[7] **RETURN TO** Step 7.0[22] in the procedure. ________

Attachment 5
(Page 1 of 1)

ASME OM Code Inservice Testing Review Form

Valves Tested	Acceptable	Not Acceptable	N/A or Not Tested
3-FCV-73-18 (Step 7.0[6.9.3] or Step 7.0[6.10.2])	☐	☐	☐
3-ISV-73-23 (Step 7.0[14.2])	☐	☐	☐
3-CKV-73-559 (Step 7.0[16.7])	☐	☐	☐
3-CKV-73-603 (Step 7.0[14.2])	☐	☐	☐

HPCI Pump	Acceptable	Not Acceptable	N/A or Not Tested
Differential Pressure (Step 7.0[15.7.2])	☐	☐	☐

Date Received: ______________________

ASME OM Code Reviewer ______________________ Date ______________

IF any evaluation results are found to be NOT Acceptable, **THEN CONTACT** OPS immediately. (Otherwise, N/A) ________ Date ________

ASME OM Code data enter in 3-SI-3.1.5 and 3-SI-3.2.1

ANII Reviewer ______________________ Date ______________

REMARKS: ____

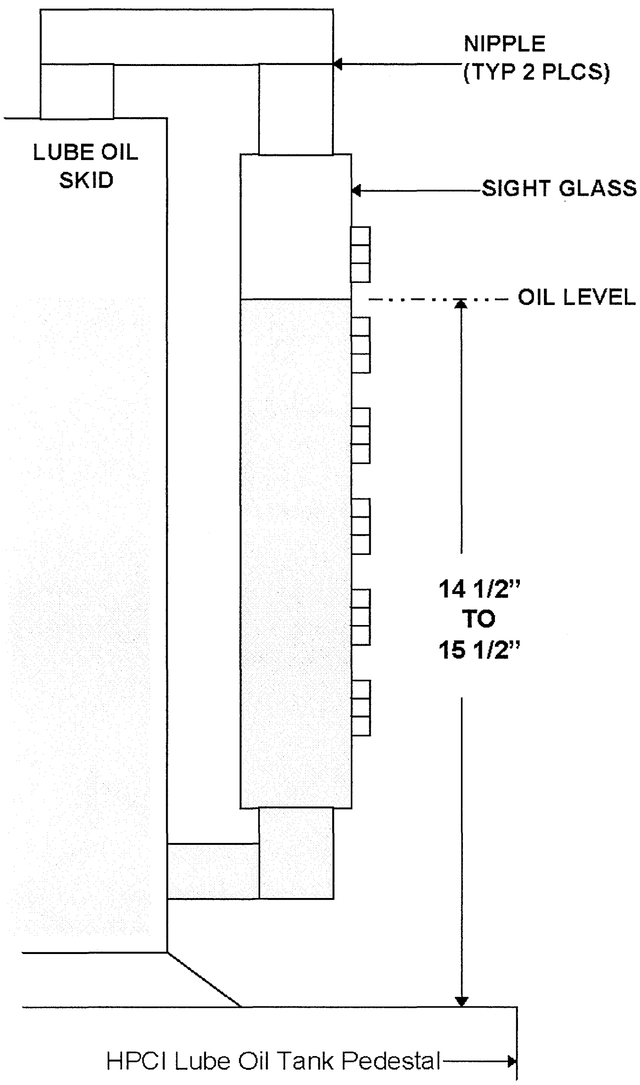
____

____

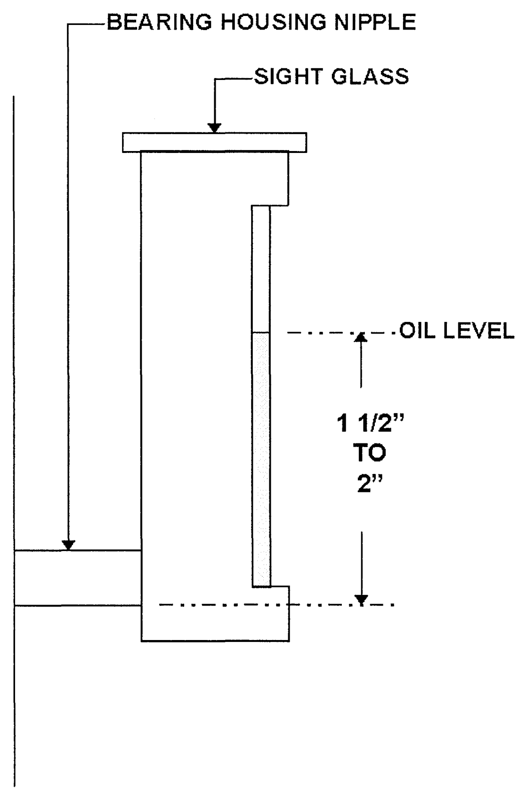
____

Attachment 6
(Page 1 of 1)

HPCI Lube Oil Skid and Booster Pump Oil Level Settings



HPCI LUBE OIL TANK
SIGHT GLASS SIDE VIEW

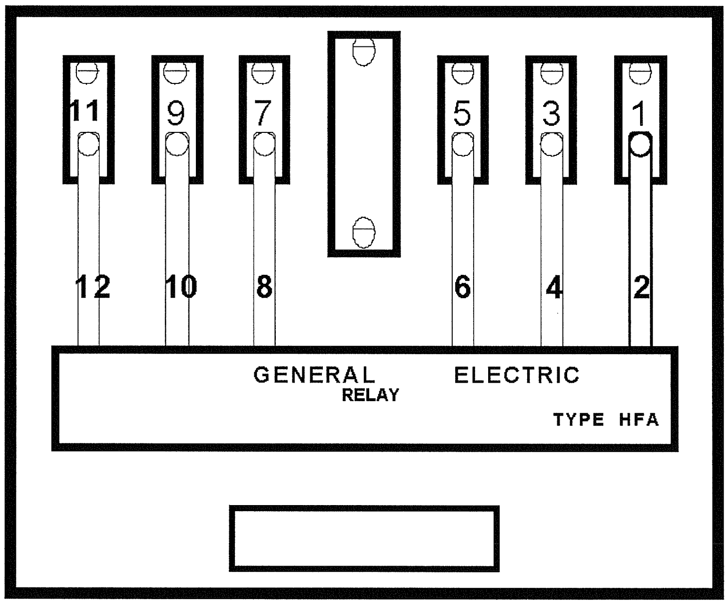


HPCI BOOSTER PUMP
SIGHT GLASS SIDE VIEW
(TYP 2 PLCS)

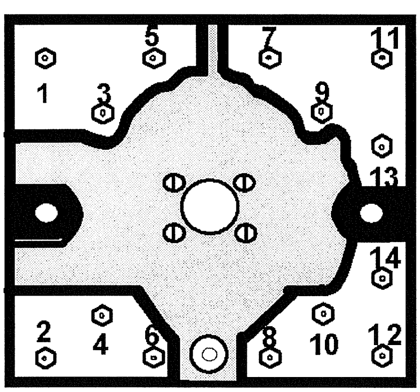
Attachment 7
(Page 1 of 1)

HFA Relay Contact Layout

FRONT



REAR



Attachment 8
(Page 1 of 4)

Installation and Removal of Yokogawa Recorders for 3-FCV-73-18

Date ____________

1.0 INSTALLATION OF THE YOKAGAWA RECORDER.

[1] **IF** the Yokogawa will be connected in Panel 3-9-3, **THEN**

CONNECT the YOKOGAWA as follows: (Otherwise **N/A**)

- For relay 3-RLY-23A-K31

  CONNECT 1 Channel across Terminals BB-90 and BB-87. ________

- For 3-ZS-73-18A

  CONNECT 1 Channel across Terminals BB-90 and BB-91. ________

- For 3-HS-73-18A

  CONNECT 1 Channel across Terminals AA-69 and AA-70. ________

Attachment 8
(Page 2 of 4)

Installation and Removal of Yokogawa Recorders for 3-FCV-73-18

Date ____________

1.0 INSTALLATION OF THE YOKAGAWA RECORDER. (continued)

[2] **IF** the Yokogawa will be connected in Panel 3-9-39, **THEN**

CONNECT the YOKOGAWA as follows: (Otherwise **N/A**)

- For relay 3-RLY-23A-K31

  CONNECT 1 Channel across Terminals BB-85 and BB-86. ________

- For 3-ZS-73-18B

  CONNECT 1 Channel across Terminals CC-25 and CC-26. ________

- For 3-HS-73-18A

  CONNECT 1 Channel across Terminals BB-11 and BB-12. ________

- For 3-PCV-73-18B

  CONNECT 1 Channel across Terminals CC-31 and CC-32. ________

Attachment 8
(Page 3 of 4)

Installation and Removal of Yokogawa Recorders for 3-FCV-73-18

Date ____________

2.0 REMOVING OF THE YOKAGAWA RECORDER.

[1] **IF** the Yokogawa was installed in Panel 3-9-3, **THEN**

REMOVE the Yokogawa Channels from the following terminals: (Otherwise **N/A**)

- Channel across Terminals BB-90 and BB-87 ________ 1st ________ 2nd
- Channel across Terminals BB-90 and BB-91 ________ 1st ________ 2nd
- Channel across Terminals AA-69 and AA-70 ________ 1st ________ 2nd

Attachment 8
(Page 4 of 4)

Installation and Removal of Yokogawa Recorders for 3-FCV-73-18

Date ____________

2.0 REMOVING OF THE YOKAGAWA RECORDER. (continued)

[2] **IF** the Yokogawa was installed in Panel 3-9-39, **THEN**

REMOVE the Yokogawa Channels from the following terminals: (Otherwise **N/A**)

- Channel across Terminals BB-85 and BB-86 ________ 1st ________ 2nd
- Channel across Terminals CC-25 and CC-26 ________ 1st ________ 2nd
- Channel across Terminals BB-11 and BB-12 ________ 1st ________ 2nd
- Channel across Terminals CC-31 and CC-32 ________ 1st ________ 2nd

Attachment 9
(Page 1 of 1)

Annunciators Affected by Surveillance Procedure Performance

Panel Location	Description	Location
3-9-3	HPCI PUMP DISCH FLOW LOW 3-FA-73-33	3-XA-55-3F Window 5
3-9-3	HPCI TURBINE TRIPPED 3-ZA-73-18	3-XA-55-3F Window 11
3-9-3	HPCI TURBINE INLET DRAIN POT LEVEL HIGH 3-LA-73-5	3-XA-55-3F Window 26

This Attachment provides the UO with a listing of Main Control Room alarms that will be affected by performance of this SR. This Attachment is for information only.

Attachment 10
(Page 1 of 1)

ASME OM Code Restroke Time Record Form

VALVE UNID	NORMAL STROKE TIME (SEC)	MEASURED INITIAL STROKE TIME (SEC)	MEASURED RE-STROKE TIME (SEC)	MAXIMUM ALLOWED STROKE TIME (SEC)
3-FCV-073-0018 (OPEN)	0.8 - 2.2			3.0

TVA

Browns Ferry Nuclear Plant

Unit 3

General Operating Instruction

3-GOI-100-12

Power Maneuvering

Revision 0031

Quality Related

Level of Use: Reference Use

Effective Date: 12-01-2007

Responsible Organization: OPS, Operations

Prepared By: William Fuller

Approved By: John Kulisek

Current Revision Description

Type of Change: Corrective Action Tracking Number: 034

PERs 961778, 126211, 116666, 132198

PCRs 07004255, 07004297

The primary reason for this revision is to help minimize unplanned radiological exposures to plant personnel during normal plant operations. Because the performance of this procedure does carry risks for such events, the following changes are made.

- The procedure is revised to identify points in the procedure requiring Radiation Protection notification to ensure any needed radiological controls are implemented to prevent unintended radiological exposure during a reactor startup. The previous revision contained steps requiring logging of Radiation Protection technician's name and also contained a signature line for the Radiation Protection supervisor, if needed. These logging and signature lines are removed from the procedure and their function replaced by the new Appendix A, added to the procedure.
- The function of Appendix A is to ensure proper communication between Operations and Radiation Protection, and that Radiation Protection is allowed sufficient opportunity to implement any needed radiological controls. A set of instructions is included with Appendix A to insure proper data entry and control of any applicable radiological protection hold points. The appendix is designed to encompass Radiation Protection notifications from this GOI and also those initiated in any support procedure implemented by this GOI.
- P&L Step 3.7, Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs), is added to provide information regarding how Radiation Protection Notifications and Radiological Protection Hold Points are to be controlled. The P&L also addresses the function of Appendix A.

The above changes are all primarily administrative in nature. Other changes to the procedure are as follows:

- Illustration 1 is changed to remove reference to specific reactor thermal limit values. The thermal limits values frequently change. The change for this revision is to reference the 0-TI-248 section by title for the limits.

Table of Contents

1.0 PURPOSE

This instruction provides precautions and limitations, prerequisites and procedural steps for power maneuvering between approximately 50% and 100% power.

The following are examples of conditions that may require use of this procedure:

- Load Following, as requested by TVA Operations Duty Specialist (ODS)
- Control rod pattern adjustment
- Control rod testing
- Removing and/or returning a Recirc pump to service
- Maintenance of plant equipment, such as Reactor Feed Pumps, Condensate or Condensate Booster Pumps, Circulating Water Pump, Condenser Waterbox, etc., that are required to support full power operations.

2.0 REFERENCES

2.1 Technical Specifications

Section 3.1, Reactivity Control Systems.

Section 3.1.3, Control Rod Operability.

Section 3.1.6, Rod Pattern Control.

Section 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR).

Section 3.2.2, Minimum Critical Power Ratio (MCPR).

Section 3.2.3, Linear Heat Generation Rate (LHGR).

Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation.

Section 3.3.2.1, Control Rod Block Instrumentation.

Section 3.3.8.2, Reactor Protection System (RPS) Electric Power Monitoring.

Section 3.4.1, Recirculation Loops Operating.

Section 3.4.2, Jet Pumps.

Section 3.4.6, RCS Specific Activity.

Section 3.7.5, Main Turbine Bypass System.

2.1 Technical Specifications (continued)

Section 5.2.2, Unit Staff.

Section 5.4, Procedures.

Section 5.5, Programs and Manuals.

2.2 Technical Requirements Manual-TRM

TRM Section 3.1, Reactivity Control.

TRM Section 3.3.1, Reactor Protection System (RPS) Instrumentation.

TRM Section 3.3.4, Control Rod Block Instrumentation.

TRM Section 3.3.5, Surveillance Instrumentation.

TRM Section 3.4.1, Coolant Chemistry.

2.3 Final Safety Analysis Report

Chapter 3.0, Reactor.

Chapter 4.0, Reactor Coolant System.

Chapter 7.0, Control And Instrumentation.

Chapter 10.0, Auxiliary Systems.

Chapter 13.0, Conduct of Operations.

2.4 Plant Instructions

3-AOI-100-1, Reactor Scram.

3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.

3-OI-2, Condensate System.

3-OI-2A, Condensate Demineralizers System.

3-OI-3, Reactor Feedwater System.

3-OI-68, Reactor Recirculation System.

3-OI-85, Control Rod Drive System.

2.4 Plant Instructions (continued)

3-OI-92B, Average Power Range Monitoring System.

3-OI-92C, Rod Block Monitoring System.

3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check.

3-SR-3.3.1.1, Core Thermal Hydraulic Stability.

3-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation.

3-SR-3.4.1(DLO), Reactor Recirculation System Dual Loop Operation.

3-SR-3.3.2.1.4(A), Rod Block Monitor (RBM) Calibration and Functional Test.

3-SR-3.3.2.1.4(B), Rod Block Monitor (RBM) Calibration and Functional Test.

OPDP-1, Conduct of Operations.

SPP-2.2, Administration of Site Technical Procedures.

SPP-10.3, Verification Program.

SPP-10.4, Reactivity Management Program.

0-TI-248, Station Reactor Engineer.

2.5 Miscellaneous Documents

BWROG-94078, BWR Owner's Group Guidelines for Stability Interim Corrective Action.

GE SIL 380, BWR Core Thermal Hydraulic Stability.

INPO SER 89-006, Withdrawal of Safety Rod Group Out of Sequence.

INPO SER 91-024, Inadequate Control of Reactivity Changes During a Plant Shutdown Results in an Unplanned Plant Transient.

INPO SER 92-008, Reactivity Management Expectations During Plant Shutdowns.

INPO SER 92-19, Power Oscillations at Boiling Water Reactors.

NRC Bulletin 88-07, Supplement 1, Power Oscillations in Boiling Water Reactors.

NRC Generic Letter 94-02, Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors.

2.5 Miscellaneous Documents (continued)

NRC Information Notice 92-74, Power Oscillation at Washington Nuclear Power Unit 2.

NRC Notice of Violation 94-24.

NSRB Item A258-4, Review procedures to preclude an event similar to SER 24-91, inadequate control of reactivity changes during plant shutdown results in unwanted transient.

Scram Frequency Reduction Committee Item SFRC-17, G-20-1 and 2.

T.A. Keys Memorandum to K.L. Welch, Use of Increased Core Flow (ICF) at Browns Ferry Nuclear Plant (L32 920709 801).

Letter from O. D. Kingsley to W. J. Museler, DOWNPOWERING OF NUCLEAR UNITS UNDER LOW SYSTEM LOAD CONDITIONS, March 1, 1996 (A00 960226 150).

TVA-BFN-TS-384, Technical Specification (TS) Change TS-384 Request for License Amendment for Power Uprate Operation (R08-980316-888).

NEDC-32751P, Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 and 3 (R08-980316-888)

GE-NE-B13-01866-39, Task Report 39 Summary of System Evaluations and Proposed Changes to Design Criteria Documents (W79-980427-005).

LETTER TVAPUR- PROC-98003, Turbine Stop Valve and Turbine Control Valve Surveillance Test Procedures (W79 980622-001).

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General

A. While performing this procedure, plant conditions/status changes such that the Unit Supervisor determines they are outside the scope of this procedure, he/she may transition to 3-GOI-100-12A or 3-GOI-100-1A, as appropriate.

3.2 Reactivity

A. [INPO/C] Activities that can directly affect core reactivity are of a critical nature and require strict procedural compliance, along with conservative actions. [INPO SER 89-006]

B. [NSRB/C] Reactivity can be added without moving control rods due to changing plant conditions (such as lowering moderator temperature, lowering Xenon concentration, rising reactor pressure, and rising feedwater flow) especially at low power. Awareness of these conditions and monitoring core instrumentation for these changes is required. [A258-4]

C. Reactor Engineering should be contacted to monitor flux shaping prior to all power reductions.

D. [QA/C] SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with the following reactivity control equipment bypassed unless bypassing of this equipment is specifically allowed within approved procedures:

1. Rod Worth Minimizer
2. Rod Block Monitor
3. Average Power Range Monitors
4. Integrated Computer System [ISE-NPS-92-R01]
5. OPRM Trip Function

E. Power Maneuvering Recommendations will be made by Reactor Engineering (REFER TO 0-TI-248 for more detailed information.)

F. Refer to 0-TI-248, Station Reactor Engineer, for Feedwater Temperature Graph and Power To Flow Map.

3.3 Technical Specifications

A. When the Reactor Recirculation System is operating in single loop operation, 3-SR-3.4.1(SLO) is required to be performed within 24 hours of entering single loop operations to ensure the requirements of Tech Specs 3.4.1 are met.

B. [NRC/C] Core Thermal-Hydraulic Stability, is required by 3-SR-3.3.1.1.I to be verified outside Regions I, II & III. when OPRM's are INOP. [NCO 940245010]

C. Whenever Forebay Temperature is >92.5°F, as indicated on 2-TS-27-144, Unit 3 power is required to be derated to within the limits shown in Illustration 2, per Tech Specs 3.7.1.2.

3.4 Condensate System Limits at Normal Steady-StateOperations

A. [II/C] Condensate flow should always be maintained within the following limits, using 3-FC-2-29 in BAL if possible, to prevent Condensate Pump damage:

1. One Condensate Pump operation, greater than 1.5 X 10^6 lbm/hr but less than 6.25 x 10^6 lbm/hr.
2. Two Condensate Pump operation, greater than 3.0 X 10^6 lbm/hr but less than 12.5 x 10^6 lbm/hr.
3. Three Condensate Pump operation, greater than 4.5 X 10^6 lbm/hr but less than 15.0 x 10^6 lbm/hr. [II-B-91-158]

B. Normal maximum line current to Condensate Pump Motors should not exceed 118 amps steady-state operations.

C. Normal maximum line current to Condensate Booster Pump Motors should not exceed 225 amps steady-state operations.

D. Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally are required to be in direct communication with the Control Room. Evolutions resulting in changes in condensate/feedwater flow (condensate/booster pump start, feedwater pump start, changes in reactor power, feedwater flow, steam flow, etc.) will affect flowrates through 3-FCV-002-0190, steam-jet air-ejector condenser(s), steam packing exhauster condenser, and off-gas condenser. 3-FI-2-42, on Panel 3-9-6 should be maintained between 2 X 10^6 ^{lbm}/hr and 3 X 10^6 ^{lbm}/hr.

3.5 Reactor Feedwater Pumps limits at Normal Steady-State Operations

A. Individual Reactor Feedpump speed should be less than 5050 RPM.

3.6 Downpowering Of Nuclear Units Under Low System Load Conditions

A. Due to having five nuclear units in an operating status, the frequency of downpowering units under low system load conditions is expected to rise. The following communications process will be used to coordinate downpowering a unit at BFN under low load conditions:

1. The Electrical System Operator (ESO) will anticipate the potential need to downpower nuclear units as far in advance as reasonable, normally one to two days. The ESO will inform the Operations Duty Specialist (ODS) of this potential need.

2. The ODS will notify the Browns Ferry Shift Manager that a potential need to downpower exists.

3. The Shift Manager will notify the Operations Superintendent who will notify the Operations Manager and Duty Plant Manager.

4. BFN will initiate a telecon with other operating nuclear units and senior nuclear corporate management (normally, Senior Vice President, Nuclear Operations, or, President, TVA Nuclear and Chief Nuclear Officer) to formulate a contingency plan. The plan will address which units are to be downpowered based on existing plant conditions, the reduction capability of each unit, time to reach reduced power as well as return to full power, and the preferred order for downpowering.

5. The contingency plan will be communicated to the appropriate site management and Shift Manager for the impacted units as well as the transmission/power supply organization.

6. The ESO will notify the designated Shift Managers approximately two to four hours before the need to actually downpower. The Shift Manager will notify the Operations Superintendent of any actual downpower.

7. Any change to unit status that would impact the agreed upon contingency plan will cause the telecon to be reconvened with all affected parties and a revised contingency plan developed. This will be initiated by the site management who identifies the need to revise the plan.

3.7 Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs) [SOER 01-1, BFN PER 126211, PER 961778, PER 116666]

A. This General Operating Instruction initiates processes that cause a change in area radiation levels in the plant. Generally, the procedure is used to reduce power to some predetermined level, and then, after the purpose of the power reduction is satisfied, the unit is returned to full power operation. The impact on radiation levels in the plant is somewhat dependent of the purpose of the power reduction, but generally plant radiation levels follow reactor power down and then rise as the unit is returned to full power operation. The performance of this procedure, in addition to the various other procedures used, carries the risk of unintended radiological exposures and also radiation levels that warrant changes in High Radiation Area or Locked High Radiation Area Controls.

Depending upon the extent of the power reduction, this GOI relies on System Operating Instructions (support procedures) for system alignments required for the various process systems. Many of these alignments can and do result in changing the radiological impacts for the areas affected by the alignments. Therefore, an increase in area monitoring may be required to determine expected dose rates for areas that might require plant personnel to be present. As the Unit is returned to full power operation, the risk of unintended radiation exposures is increased if plant personnel remain in affected areas.

B. To reduce the probability of unintended radiation exposures, the following controls are established by this procedure:

1. Radiological Protection Hold Points (RPHPs) are pre-established at appropriate locations in this GOI and in the support procedures. The function of RPHPs is to allow Radiation Protection to help ensure no unintended radiological exposures occur as the result of a system configuration change or raising reactor power. This may require holding at the point identified in the procedure until verifying personnel are not in an area before continuing in the procedure. These RPHPs also allow a determination as to whether actions are required to relax or implement RCI-17, Control of High Radiation Areas and Very High Radiation Areas, controls.

2. The Radiation Protection notification steps have an (R) placed in the step initial line, which means these steps can <u>NOT</u> be omitted unless the action associated with the step is not performed, or the Radiation Protection notification requirements are currently satisfied for the action, or the step allows the notification to be N/A'd as determined by the Unit Supervisor.

3.7 Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs) [SOER 01-1, BFN PER 126211, PER 961778, PER 116666] (continued)

3. An Appendix (Appendix A, Radiation Protection Notifications) is provided to record Radiation Protection notifications, RPHPs, and release of RPHPs, as necessary. The instructions for Appendix A are used to identify the appropriate required logging of Radiological Protection entries. The primary function of the appendix is to ensure proper communication with Radiation Protection personnel and that they are allowed sufficient opportunity to implement needed radiological controls.

4. Radiation Protection notification steps that require a RPHP are clearly worded that an RPHP is in effect. For these steps, it should be made clear to Radiation Protection that an RPHP is in effect so that they understand that a signature on Appendix A will be necessary.

   Radiation Protection notification steps that are not identified as RPHP steps are considered courtesy notification steps to Radiation Protection. These steps serve the purpose of informing Radiation Protection of evolutions that are about to be implemented that may impact plant radiological conditions and allow them to respond or “get their ducks in a row”. None of these steps imply that a hold in the procedure is necessary unless Radiation Protection identifies one may be necessary at some point after the notification is made. In many cases, the courtesy notifications are related to an RPHP notification that will be reached later in the procedure.

   These courtesy steps may also inform Radiation Protection that a system has been returned to normal, has been shutdown, or a pump that was previously started, is now shutdown. This information may be useful to Radiation Protection for determining if area surveys should be performed due to changing radiological conditions in an area. The courtesy notification steps generally require an entry of the notification in the NOMS narrative log, but may or may not require Appendix A entry by operations, depending upon expected radiological impact of the associated evolution(s).

C. Because this procedure may be implemented to recover from system operation problems and/or allow maintenance on plant equipment that may not be operating correctly, there are a multitude of scenarios that can occur while the procedure is in effect. If, at any time while performing this procedure, or while performing a support procedure, Radiation Protection personnel, or Unit Operator, Unit Supervisor, or other knowledgeable shift member identifies the need for a RPHP, then the following is performed:

3.7 Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs) [SOER 01-1, BFN PER 126211, PER 961778, PER 116666] (continued)

1. "RPHP" is written to the left of the affected procedure step number (this GOI or the support procedure). If the RPHP is identified for a support procedure, then RPHP is placed to the left of the step in this GOI that initiates the support procedure.

2. Appropriate notifications made to Radiation Protection personnel, as necessary.

3. The instructions for Appendix A are to be used to identify the appropriate required logging of Radiological Protection entries.

D. Removal of any Radiation Protection Notification from this procedure requires Operations Management and Radiation Protection Management approval unless the action(s) related to the notification is also removed.

Removal or addition of any procedure actions that require Radiation Protection notification requires that Radiation Protection be notified.

4.0 PREREQUISITES

[1] Reactor in MODE 1 with power greater than 50%.

________ Initials   ________ Date   ________ Time

Performed by: ____________________ Name (Print)   ________ Initials

Reviewed by: ____________________ Shift Manager Signature   ________ Date

5.0 INSTRUCTION STEPS

NOTES

1) [NRC/C] Sequential completion is preferred in Section 4.0[1] unless unit conditions dictate otherwise and the Unit Supervisor approves. [IR 84-45]
2) [NRC/C] Those steps preceded by an (R) are required for all power maneuvers and can not be omitted unless provided for in the procedure. [IR 84-45]
3) [NRC/C] Those steps not preceded by an (R) may be signed off as NA for all power maneuvers and initialed by the Unit Supervisor as appropriate. [IR 84-45]
4) Initials are NOT required after the step is reached where power reduction is terminated up to the step where power ascension is commenced. These steps may be marked N/A.

[1] **REVIEW** all Precautions and Limitations listed in Section 3.0.

(R) ____________ ____________ ____________
Initials Date Time

[2] **VERIFY** Prerequisite listed in Section 4.0 is satisfied.

(R) ____________ ____________ ____________
Initials Date Time

[3] **NOTIFY** Operations Duty Specialist (ODS) and/or Chattanooga Load Coordinator of impending power reduction.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

NOTE
During the power reduction, Radiation Protection should be kept informed of systems or equipment removed from service, significant power changes, and other actions or conditions that may impact radiological control areas.

[4] **NOTIFY** Radiation Protection of purpose for power reduction, the target power level (see above note), and **RECORD** time Radiation Protection notified in NOMS Narrative Log.

(R) ____________ ____________ ____________
Initials Date Time

[4.1] **VERIFY** appropriate data recorded on Appendix A in accordance with Appendix A instructions.

(R) ____________ ____________ ____________
Initials Date Time

[5] **IF** this instruction was entered due to a Recirc Pump startup or shutdown, **THEN**

PERFORM the following:

- **N/A** Step 5.0[6] through Step 5.0[11].
- **ENTER** 3-GOI-100-12 at Step 5.0[12].

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[6] **IF** power is being reduced(less than 10%) for any of the following reasons: (N/A if entering 3-GOI-100-12 to recover from a Recirc Pump Trip or power reduction of >10%)

- Weekly Control Rod Exercise
- Main Turbine Valve Testing
- Ultimate Heat Sink temperature >92.5°F

PERFORM the following:

[6.1] **REDUCE** Recirculation flow. **REFER TO** 3-OI-68.

__________ Initials __________ Date __________ Time

[6.2] **MAINTAIN** Reactor thermal power within the limits shown on Illustrations 1, 2, ICS, and 0-TI-248 as appropriate.

__________ Initials __________ Date __________ Time

[6.3] **WHEN** desired to raise power after testing is complete, **THEN**

PERFORM the following as directed by Unit Supervisor. (N/A Steps 5.0[7] through 5.0[19].

- **RAISE** Recirculation flow. **REFER TO** 3-OI-68.
- **MAINTAIN** thermal power within limits shown on Illustrations 1, 2, ICS, and 0-TI-248, Station Reactor Engineer.

__________ Initials __________ Date __________ Time

5.0 INSTRUCTION STEPS (continued)

[7] **IF** required for power maneuvering, **THEN**

PERFORM the following as directed by Reactor Engineer using 3-SR-3.1.3.5(A). (N/A if entering 3-GOI-100-12 to recover from a Recirc Pump Trip)

A. **OBTAIN** the Control Rod Movement Data Sheet.

B. **ALIGN** control rods.

(R) ____________ ____________ ____________
Initials Date Time
Reactor Engineer

NOTE
Refer to Illustration 1, ICS and/or 0-TI-248 for Reactor Thermal Limits.

[8] **REDUCE** reactor power by a combination of control rod insertions and core flow changes, as recommended by Reactor Engineer.
REFER TO 3-SR-3.1.3.5(A) and 3-OI-68. (N/A if entering 3-GOI-100-12 to recover from Recirc Pump Trip)

(R) ____________ ____________ ____________
Initials Date Time

[9] **PERFORM** the following while reducing Reactor power:
(N/A if entering 3-GOI-100-12 to recover from a Recirc Pump Trip)

[9.1] **MONITOR** Core thermal limits using Illustration 1, ICS, and/or 0-TI-248.

____________ ____________ ____________
Initials Date Time

[9.2] **MONITOR** Power reduction on Nuclear Instrumentation.

(R) ____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

CAUTION
When operating with less than the full complement of condensate pumps, condensate booster pumps, and/or reactor feedpumps, careful monitoring of motor amp limitations, feedpump speed limitations, and reactor vessel makeup capacity should be performed. This should include discussion between shift operating crews for contingency actions (e.g. tripping one of the remaining Recirc Pumps) should any remaining Condensate/Feedwater pumps trip.

NOTE
A condensate pump, condensate booster pump, and/or a reactor feedpump may be removed from service at less than 85% power to support maintenance activities as directed by the Shift Manager/Unit Supervisor.

[10] **WHEN** Reactor power is less than 85%, **THEN**

PERFORM the following: (N/A if entering 3-GOI-100-12 to recover from a Recirc Pump Trip).

[10.1] **SHUT DOWN** one of three Reactor Feedpumps, as directed by the Shift Manager or Unit Supervisor. **REFER TO** 3-OI-3. (N/A if NOT performed.)

______ Initials	______ Date	______ Time

[10.2] **REMOVE** Condensate Demineralizers as desired. **REFER TO** 3-OI-2A. (N/A if NOT performed.)

______ Initials	______ Date	______ Time

5.0 INSTRUCTION STEPS (continued)

CAUTIONS
1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally is required to be in direct communication with the Control Room.

[10.3] **MAINTAIN** flow between 2 x 10^6lbm/hr and 3 x 10^6 lbm/hr on SJAE/OG CNDR CNDS FLOW, 3-FI-2-42 using COND SPE BYPASS FLOW CONTROL, 3-HS-2-190A. **REFER TO** 3-OI-2.

Initials ______ Date ______ Time ______

[10.4] **SHUT DOWN** one of three Condensate Booster Pumps, as directed by Shift Manager or Unit Supervisor. **REFER TO** 3-OI-2. (N/A if NOT performed.)

Initials ______ Date ______ Time ______

[10.5] **SHUT DOWN** one of three Condensate Pumps, as directed by Shift Manager or Unit Supervisor. **REFER TO** 3-OI-2. (N/A if NOT performed.)

Initials ______ Date ______ Time ______

NOTE
Duration of out of service time for remaining equipment should be taken into consideration in the Step 5.0[10.6] evaluation.

[10.6] **REQUEST** Reactor Engineering to evaluate the need to lower Control Rod Line (as a contingency) should the remaining Condensate or Feedwater Pumps trip during time period pumps will be out of service.

Initials ______ Date ______ Time ______

5.0 INSTRUCTION STEPS (continued)

[11] **IF** necessary to continue power reduction to approximately 50%, **THEN**

REDUCE Reactor power by combination of control rod insertions per 3-SR-3.1.3.5(A) and core flow changes per 3-OI-68, as recommended by Reactor Engineer and directed by Unit Supervisor. (N/A if NOT performed or if entering 3-GOI-100-12 to recover from a Recirc Pump Trip)

__________ Initials __________ Date __________ Time

[12] **IF** Reactor Power is required to be lowered for Recirc Pump start-up or shut down, **THEN**

LOWER Reactor Power to desired range required by 3-OI-68. (Otherwise N/A).

__________ Initials __________ Date __________ Time

[13] **IF** continued power reduction is necessary (typically below approximately 50% power) and the Unit Supervisor determines reduction is outside scope of this procedure, **THEN**

EXIT this procedure and **PERFORM** 3-GOI-100-12A. (Otherwise N/A)

__________ Initials __________ Date __________ Time

NOTE
Illustration 1 provides Reactor thermal limits.

[14] **REVIEW** Precaution & Limitations. **REFER TO** Section 3.0.

(R) __________ Initials __________ Date __________ Time

5.0 INSTRUCTION STEPS (continued)

[15] **WHEN** desired to restore Recirc System to dual loop operation, **THEN**

PERFORM the following: (N/A if Recirc System is already in dual loop operation)

[15.1] **RESTORE** Recirc System to dual loop operation. **REFER TO** 3-OI-68, Recirc Pump Startup.

_____________ _____________ _____________
Initials Date Time

[15.2] **NOTIFY** Reactor Engineering to perform 3-SR-3.4.1(DLO) and 0-TI-248, as necessary.

_____________ _____________ _____________
Initials Date Time

[16] **BEFORE** raising reactor power, **NOTIFY** Radiation Protection that an RPHP is in effect for intentions to raise reactor power level, and **RECORD** time Radiation Protection notified in NOMS Narrative Log.

(R) _____________ _____________ _____________
Initials Date Time

[16.1] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A instructions.

(R) _____________ _____________ _____________
Initials Date Time

[17] **WHEN** desired to restore Reactor power, **THEN**

PERFORM the following:

[17.1] **RESTORE** Reactor power using control rod withdrawals in combination with core flow changes, as recommended by Reactor Engineer and directed by Unit Supervisor. **REFER TO** 3-SR-3.1.3.5(A) and 3-OI-68.

_____________ _____________ _____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in Condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally is required to be in direct communication with the Control Room.

[18] **WHEN** Reactor power is approximately 70%, **THEN**

PERFORM the following as directed by Unit Supervisor:

[18.1] **PLACE** additional condensate demineralizers in service to support starting third Condensate Pump, Condensate Booster Pump, and Reactor Feedpump. **REFER TO** 3-OI-2A. (N/A if NOT performed)

_______ Initials _______ Date _______ Time

CAUTION

When operating with less than the full complement of condensate pumps, condensate booster pumps, and/or reactor feedpumps, careful monitoring of motor amp limitations, feedpump speed limitations, and reactor vessel makeup capacity should be observed.

NOTE

A Condensate pump, Condensate booster pump, and/or a Reactor feedpump may be returned to service between 70% and 85% power following maintenance activities as directed by the Shift Manager or Unit Supervisor.

[18.2] **START** third Condensate Pump. **REFER TO** 3-OI-2. (N/A if NOT performed.)

_______ Initials _______ Date _______ Time

[18.3] **START** third Condensate Booster Pump. **REFER TO** 3-OI-2. (N/A if not performed.)

_______ Initials _______ Date _______ Time

5.0 INSTRUCTION STEPS (continued)

[18.4] **MAINTAIN** flow between 2 x 10^6 lbm/hr and 3 x 10^6lbm/hr on SJAE/OG CNDR CNDS FLOW, 3-FI-2-42 using COND SPE BYPASS FLOW CONTROL, 3-HS-2-190A. **REFER TO** 3-OI-2.

__________ Initials __________ Date __________ Time

[18.5] **PLACE** third Reactor Feedpump in service. **REFER TO** 3-OI-3. (N/A if NOT performed.)

__________ Initials __________ Date __________ Time

[19] **IF** desired to raise power with only two(2) Reactor feedpumps in service, **THEN**

RAISE Reactor power, as desired, maintaining each Reactor feedpump less than 5050 RPM. (N/A if NOT performed)

__________ Initials __________ Date __________ Time

[20] **WHEN** desired to restore Reactor power to 100%, **THEN**

PERFORM the following as directed by Unit Supervisor and recommended by the Reactor Engineer:

- **RAISE** power using control rods or core flow changes.
  REFER TO 3-SR-3.3.5(A) and 3-OI-68.
- **MONITOR** core thermal limits (Illustration 1).

__________ Initials __________ Date __________ Time

5.0 INSTRUCTION STEPS (continued)

	NAME (print)	INITIALS
Performed by:	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
	____________________	________
Reviewed by:	____________________	________
	Shift Manager	Date

Illustration 1
(Page 1 of 1)

Reactor Thermal Limits

Administrative Reactor Thermal Limits for MFLPD, MFLCPR, MAPRAT, and CTP (MWt) are listed in 0-TI-248, Appendix for Administrative Limits. These limits should be reviewed with Reactor Engineer.

Monitoring of core thermal limits at the following frequencies is recommended:

A. Following completion of planned power rises with control rods or recirc flow.

B. Following any unexpected power change.

C. Once every two hours during steady state operation.

If core monitoring software becomes unavailable, the Shift Manager and Reactor Engineer are required to determine the appropriate frequency for monitoring core thermal limits using the backup core monitoring computer taking into consideration current core conditions and margin to thermal limits. Power changes should not normally be made without the core monitoring software being available.

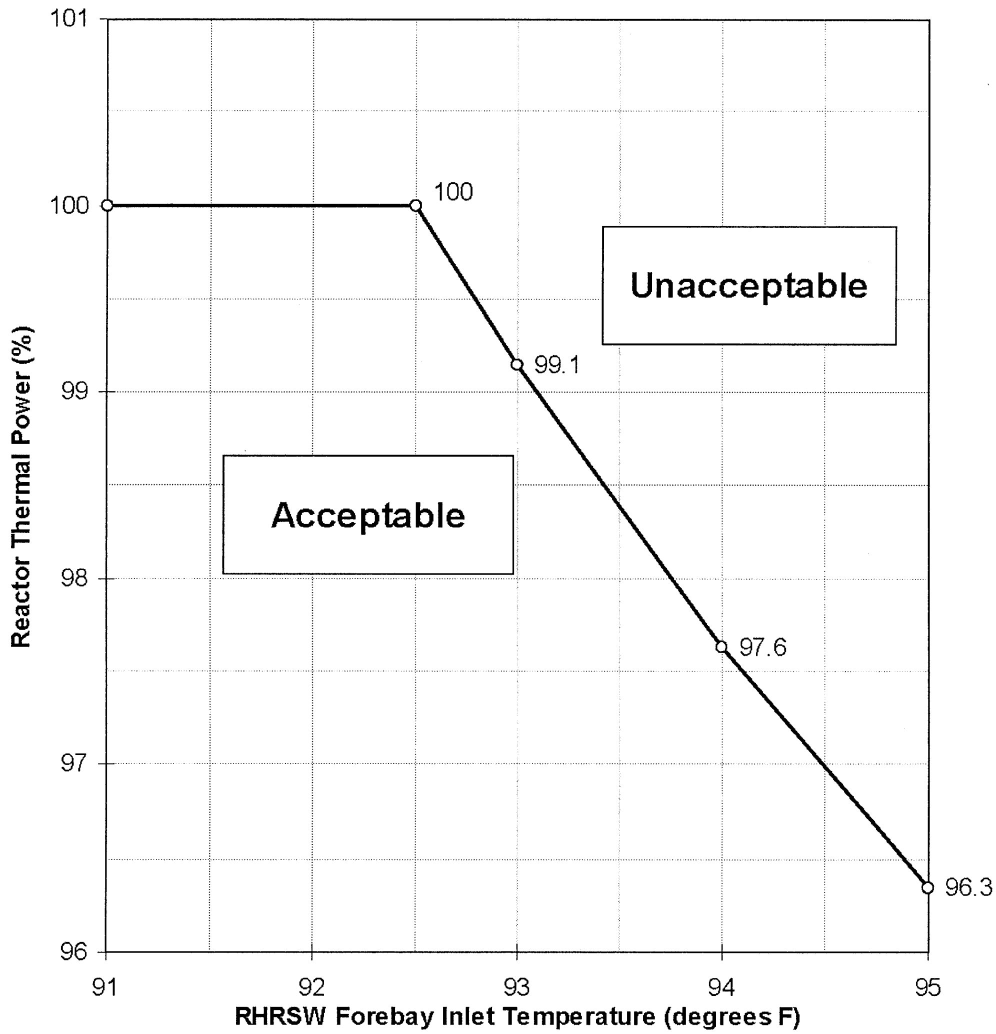
Maximum steady-state power averaged over 8 hours is 3458 MWt. However, the reactor should not be operated such that the steady state power (as indicated by 30 min avg, 1 hr avg, or 2 hr avg) is above 3458 MWt.

Minor variations in process parameter inputs to the process computer may result in individual edits or indications above 3458 MWt while true steady state core thermal power is ≤3458 MWt. Normal variation is within 5 MWt of steady-state core thermal power. Running averages (from core thermal power summary on the nuclear heat balance display) are not as sensitive. The following g+uidance is provided:

RESULT (MWt)	GUIDANCE
> 3463	REDUCE power.
3458 to 3463	ALLOW time for recent perturbations to settle. Evaluate trend. IF the trend indicates steady state core thermal power will be above 3458, THEN REDUCE power.
> 3458(any running average)	REDUCE power.

Illustration 2
(Page 1 of 1)

Reactor Thermal Power Versus Ultimate Heat Sink Temperature Limit



Appendix A
(Page 1 of 2)

Radiation Protection Notifications

INSTRUCTIONS FOR APPENDIX A DATA ENTRY

This appendix provides record of Radiation Protection notifications, RPHPs, and required signatures made during the performance of this GOI. Each notification step in this procedure, or in any referenced support procedure, that requires Appendix A be entered requires the following instructions to be used to complete the appropriate parts of the data entry page. Copies are made as needed to support this data entry.

A. Ops **ENTER** name of the Radiation Protection Representative notified with date and time of notification. Time of notification is also required in NOMS narrative log.

B. Ops **ENTER** step number (including Section number) associated with notification requirement. If the notification is directed from a support procedure, then enter the procedure number and current revision number

C. For all RPHP notifications, Radiation Protection **DETERMINE** if the RPHP is required to prevent unintended exposures and/or to implement RCI-17, Control of High Radiation Areas and Very High Radiation Area controls. **IF** RPHP is identified in a support procedure to this GOI, **THEN DETERMINE** if an RPHP is also necessary for the GOI. **CONFER** with Operations, as necessary.

D. For each identified procedure RPHP, Radiation Protection Supervisor's signature is required to release the RPHP for the action associated with affected step. This signature signifies one of two conditions: [SOER 01-1, Tech Spec 5.7, BFN PER 126211]

   1. Radiation Protection actions are completed to prevent unintended exposures and/or RCI-17 requirements have been met and any personnel working within affected areas are on an appropriate RWP for the anticipated radiological conditions.

   OR

   2. No actions were necessary because appropriate controls were already in place.

E. **WHEN** the use of this procedure is completed, **FORWARD** copies of the completed appendix pages to the Radiation Protection Supervisor.

If, while performing this procedure, or while performing a support procedure, Radiation Protection personnel, Unit Operator, Unit Supervisor, or other knowledgeable shift member identifies the need for a RPHP, then "RPHP" is written to the left of the affected procedure step number (this GOI or the support procedure. If the RPHP is identified for a support procedure, then RPHP is also placed to the left of the step in this GOI that initiates the support procedure and then A through E above is performed, as applicable.

Appendix A
(Page 2 of 2)

Name Of Radiation Protection Person Notified: ______________________________

Date: ____ / ____ / _______ Time:________________

Step#________ Procedure: ______________ (if not this procedure) Rev:______

RPHP Required by OI? ____(Y) ____(N) RPHP Required For GOI? ____(Y) ____(N)

RCI-17 Controls Necessary? ____(Y) ____(N)

Radiation Protection Supervisor Signature for Release

_______________________________ Date: ____ / ____ / _______ Time:________________

Comments:

Name Of Radiation Protection Person Notified: ______________________________

Date: ____ / ____ / _______ Time:________________

Step#________ Procedure: ______________ (if not this procedure) Rev:______

RPHP Required by OI? ____(Y) ____(N) RPHP Required For GOI? ____(Y) ____(N)

RCI-17 Controls Necessary? ____(Y) ____(N)

Radiation Protection Supervisor Signature for Release

_______________________________ Date: ____ / ____ / _______ Time:________________

Comments:

FORWARD copies of the completed appendix pages to the Radiation Protection Supervisor.

SIMULATOR EXERCISE GUIDE

TITLE	:	POWER REDUCTION, CORE SPRAY 2D PUMP INADVERTANT START, RECIRCULATION PUMP TRIP, REACTOR POWER OSCILLATIONS, ATWS WITH MSIVS OPEN
REVISION	:	0
DATE	:	January 2, 2008
PROGRAM	:	BFN Operator Training – HLT

NOTE: Provide examiners with copy of 2-OI-99 section 8.3
Also provide copy of 0-OI-57B Illustration 3

PREPARED BY:	(Operations Instructor)	\ 1/2/08 Date
REVIEWED BY:	N/A (LOR Lead Instructor or Designee)	\ Date
REVIEWED BY:	(Operations Training Manager or Designee)	\ 1/5/08 Date
	(Operations Superintendent or Designee) (Required for Exam Scenarios only)	\ Date
VALIDATION BY:	(Operations SRO) (Required for Exam Scenarios only) Date	\ 1/4/08
LOGGED-IN:	(Librarian)	\ Date
TASKS LIST UPDATED:		\ Date

NUCLEAR TRAINING REVISION/SROAGE LOG				
REVISION NUMBER	DESCRIPTION OF REVISION	DATE	PAGES AFFECTED	REVIEWED BY
0	INITIAL	04/01/07	All	

I. PROGRAM: BFN Operator Training

II. COURSE: Examination Guide

III. TITLE: POWER REDUCTION,CORE SPRAY SR FAILURE, RECIRCULATION PUMP TRIP, REACTOR POWER OSCILLATIONS, ATWS WITH MSIVS OPEN

IV. LENGTH OF LESSON: 1 to 1 ½ hours

V. Training Objectives

A. Terminal Objective

1. Perform routine shift turnover, plant assessment and routine shift operation in accordance with BFN procedures.

2. Given uncertain or degrading conditions, the operating crew will use team skills to conduct proper diagnostics and make conservative operational decisions to remove equipment/unit from operation. (SOER 94-1 and SOER 96-01)

3. Given abnormal conditions, the operating crew will place the unit in a stabilized condition per normal, abnormal, annunciator, and emergency procedures.

B. Enabling Objectives

1. The operating crew will recognize and respond to an inadvertent start of a Core Spray pump and determine required actions per Technical Specifications.

2. The operating crew will recognize and respond to a recirculation pump trip with reactor power oscillations in accordance with 3-AOI-68-1.

3. The operating crew will recognize and respond to CRD pump 3A trip per 3-AOI-85-3.

4. The operating crew will recognize and respond to an ATWS in accordance with EOI-1 and C-5.

5. The operating crew will recognize and respond to loss of 3A 480v RMOV board and determine required actions per Technical Specifications

6. The operating crew will recognize and respond to high radiation in accordance with EOI-3.

VI. References: The procedures used in the simulator are controlled copies and are used in development and performance of simulator scenarios. Scenarios are validated prior to use, and any procedure differences will be corrected using the procedure revision level present in the simulator. Any procedure differences noted during presentation will be corrected in the same manner. As such, it is expected that the references listed in this section need only contain the reference material which is not available in the simulator.

VII. Training Materials: **(If needed, otherwise disregard)**

A. Calculator

B. Control Rod Insertion Sheet

C. Stopwatch

D. Hold Order / Caution tags

E. Annunciator window covers

F. Steam tables

IX. Console Operation Instructions

A. Scenario File Summary

1. File: bat HLTS3-3

MF/RF/IOR		DESCRIPTION
1)	imf fw26b 0	'B' FW flow failure
2)	bat tohpci	Tags out HPCI
3)	imf th23 (e3 0) 2.5 15:00	Fuel failure
4)	imf rp08a	RPS A1 scram failure
5)	imf rp08b	RPS A2 scram failure
6)	trg e1 CSD	Set trigger e1 to file CSD
7)	trg e2 CSDHS	Set trigger e2 to file CSDHS
8)	trg e3 MODESW	Set trigger e3 to file MODESW
9)	trg e1= dor zdihs7542a	Trigger e1 initiates command
10)	trg e2= ior zloil7542 on	Trigger e2 initiates command
11)	imf tc02 0	Fails Bypass valves closed

2. File: bat HLTS3-3-1

MF/RF/IOR		DESCRIPTION
1)	imf th03a (none 10:00)	Trips 2A Recirc. pump
2)	imf th03b	Trips 2B Recirc. pump
3)	imf cr02a 65 10:00	Power Oscillations
4)	bat atws90	90% Hydraulic ATWS
5)	Imf tc01 (e3 5:00)	Fail bypass valves closed 5 min after mode sw

3. File: bat HLTS3-3-2

MF/RF/IOR		DESCRIPTION
1	mrf ed27 rackin	Rackin alternate feeder for 2A RMOV bd
2	mrf ed09 alt	Transfer 2A RMOV bd to alternate
3	bat NRC/HLTS10-3 (none 1:00)	Initiate file NRC/HLTS10-3

IX. Console Operation Instructions

A. Scenario File Summary

3. File: bat HLTS3-3-3

	MF/RF/IOR	DESCRIPTION
1	mrf rp01 reset	Reset RPS A
2	mrf rp09 reset	Reset RPS A Gross Failure alarm

4. File bat tohpci

	MF/RF/IOR	DESCRIPTION
1)	ior ypomtrglesh fail__cn__po	Tag gland seal exhauster
	ior ypovfcv733a close	73-3 close
2)	ior ypovfcv733 fail__now	Tag FCV 73-3
	ior ypovfcv7316 fail_now	Tag FCV 73-16
3)	ior ypovfcv7381 fail__now	Tag FCV 73-81
4)	ior zdihs7347a ptl	Tag HPCI Aux oil pump
5)	ior zohs7347a[1] off	
6)	imf hp05	HPCI trip

5. File bat app01f

	MF/RF/IOR	DESCRIPTION
1)	mrf rp13a byp	Bypasses automatic scrams
2)	mrf rp13b byp	(Appendix 1F)

6. File bat xferrmov2a

	MF/RF/IOR	DESCRIPTION
1	mrf ed27 rackin	Rackin alternate feeder for 2A RMOV bd
2	mrf ed09 alt	Transfer 2A RMOV bd to alternate

IX. Console Operation Instructions

A. Scenario File Summary

7. File bat app02

	MF/RF/IOR	DESCRIPTION
1)	mrf rp12a test	Bypasses ARI
2)	mrf rp12b test	(Appendix 2)

8. File bat app08ae

	MF/RF/IOR	DESCRIPTION
1)	mrf rp06a byp	Bypasses MSIV isolation on low
2)	mrf rp06b byp	RPV water level (Appendix 8A)
3)	mrf rp06c byp	
4)	mrf rp06d byp	
5)	mrf rp14a byp	Bypasses Rx Bldg ventilation
6)	mrf rp14b byp	isolation on low RPV level

9. File bat atws90

	MF/RF/IOR	DESCRIPTION
1)	imf rd17a	SDV level switch failure
2)	Imf rd17b	
3)	imf rd09a 90	90% hydraulic ATWS
4)	Imf rd09b 90	

10. File bat sdvtd

	MF/RF/IOR	DESCRIPTION
a)	dmf rd17a	Deletes SDV level switch failure
b)	dmf rd17b	
c)	imf rd17a (none 7:00)	Inserts level switch failure after 7 minutes
d)	imf rd17a (none 7:00)	

IX. Console Operation Instructions

B. Console Operator Manipulations

ELAP TIME	PFK	DESCRIPTION/ACTION
Sim.Setup	reset 28	100% MOC
Sim. Setup	restorepref HLTS3-3	Establishes Function Keys
Sim. Setup	setup	Verify Function Keys
Sim. Setup	esc	Clears Function Key Popup
Sim.Setup	F3	See scenario file summary (bat HLTS3-3)
Sim.Setup	manual	Tag out HPCI. Hang out of service cover on "B" FW Flow Indicator
Sim.Setup	manual	HPCI AOP and SPE pumps in PTL. Place Main Generator Voltage Regulator in Manual

ELAP TIME	PFK	DESCRIPTION/ACTION
2 minutes after Unit at 95% power	F4 ior zdihs7542a start	Core Spray pump 3D start
If lockout light does not illuminate when CS pump stopped, then	F5 Ior zloil7542 on	Illuminates lockout light
2 minutes after Tech Specs addressed for CS pump	F6 imf rd07 xx-xx	Control Rod xx-xx drifting in.
3 minutes after 3B CRD pump in service	F7 imf ed12a	Trip 3A 480v RMOV board normal feeder

ROLE PLAY: When sent to investigate board loss, Report the breaker on the 480v S/D bd feeding the normal feeder is tripped and will not re-close

ELAP TIME	PFK	DESCRIPTION/ACTION
When requested to transfer board	F8 bat xferrmov2a	Transfer RMOV board 3A to alternate
If requested to place A RPS normal	F9 bat HLTS3-3-3	Resets A RPS and gross failures
If requested to place A RPS on xfmr	F10 mrf rp04 a	Place A RPS on xfmr

↓MORE FOLLOWS↓

IX. Console Operation Instructions

B. Console Operator Manipulations

ELAP TIME	**PFK**	**DESCRIPTION/ACTION**
If requested to reset gross failures	F11 mrf rp09 reset	Resets gross failures
If requested to secure A and B SBGT	F12 bat sgt_stop	Stops SBGT trains A and B
Reset local RWCU panel	<shift>F1	mrf an01e reset
3 minutes after Tech Specs for RMOV board addressed	<shift>F2 bat HLTS3-3-1	Initiates Recirc pump trips and power oscillations. ATWS
After scram inserted	<shift>F3 bat sdv	SDV switches enabled
When appendix 2 requested, wait 3 minutes	<shift>F4 bat app02	Bypass ATWS/ARI circuit
When requested to perform appendix 1F, wait 5 minutes	<shift>F5 bat app01f	Jumper out scram logic
When requested to perform Appendix 8A and 8E, wait 5 minutes	<shift> F6 bat app08ae	Allows restart of Reactor/Refuel zone ventilation
When scram is reset	<shift> F7 bat sdvtd	SDV switches enabled
If requested to close 3-FCV-85-586, wait 5 minutes	<shift> F8 mrf rd06 close	Provides drive water pressure for rod insertion
If requested to open 3-FCV-85-586, wait 1 minute	<shift> F9 mrf rd06 open	Pressurizes charging water header
When Reactor is manually scrammed (2^{nd} time)	<shift>F3 bat sdv	SDV switches enabled
When Reactor is scram reset (2^{nd} time)	<shift> F7 bat sdvtd	SDV switches enabled
		Removes power oscillations
	<shift> F10 dmf cr02a	Deletes ATWS
	<shift> F11 bat atws-1	

Terminate the scenario when the following conditions are satisfied or upon request of the Chief Examiner:

1. All rods fully inserted
2. RPV water level +2" to +51"
3. Reporting requirements made

IX. SCENARIO SUMMARY:

The unit is operating at 100% power with a 5% power reduction scheduled. HPCI is tagged out for maintenance on the Auxiliary Oil Pump and is expected to be returned to service within the next 36 hours. It has been out of service for 14 hours.

The Main Generator voltage regulator was placed in Manual to allow PMs on the Automatic regulator. PMs are complete and the voltage regulator can be returned to Automatic.

Core Spray 3D pump inadvertent start is received and the Crew must consult Tech Specs to determine required actions.

Control Rod 18-35 drifts inadvertently into the core.

3A 480v RMOV board is lost due to breaker failure, the board will be transferred to alternate supply. RPS half scram and PCIS isolations must be reset. SRO will refer to Tech Specs.

3B Recirc pump trips, resulting in power oscillations with some fuel failure. While responding to the power oscillations per AOI-68-1, 3A Recirc pump trips and a manual scram must be inserted. The crew will experience a hydraulic ATWS and respond per 3-EOI-1. The SDV will fail to drain totally, thus requiring multiple reactor scrams to insert control rods.

X. Information to Evaluators:

A. Ensure recorders are inking and recording and ICS is active and updating.

B. Assign Crew Positions based on the required rotation.

1. SRO: Unit Supervisor ______________________
2. ATC: Board Unit Operator ______________________
3. BOP: Desk Unit Operator ______________________

C. Conduct a shift turnover with the Shift Manager and provide the Shift Manager with a copy of the Shift Turnover.

D. Direct the shift crew to review the control board and take note of present conditions, alarms, etc.

E. Terminate the scenario when the following conditions are satisfied are at the request of the floor/lead instructor/evaluator.

1. All rods inserted
2. Water level +2” to +51”
3. Reporting requirements have been made

XI. Simulator Event Guide

Event 1: POWER REDUCTION AND VOLTAGE REGULATOR TO AUTOMATIC

POSITION	TIME	EXPECTED ACTIONS
ATC	_______	Reduces power with recirculation flow
BOP	_____	Peer checks during power reduction
SRO	_______	Directs BOP to return voltage regulator to automatic per 3-OI-47 section 8.14.
BOP	_____	VERIFY VOLTAGE REGULATOR MAN/AUTO SEL, 3-HS-57-27, is in MAN.
		PLACE GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, to **RAISE UNTIL** the upper limit is reached (red light illuminated).
		PLACE GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 32-HS-57-26, to **LOWER UNTIL** the lower limit is reached (green light illuminated).
		ADJUST GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, **UNTIL** GEN TRANSFER VOLTS, 2-EI-57-41, indicates zero.
		PLACE VOLTAGE REGULATOR MAN/AUTO SEL, 3-HS-57-27, in AUTO.

XI. Simulator Event Guide

Event 2: SPURIOUS START OF 3D CORE SPRAY PUMP

POSITION	TIME	EXPECTED ACTIONS
BOP	_______	Reports start of 3D Core Spray pump and verifies no valid automatic start signal.
SRO	_______	Directs trip of 3D Core Spray pump
BOP	_____	Trips 3D Core Spray pump. Informs SRO that Lockout indicator for 3D Core Spray Pump is illuminated.
SRO	_____	Consults Tech Spec 3.5.1 determines that a 72 hour LCO is in effect with HPCI and one (1) Low Pressure ECCS system inoperable.

XI. Simulator Event Guide

Event 3: CONTROL ROD DRIFT IN

POSITION	TIME	EXPECTED ACTIONS
ATC	_______	Announces "Rod Drift" alarm
		Identifies rod 18-35 as drifting in
SRO	_______	Directs actions per 3-AOI-85-5
SRO	_______	Directs rod be continuously inserted to 00
ATC	_______	Continuously inserts rod to 00
ATC	_______	Informs Reactor Engineer
ATC	_______	Checks Thermal Limits
		Verifies CRD operating parameters within limits
ATC	_______	Directs AUO to check the following per 3-AOI-85-5: - scram pilot air header aligned - check scram outlet valve for leakage - check scram inlet valve for leakage
ATC	_______	Directs charging water to 18-35 be closed
SRO	_______	Declares accumulator inoperable per Tech Spec 3.1.5 and addresses actions (when charging water is isolated)
ATC	_______	Directs scramming of affected rod from panel 9-16 in Aux. Inst. Room
ATC	_______	Directs operator to Aux. Inst. Room for rod scram
ATC	_______	Establishes communication with operator and AUO
		Directs operator to scram rod by taking scram switch to "down" position
		Verifies rod Full In overtravel
		Direct operator to return scram switch to 'up' position
ATC	_______	Reports rod settles to 00 position
ATC	_______	Directs reopening Charging water isolation valve
ATC	_______	Resets drift and accum. Lights/alarms
SRO	_______	Initiates actions to determine CR operability and suggests actions including maintenance and inspection

XI. Simulator Event Guide

Event 4: LOSS OF 3A 480V RMOV BD

POSITION	TIME	EXPECTED ACTIONS
CREW	_____	Announces loss of 3A 480v RMOV board, RPS half scram and PCIS isolations.
		Refers to 0-OI-57B p&I 3.0 Z, Illustration 3 and 3-47E751-1 require removing 80 kva accident load to prevent overload on D D/G (3D Core Spray would be a good choice)
SRO	_____	Directs Outside US to transfer RMOV board to alternate and restore RPS A.
	_____	Directs ATC to reset half scram after RPS is restored, BOP to reset PCIS and recover from isolations per 3-OI-99 section 8.3. (attached)
	_____	Refers to ITS and determines Unit is in 8 hour LCO from 3.8.7.B.
ATC	_____	Resets A Channel half scram after RPS A is restored.
BOP	_____	Resets PCIS and restores: Reactor and Refuel zone ventilation SBGT ECCS Keep Fill system Drywell DP air compressor Drywell Floor and Equipment drain pumps Radiation Monitoring system TIP system

XI. Simulator Event Guide

Event 5: RECIRCULATION PUMP TRIP/POWER OSCILLATIONS/ATWS

POSITION	TIME	EXPECTED ACTIONS
ATC	_____	Recognizes B Recirculation pump trip
SRO	_____	Directs 3-AOI-68-1 entry
	_____	Contacts Rx Engr to place APRM's in SLO mode (If time permits)
ATC	_____	Recognizes power oscillations
	_____	Inserts rods on emergency shove sheet
BOP	_____	Verifies flow on A Recirc pump < 46,600 gpm and jet pump flow > 41,100 lbm/hr
ATC/BOP	_____	Notices failure to Scram on OPRM Trip or anticipates trip and inserts a manual scram
SRO	_____	Directs manual reactor scram
ATC	_____	Inserts manual scram Places Mode switch in shutdown Recognizes hydraulic ATWS Provides scram report
SRO	_____	Enters EOI-1/C-5 and verifies
	_____	Verifies Mode switch to shutdown
	_____	ARI initiated
	_____	Directs ADS inhibited
BOP	_____	Inhibits ADS
SRO	_____	Directs Appendix 1F and Appendix 2
	_____	Directs Appendix 1D

XI. Simulator Event Guide

Event 6: ATWS WITH FUEL FAILURE

POSITION	TIME	EXPECTED ACTIONS
BOP	_____	Recognizes 3A Recirc pump trip, if not tripped by ATC Operator due to ATWS
Crew	_____	Recognize and reports "OG Pretreatment Radiation High" "OG Annual Release Limit Exceeded" "Turbine Building High Radiation" Notifies RadCon, and Chemistry Evacuates appropriate area of Turb. Bldg.
SRO	_____	Directs RPV pressure be maintained 800-1000 psig
BOP	_____	Controls RPV pressure between 800-1000 psig with SRVs
SRO	_____	Directs water level be lowered to control power per Appendix 4 Directs Appendix 8A and 8E be performed Reports SAE 1.2-S
Crew	_____	Monitors suppression pool temperature
ATC	_____	After Appendix 1F and 2 complete resets scram and drains SDV
ATC	_____	Inserts control rods per Appendix1D
BOP	_____	Maintains water level as directed with RFP / RCIC
BOP	_____	Maintains Pressure control using the following appendices: 11D Main Steam line Drains 11F RFPT 11A SRV's

XI. Simulator Event Guide

Event 6: ATWS WITH FUEL FAILURE (continued)

POSITION	TIME	EXPECTED ACTIONS
Crew	_____	Recognizes some control rod movement, but all control rods not in
SRO	_____	Directs reactor reset, drain SDV, and re-scram
	_____	Directs SLC injection if Torus temperature approaches 110 deg.
	_____	Enters EOI-2 on Torus water level
	_____	Directs Venting per Appendix 12
	_____	Directs placing H_2O_2 monitors in service
BOP	_____	Performs Appendix 12 to vent Torus
	_____	Places H_2O_2 monitors in service
ATC	_____	Maintains water level as directed to control power
ATC	_____	After SDV drained directs 3-85-586 re-opened
ATC	_____	After accumulators recharged, scrams reactor and verifies all rods in
SRO	_____	Directs level be restored +2" to +51"
SRO	_____	Directs SLC stopped (if injected)
Crew	_____	Recognize RM-90-29A Rx Bldg High Radiation (conditional)
BOP	_____	Evacuates Reactor Building
SRO	_____	Enters EOI-3 on high Rx Bldg radiation and directs ventilation restored per Appx 8F.
ATC/BOP	_____	Closes MSIV's as directed by ARP for MSL HiHi Rad (If received)

XII. Crew Critical Tasks (CCT)

	TASK	SAT/UNSAT
1.	Manual scram due to Scram failure of OPRM Trip	______________
2.	Prevent ADS actuation	______________
3.	Controls power by : Inserting control rods per RC/Q-21 Lowering water level	______________
4.	Maintains RPV water level above -180" with rods out	______________
5.	When all rods are inserted restores and maintains RPV water level above TAF.	______________

XIII. Scenario Verification Data

EVENT	K/A Number	RO	SRO
1. Control Rod Drift	201002A2.02	3.2	3.3
2. Loss of 3A 480V RMOV Board	262001K3.06	3.8	4.1
3. APRM Failure	215005A2.03	3.6	3.7
	2.1.12	2.9	4.0
4. Recirculation Pump Trip/Power Oscillations/ATWS	295001	3.3	3.4
	295025	3.8	3.9
	295037	3.9	4.0

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER HLTS-3-3

7 Total malfunctions Inserted; List: (4-8)

1. Recirc Trip
2. Power Oscillations
3. ATWS
4. Fuel Failure
5. Core Spray pump 3D trip
6. Loss of 480v RMOV bd 3A
7. Control Rod Drift

2 Malfunctions That Occur After EOI Entry; List: (1-4)

1) ATWS
2) Fuel Failure

3 Abnormal Events; List: (1-3)

1) Control Rod Drift
2) Recirc Trip
3) 480v RMOV bd trip

2 Major Transients; List: (1-2)

1) ATWS
2) Fuel Failure

3 EOIs used; List: (1-3)

1) EOI-1
2) EOI-2
3) EOI-3

1 EOI Contingencies Used; List: (0-3)

1) C5

75 Run Time (minutes)

35 EOI Run Time (minutes); __50__ % of Scenario EOI Run Time

5 Crew Critical Tasks (2-5)

Yes Technical Specifications Exercised (yes/no)

XV. SHIFT TURNOVER INFORMATION

Equipment out of service/LCOs: HPCI tagged out for 14 hours to repair Auxiliary Oil Pump. Expected back in 3 hours. Flow indicator 3-78B out of service, IM's are looking for a new transmitter. Main Generator voltage regulator in manual for PMs on Automatic voltage regulator. Spare RBCCW pump in service to Unit 2.

Operation/Maintenance for the Shift: Reduce power to 95% with recirculation flow (due to system load not required). PMs on voltage regulator complete, return voltage regulator to automatic.

Unusual Conditions/Problem Areas: None

TVA

Browns Ferry Nuclear Plant

Unit 2

Operating Instruction

2-OI-99

Reactor Protection System

Revision 0073

Quality Related

Level of Use: Continuous Use

Effective Date: 04-02-2007

Responsible Organization: OPS, Operations

Prepared By: Terry Kenneth Boyer

Approved By: James A. McCrary

Current Revision Description

Type of Change DCN 60717-03, Editorial Tracking Number: 084

PCR: 06002112, 06003343

DCN 60717-03 removed Containment Isolation valves 2-FCV-32-0062 and 2-FCV-32-0063, associated DCA compressor suction piping, associated electrical and pneumatic controls, and PCIS trip signal circuitry in the Control Room.

Page 45 - Deleted Step 8.3[8]. 2-HS-32-62A and 2-HS-32-63A were removed by DCN 60717-03. (PCR 06002112)

Page 62 - Deleted FCV-32-62 and FCV-32-63 which were removed by DCN 60717-03. (PCR 06002112)

Page 64 - Changed the FUNCTION/SYSTEM of FCV-75-57 to PSC Pump Suction Inboard Isolation Valve. Editorial change to reflect plant conditions. (PCR 06003343)

Page 65 - Changed the FUNCTION/SYSTEM of FCV-75-58 to PSC Pump Suction Outboard Isolation Valve. Editorial change to reflect plant conditions. (PCR 06003343)

THIS REVISION DOES NOT AFFECT SYSTEM STATUS

Table of Contents

Table of Contents (continued)

ATTACHMENTS

8.6 Preventing RWCU Isolations When Transferring RPS Power Supplies In Cold Shutdown Condition (continued)

[3.5] **IF** operation of the RWCU System isolation valves is required while performing this section, **THEN**

REQUEST the operator at the breaker to PERFORM the following:

[3.5.1] **PLACE** IN ON, 3-BKR-069-0002, RWCU SYSTEM ISOL FCV-69-2 at 250V RMOV BD 3B, El 593, Compt 3D. ☐

[3.5.2] **PLACE** IN ON, 3-BKR-069-0001, RWCU SYSTEM ISOL FCV-69-1 at 480V RMOV BD 3A, El 621, Compt 16E. ☐

[3.5.3] **PLACE** IN ON, 3-BKR-069-0012, RWCU ISOLATION VALVE FCV-69-12 at 480V RMOV BD 3B, El 593, Compt 17B. ☐

[4] **TRANSFER** RPS A power supply. **REFER TO** Section 8.1 or 8.2. ☐

[5] **WHEN** RPS power supplies are transferred and RPS Restoration is completed, **THEN**

VERIFY PCIS RESET. ☐

[6] **REQUEST** the operator at the breaker to PERFORM the following:

[6.1] **PLACE** IN ON, 3-BKR-069-0002, RWCU SYSTEM ISOL FCV-69-2 at 250V RMOV BD 3B El 593, Compt 3D. ☐

[6.2] **PLACE** IN ON, 3-BKR-069-0001, RWCU SYSTEM ISOL FCV-69-1 at 480V RMOV BD 3A El 621, Compt 16E. ☐

[6.3] **PLACE** IN ON, 3-BKR-069-0012, RWCU ISOLATION VALVE FCV-69-12, at 480V RMOV BD 3B El 593, Compt 17B. ☐

CAUTIONS

1) This section shall only be used when the Reactor is in the cold shutdown condition (Mode 4 or Mode 5).
2) The amount of time in which Shutdown Cooling valves are prevented from operating by the performance of this section should be minimized. This will minimize the potential of preventing or delaying a real isolation requirement to operate the required components. The safety features of these isolation functions are important to safe operation of the plant, even with Primary Containment not required.

NOTES

1) The performance of this section will require one or two personnel qualified to operate electrical breakers with radios in direct communication with the Unit Operator in the Control Room.
2) Shift Manager/Unit Supervisor permission is required to perform this Section.

8.7 Preventing Shutdown Cooling Isolations When Transferring RPS

8.7.1 Preventing Shutdown Cooling Isolation Initial Lineup

[1] **VERIFY** the following:

- Unit 3 is in Cold Shutdown Condition (Mode 4 or Mode 5). ☐
- Reactor Mode Switch is in SHUTDOWN or REFUEL. ☐
- Tech Spec Section 3.5.2 required actions are met if applicable. ☐
- Shutdown Cooling integrity is maintained. (REFER TO Tech Spec Section 3.6.1.1 and Table 3.3.6.1-1, function 6.b.) ☐

[2] The Operators are aware that during the performance of Sections 8.7.2 (RPS 3A) or Sections 8.7.3 (RPS 3B), that the following should be performed if Containment System Isolation is required.

- **CLOSE** the associated Shutdown Cooling Isolation valve breakers.

  AND

- **VERIFY** the associated valves closed. ☐

8.7.1 Preventing Shutdown Cooling Isolation Initial Lineup (continued)

[3] **IF** Shutdown Cooling Loop I is in service and Transferring RPS A, **THEN**

PERFORM the following: (Otherwise N/A this Section)

[3.1] As Directed by the Unit Supervisor,

ALIGN RHR Loop II for Shutdown Cooling Loop. (REFER TO 3-OI-74) (N/A if RHR Loop II will not be aligned.) ☐

[3.2] **IF** Shutdown Cooling Loop I will remain in service, **THEN**

PERFORM the following: (Otherwise N/A)

A. At 480 RMOV 3D - Compartment 2C

OPEN 3-BKR-074-0053, RHR SYS I INBD INJECTION VLV FCV--74--53 (MO10-25A). ☐

B. **NOTIFY** the Unit Supervisor breaker is open and **ENTER** any applicable LCO's. ☐

[4] **IF** Shutdown Cooling Loop II is in service and Transferring RPS B, **THEN**

PERFORM the following: (Otherwise N/A this Section)

[4.1] As Directed by the Unit Supervisor,

ALIGN RHR Loop I for Shutdown Cooling Loop. . (REFER TO 3-OI-74) (N/A if RHR Loop I will not be aligned.) ☐

[4.2] **IF** Shutdown Cooling Loop II will remain in service, **THEN**

PERFORM the following: (Otherwise N/A)

A. At 480V RMOV Bd 3E - Compartment 2C

OPEN 3-BKR-074-0067, RHR SYS II INBD INJECTION VLV FCV-74-67 (MO10-25B). ☐

B. **NOTIFY** the Unit Supervisor breaker is open, and **ENTER** any applicable LCO's. ☐

8.7.2 Preventing Shutdown Cooling Isolations When Transferring RPS A Power Supplies In Cold Shutdown Condition

[1] **VERIFY** Section 8.7.1 has been performed. ☐

[2] **VERIFY** Personnel are ready to transfer RPS A Power Supply. ☐

[3] **ESTABLISH** radio communication with the Control Room Unit Operator. ☐

[4] At 480V RMOV BD 3A, El 621, Compt 8C.

A. **PLACE** 3-BKR-074-0048 RHR SHUTDOWN COOLING SUCT ISOL VLE FCV-74-48 (MO10-18), in the OFF position ☐

B. **STATION** an operator by the breaker for closure at the request of the Control Room Unit Operator. ☐

C. **IF** operation of the RHR SHUTDOWN COOLING SUCT INBD ISOL VLV is required while performing this section, **THEN**

PLACE 3-BKR-074-0048 RHR SHUTDOWN COOLING SUCT ISOL VLV FCV-74-48 (MO10-18), in ON the position. ☐

[5] **TRANSFER** RPS A power supply. (REFER TO Section 8.1 or 8.2.) ☐

[6] **WHEN** RPS A power supply has been transferred and RPS Restoration has been completed, **THEN**

CONTINUE with this section of the procedure. ☐

8.7.2 Preventing Shutdown Cooling Isolations When Transferring RPS A Power Supplies In Cold Shutdown Condition (continued)

[7] On Panel 3-9-4

VERIFY PCIS will reset. ☐

[8] On Panel 3-9-3,

RESET RHR Loop I Logic as follows:

A. **MOMENTARILY DEPRESS** RHR SYS I SD CLG INBD INJECT ISOL RESET, 3-XS-74-126. ☐

B. **VERIFY** 3-IL-74-126 extinguished. ☐

[9] On Panel 3-9-3,

RESET RHR Loop II Logic as follows:

- **MOMENTARILY DEPRESS** RHR SYS II SD CLG INBD INJECT ISOL RESET, 3-XS-74-132. ☐
- **VERIFY** 3-IL-74-132 extinguished. ☐

[10] In Unit 3 Aux. Instrument Room,

VERIFY the following relays are energized:

- Relay 16A-K29 on Panel 3-9-42 ☐
- Relay 16A-K30 on Panel 3-9-43 ☐

[11] At 480V RMOV BD 3A, Compt 8C

PLACE 3-BKR-074-0048, RHR SHUTDOWN COOLING SUCT ISOL VLV FCV-74-48 (MO10-18) in the ON position. ☐

[12] At 480 RMOV 3D - Compartment 2C

PLACE 3-BKR-074-0053, RHR SYS I INBD INJECTION VLV FCV-74-53 (MO10-25A), in the ON position. (N/A if breaker was not operated in step 8.7.1[3.2]. ☐

[13] **ALIGN** Shutdown Cooling as required for current plant conditions. (Refer To 3-OI-74) ☐

8.7.3 Preventing Shutdown Cooling Isolations When Transferring RPS B Power Supplies In Cold Shutdown Condition

[1] **VERIFY** Section 8.7.1 has been performed. ☐

[2] **VERIFY** Personnel are ready to transfer RPS B Power Supply. ☐

[3] **ESTABLISH** radio communication with the Control Room Unit Operator. ☐

[4] At 250V RMOV BD 3A, Compt R1A.

A. **PLACE** 3-BKR-074-0047 RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV FCV-74-47 (MO10-17), in the OFF position. ☐

B. **STATION** an operator by the breaker for closure at the request of the Control Room Unit Operator. ☐

C. **IF** operation of the Shutdown Cooling Loop II isolation valves becomes necessary while performing this section, **THEN**

PLACE 3-BKR-074-0047, RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV FCV-74-47 (MO10-17), in the ON position. ☐

[5] **TRANSFER** RPS B power supply. (Refer To Section 8.1 or 8.2.) ☐

[6] **WHEN** RPS B power supply has been transferred and RPS Restoration has been completed, **THEN**

CONTINUE with this section of the procedure. ☐

Illustration 3
(Page 1 of 3)

Board Restriction Verification Form

NOTES
1) This form is completed and verified to confirm compliance with Precaution and Limitation 3.0N and 3.0Y, and associated board and board restrictions as given by reference drawings.
2) Board restriction verification is performed twice per shift during normal plant conditions while associated alignment is in place.
3) Board restriction verification is performed during normal plant conditions when manual or automatic loads are added to boards affected by the associated alignment.
4) Independent verification of compliance with affected board restrictions is performed by a qualified STA or SRO following first-party completion of this form.
5) This verification is to be completed for each individual board affected by the manipulation/alignment.

1.0 BOARD RESTRICTION VERIFICATION

[1] **RECORD** board affected by the manipulation/alignment.

____

_______ 1st

_______ IV

[2] **RECORD** board restriction from the associated drawing:

____

____

____

____

_______ 1st

_______ IV

Illustration 3
(Page 2 of 3)

Board Restriction Verification Form

1.0 BOARD RESTRICTION VERIFICATION (continued)

[3] **RECORD** actual board load after manipulation/alignment is completed. (Use space below, as necessary, for calculations.

Actual Board kVA determination can be done by multiplying board voltage, amps, square root of 3 and .001.

(kVA =.001 X 1.732 X (V) X (I))

The applicable Board Amp Limit can be determined by dividing kVA limit above by board voltage(V) (.480kv or 4.16kv) times the square root of 3

(i.e., 1.732). e.g., $\text{AMP Limit} = \frac{\text{KVA Limit}}{(\text{V} \times .001) \times 1.732}$

Some print notes require reducing Unit 2 auto starting loads under accident conditions by some kVA value. To determine appropriate load reduction, use 1 hp = 1 kVA. Affected loads, which are to be prevented from starting, can be 4 kV load or a 480 V load which is powered from the 4 kV Shutdown Board. kVA determination can be done by multiplying board voltage, amps, square root of 3 and .001.

(kVA =.001 X 1.732 X (V) X (I).)

____

____

____

____

____
1st

____
IV

Illustration 3
(Page 3 of 3)

Board Restriction Verification Form

1.0 BOARD RESTRICTION VERIFICATION (continued)

[4] **VERIFY** actual board load after manipulation/alignment will be BELOW board restriction from the associated drawing.

_________ 1st

_________ IV

[5] **LOG** results of this verification in Narrative Log.

_________ 1st

_________ IV

SIMULATOR EVALUATION GUIDE

TITLE : RWM FAILURE, RBCCW PUMP TRIP, FEED PUMP CONTROL FAILURE, FUEL FAILURE, , RWCU LINE BREAK WITH FAILURE TO ISOLATE, RAPIDLY DEPRESSURIZE WITH 2 AREAS APPROACHING MAXIMUM SAFE RADIATION LEVELS

REVISION : 0

DATE : January 2, 2008

PROGRAM : BFN Operator Training - HLT

PROVIDE COPY OF GOI-100-1A TO EXAMINEES

PREPARED BY: ____________________ / 1/2/08
(Operations Instructor) Date

REVIEWED BY: N/A / ____________
(LOR Lead Instructor or Designee) Date

REVIEWED BY: ____________________ / 1/5/08
(Operations Training Manager or Designee) Date

CONCURRED : ____________________ / ____________
(Operations Superintendent or Designee) Date

VALIDATION BY : ____________________ / 1/4/08
(Operations SRO: Required for Exam Scenarios Only) Date

LOGGED-IN: ____________________ / ____________
(Librarian) Date

TASKS LIST UPDATED: ____________________ / ____________
Date

NUCLEAR TRAINING

REVISION/USAGE LOG

REVISION NUMBER	DESCRIPTION OF REVISION	DATE	PAGES AFFECTED	REVIEWED BY
0	INITIAL	1/2/2008	All	

I. Program: BFN Operator Training

II. Course: Examination Guide

III. Title: RWM FAILURE, RBCCW PUMP TRIP, FEEDPUMP CONTROL FAILURE, FUEL FAILURE, RWCU LINE BREAK WITH FAILURE TO ISOLATE, RAPIDLY DEPRESSURIZE WITH 2 AREAS APPROACHING MAXIMUM SAFE RADIATION LEVELS AND EMERGENCY DEPRESSURIZE AFTER 2 AREAS REACH MAXIMUM SAFE RADIATION LEVELS

IV. Length of Scenario: 1 to 1 ½ hours

V. Examination Objectives:

A. Terminal Objective

1. Perform routine shift turnover, plant assessment and routine shift operation in accordance with BFN procedures.

2. Given uncertain or degrading conditions, the operating crew will use team skills to conduct proper diagnostics and make conservative operational decisions to remove equipment/unit from operation. (SOER 94-1 and SOER 96-01)

3. Given abnormal conditions, the operating crew will place the unit in a stabilized condition per normal, annunciator, abnormal, and emergency procedures.

B. Enabling Objectives:

1. The operating crew will start and warm-up “B” RFP In accordance with OI-6 section 5.7.

2. The operating crew will recognize and respond to a failure of RWM in accordance with 3-OI-85-5 and Tech. Specs.

3. The operating crew will recognize and respond to a RBCCW pump trip in accordance with 3-AOI-70-1.

4. The operating crew will recognize and respond to Feedpump control failure in accordance with 3-AOI-3-1.

5. The operating crew will recognize and respond to a fuel failure in accordance with ARPs and EOI-3.

6. The operating crew will recognize and respond to a break in the RWCU system and rapidly depressurize the RPV.

VI. References: The procedures used in the simulator are controlled copies and are used in development and performance of simulator scenarios. Scenarios are validated prior to use, and any procedure differences will be corrected using the procedure revision level present in the simulator. Any procedure differences noted during presentation will be corrected in the same manner. As such, it is expected that the references listed in this section need only contain the reference material which is not available in the simulator.

VII. Training Materials: **(If needed, otherwise disregard)**

A. Calculator

B. Control Rod Insertion Sheet

C. Stopwatch

D. Hold Order / Caution tags

E. Annunciator window covers

F. Steam tables

VIII. Console Operator Instructions

A. Scenario File Summary

1. File: bat HLTS3-4

	MF/RF/IOR#	Description
a.)	ior zdihs691 null	Fails 69-1 to close
b.)	imf cu04 25	RWCU suction line break
c.)	imf cu06	
d.)	bat 7048FTC	fail 70-48 to not auto close

2. File: bat HLTS3-4-1

	MF/RF/IOR#	Description
a.)	imf rm10g 1000 5:00	Fails rm14 upscale
b.)	imf rm10e 1000 10:00	Fails rm09 upscale

3. File: bat 7048ftc

	MF/RF/IOR#	Description
a.)	ior zlohs7048a[2] on	Override red light on
b.)	ior ypovfcv7048 fail_power_now	Fails power to valve
c.)	trg e1=bat 7048-1	Set trigger to 70-48 HS
d.)	Imf fw10a	Fail 3A RFP auto trips

4. File: bat 7048-1

	MF/RF/IOR#	Description
a.)	dor zlohs7048a[2]	Delete Override on red light
b.)	dor ypovfcv7048	Restore power to valve

5. File: bat HLTS3-4-4

	MF/RF/IOR#	Description
a.)	imf fw30a (none 0) 60 30 50	Run up and stop 3A RFP controller in manual

VIII. Console Operator Instructions

B. Console Operator Manipulations

UNSECURE file NRC - PW maryanne

ELAP TIME	PFK	DESCRIPTION/ACTION
Sim. Setup	Pwrst 120 (csf)	~2% power, MOC
Sim. Setup	restorepref HLTS3-4	Establishes Function Keys
Sim. Setup	setup	Verify Function Keys
Sim. Setup	esc	Clears Function Key Popup
Sim. Setup I	Manual	Place Hold Order Tags on C RFP suction and discharge valves
Sim. Setup	manual	Ensure RWM is latched with no Insert or Withdrawal blocks and comp/prog lights reset, rod group 39 – 06-47 selected
Sim. Setup	Manual	Verify 3C RFP suct & disch valve lights extinguished. If not, bat 2crfptag
Sim Setup	<Shift F1> bat 7048ftc	Set 70-48 to not close on low pressure and fail 3A RFP auto trips

---

After RFP warmed and When requested by Examiner	F3	Fails RWM (imf rd14a)

ROLE PLAY: **If asked, have not performed a startup with RWM bypassed within last calendar quarter**

ROLE PLAY: **If requested to verify open 3-1-155 and 3-1-156, report that they are open**

*** WATCH CAREFULLY TO SEE IF NEED TO MANUALLY FIRE TRIGGER**

After Tech Specs addressed for RWM	F8	Trips A RBCCW pump (imf sw02a)
If trigger does not fire when 70-48 taken to close	<Shift F2>	Manually fire trigger

VIII. Console Operator Instructions (continued)

B. Console Operator Manipulations

If requested to align spare RBCCW pump to Unit 3 Wait 3 minutes	F9	Aligns spare RBCCW pump to Unit 3 (mrf sw02 align)
If requested to reset local RWCU panel alarms	F12	mrf an01e reset
After spare RBCCW pump aligned and RWCU returned to service	<Shift F3> Bat HLTS3-4-4	Fails RFP governor in raise direction in manual for 30 sec
Two (2) Minutes after Feedpump governor problem	F6	fuel failure (imf th23 5 15:00)
When directed by examiner	F7	RWCU line break with failure to isolate (bat HLTS3-4)

ROLE PLAY: **If requested to attempt to close 69-1 locally at the breaker, wait 5 minutes and report it will not close**

ROLE PLAY: **If requested to check Aux Inst rm, report 835 A&C and 835 B&D reading 90 deg F and fairly steady**

After attempts to close 69-1 are made	F10	Causes Rad monitors to reach max (bat HLTS3-4-1)

Terminate the scenario when the following conditions are satisfied or when requested by Chief Examiner:

1. Reactor Water level restored between +2 to +51"
2. RPV rapidly depressurized
3. RPV emergency depressurized

SECURE file NRC - PW maryanne

IX. Scenario Summary

The plant is at approximately 2% power withdrawing control rods to open sufficient bypass valves to roll the main turbine . "B" RFP needs to be started and warmed in preparation for water level control.

During the control rod withdrawal, the RWM will experience a program fault which will block rod movement. Tech. Specs will be addressed and control rod withdrawal will continue when a second licensed operator is present to ensure withdrawal is in accordance with the BPWS.

An RBCCW pump will trip causing RWCU to be secured and the spare RBCCW pump aligned to Unit 3 and the RWCU system returned to service.

The In-service RFP will experience a governor fault causing it to inject cold water into the RPV causing a power spike and some fuel failure. Later the RWCU system develops and leak and fails to isolate requiring entry into EOI-3 and subsequent rapid depressurization due to 2 areas approaching max safe radiation levels.

X. Information to Evaluators:

A. Ensure recorders are inking and recording and ICS is active and updating.

B. Assign Crew Positions based on the required rotation.

1. SRO ____________________
2. ATC ____________________
3. BOP ____________________

C. Conduct a shift turnover with the Shift Manager and provide the Shift Manager with a copy of the Shift Turnover.

D. Direct the shift crew to review the control board and take note of present conditions, alarms, etc.

E. Terminate the scenario when the following conditions are satisfied are at the request of the floor/lead instructor/evaluator.

1. Reactor water level restored at +2" to +51"

2. RPV rapidly depressurized

3. RPV emergency depressurized when 2 areas are above max safe values.

XI. Simulator Event Guide

EVENT 1: Warming up second RFP

POSITION	TIME	EXPECTED ACTIONS
SRO	______	Directs warming up B RFP in accordance with 3-OI-3
BOP	______	Warms up "B" RFP utilizing section 5.6 of 3-OI-3.
	______	Place in auto and verify open RFP min flow valve 3-FCV-3-13
	______	Place 2B start/local enable 3-HS-46-138A in start and observe RFP accelerates to 600 rpm
	______	Verify no abnormal rubbing or vibration is observed
	______	Raise speed to ~1000 rpm using 3-HS-46-9A
	______	Place TG motor 3-HS-3-127A in Auto
	______	Depress 3B trip 3-HS-3-127A and verify HP and LP stop valves close
	______	Verify TG auto engages or RFP rolling on min flow
	______	Depress 3B trip reset 3-HS-3-150A and verify blue light extinguishes and HP and LP stop valves open
	______	Place 3B start/local enable 3-HS-46-138A in start and observe RFPT speed increases to ~ 600 rpm

XI : Simulator Event Guide

EVENT 2: RWM FAILURE

POSITION	TIME	EXPECTED ACTIONS
ATC	______	Announces "RWM ROD BLOCK" 3-XA-55-5B window 35 alarm and refers to ARP. Verifies Control Rod positions
SRO	______	Directs ATC to bypass RWM per OI-85 Refers to T.S. 3.1, 3.3, table 3.3.2.1-1
	______	Contacts Rx Engineer
ATC	______	Refers to section 8.17 of OI-85 and places 3-XS-85-9025 in Bypass.
ATC	______	Checks manual bypass light lit and all others out._
SRO	______	Determines T.S> 3.3.2.1 condition C applies. Greater than 12 rods withdrawn and 2nd person to verify compliance with BPWS.

XI. Simulator Event Guide

EVENT 3: LOSS OF 3A RBCCW pump

POSITION	TIME	EXPECTED ACTIONS
BOP	______	Responds to loss of RBCCW pump 3A trip and attempts to restart 3A RBCCW pump and reports it failed to start.
SRO	______	Directs securing RWCU pumps per 3-AOI-70-1
BOP	______	Secures RWCU pumps and reports that the 3-FCV-70-48 sectionalizing valve failed to close.
US	______	Directs closure of the 3-FCV-70-48
	______	Directs placing Spare RBCCW pump in service.
	______	Dispatches personnel to investigate pump loss
		May contact Rx Engineer about heat balance
BOP	______	Closes 3-FCV-70-48
		After Spare RBCCW pump placed in service, re-opens 70-48 and returns RWCU to service per OI-69. (conditional, SRO may not direct valve to be opened after failure to auto close.)
BOP	______	Opens 69-8 Starts A(B) RWCU Pump Coordinates with AUO to roll demins in service Starts second RWCU Pump

XI. Simulator Event Guide

EVENT 4: FEEDWATER CONTROLLER FAILURE

POSITION	TIME	EXPECTED ACTIONS
ATC	______	Observes period rise by meter or annunciator and checks for cause of reactivity addition
ATC	______	Ranges IRMs as necessary to prevent a reactor scram
BOP	______	Attempts to take control of A RFP by adjusting 3-HS-46-8A and reports that "A" RFP cannot be controlled
SRO	______	**Directs tripping "A" RFP and using "B" RFP for RPV level control**
BOP	______	**Trips "A" RFP by depressing 3-HS3-125A and raises "B" RFP speed by using 3-HS-46-9A**
BOP	______	Opens "B" RFP discharge valve 3-HS-3-12A when "B" RFP discharge pressure is within 250 lbs of reactor pressure.

XI. Simulator Event Guide

EVENT 5: FUEL FAILURE DUE TO COLD WATER INJECTION

POSITION	TIME	EXPECTED ACTIONS
BOP	_____	Announces "TURBINE BUILDING HIGH RADIATION" and determines which area and evacuates that area.
	_____	Announces "OFF-GAS ANNUAL RELEASE LIMIT EXCEEDED" and responds per ARP
BOP/SRO	_____	Notifies Chemistry to perform analysis and Radcon
SRO	_____	Declares a NOUE on a valid OG pretreatment rad alarm or Main Steam Line rad Hi Hi.

XI. Simulator Event Guide

EVENT 6: RWCU LINE SUCTION BREAK

POSITION	TIME	EXPECTED ACTIONS
BOP	______	Announces “RX BLDG HIGH RADIATION” and determines which area and evacuates that area. North and South RWCU area.
BOP	______	Reports on RWCU leak detection alarms
SRO	______	Enters EOI-3 on either high temp or high radiation
ATC	______	Recognizes that 69-1 failed to isolate and attempts to manually close
Crew	______	Directs outside personnel to attempt to close 69-1 locally at the breaker.
SRO	______	Directs Rx Scram before any area temp is above the maximum safe operating temperature.
ATC	______	Scrams reactor and provides scram report
SRO	______	Directs ATC to perform actions of 3-AOI-100-1

XI. Simulator Event Guide

EVENT 6: RWCU LINE SUCTION BREAK

POSITION	TIME	EXPECTED ACTIONS
SRO/BOP	______	Continue to monitor and trend secondary area temps and radiation levels
BOP	______	Reports that 2 areas are approaching maximum safe radiation levels
SRO	______	Directs rapid depressurization ot the RPV using BPVs
BOP	______	Opens all BPVs using the Jack
BOP/ATC	______	Coordinate level control during depressurization to prevent flooding the RPV
BOP	______	Determines that 2 areas are above max safe radiation values

XI. Simulator Event Guide

EVENT 6: RWCU LINE SUCTION BREAK

POSITION	TIME	EXPECTED ACTIONS
SRO	______	**Determines that Emergency Depressurization is Required and enters C2**
	______	**Directs BOP to open all ADS valves**
BOP	______	**Opens all ADS valves**
SRO	______	When the shutdown cooling pressure interlock clears, directs BOP to place shutdown cooling in service per Appx. 17D
BOP	______	Places Shutdown Cooling in service per Appendix 17D

XII. Crew Critical Tasks

	TASKS	SAT/UNSAT
1.	Trips "A" RFP prior to reaching Main Steam Lines	______________
2.	Emergency Depressurize when 2 areas reach maximum safe values.	______________

XIII. Scenario Verification Data

EVENT	TASK NUMBER	K/A		RO	SRO	CONTROL MANIPULATION
1.	Warm up RFP	259001 A4.02		3.9	3.7	
2.	RWM Failure	201006 A4.01		3.2	3.4	
3.	RBCCW Pump Trip	295018				
		AK3.03		3.1	3.4	
		AA1.01		3.3	3.4	
		AK3.04		3.3	3.3	
4.	RFP Governor failure	259001	A2.07	3.7	3.8	
		295008	AA1.08	3.5	3.5	
5.	Fuel Failure	295014	AA1.05	3.9	3.9	
			AA1.07	4.0	4.1	
6.	RWCU Line Break	295033	EA1.05	3.9	4.0	
			EK3.01	3.3	3.5	

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER HLTS-11

7 Total Malfunctions Inserted; List: (5-8)

1) RWM Failure
2) RFP controller failure
3) Fuel Damage
4) RWCU line break
5) "A" RBCCW pump trip
6) Failure of 69-1 to close
7) Failure of RFPs to trip on Hi level

1 Malfunctions That Occur After EOI Entry; List: (1-2)

1) RWCU line Break

2 Abnormal Events; List: (2-4)

1) RFP control failure
2) RBCCW pump trip

1 Major Transients; List: (1-2)

1) RWCU line break (small LOCA)

2 EOIs used; List: (1-2)

1) EOI-1
2) EOI-3

1 EOI Contingencies Used; List: (0-2)

1) C2

63 Run Time (minutes)

52 EOI Run Time (minutes); 83 % of Scenario EOI Run Time

2 Crew Critical Tasks (2-3)

Yes Technical Specifications Exercised (yes/no)

SHIFT TURNOVER SHEET

Equipment Out of Service/LCOs C RFP is uncoupled and awaiting overspeed testing. Suction and Discharge valves are tagged.

Operations/Maintenance For the Shift: Continue with reactor startup at step 5.66.8 of 3-GOI-100-1A. Continue with warm-up of “B” RFP per OI-3 at step 5.6.2.2.16. Thrust bearing/ Overspeed/ Stop Valve and Control Valve tests are complete for “B” RFP.

Unusual Conditions/Problem Areas: Power System Alert in effect for the next 36 Hours.

TVA

Browns Ferry Nuclear Plant

Unit 3

General Operating Instruction

3-GOI-100-1A

Unit Startup

Revision 0074

Quality Related

Level of Use: Continuous Use

Effective Date: 12-05-2007

Responsible Organization: OPS, Operations

Prepared By: C. E. Heitzenrater

Approved By: John T. Kulisek

Current Revision Description

Type of Change: Enhancement Tracking Number: 081

Pages 81

PERs None

PCRs 07004789

- Page 81 Step 5.0[30.2] - Modify the 6th bullet on Reactor Water temp to allow obtaining data from either Recirc Discharge OR Suction temperature and the ability to use ICS for either reading. This closes PCR 07004789.

Table of Contents

Table of Contents (continued)

1.0 PURPOSE

This instruction provides precautions and limitations, prerequisites, and procedural steps to take the unit from either MODE 4 or MODE 3 to full power operation. It also provides an integrated plant approach to raising power back to full load after a power reduction. Provided the Reactor remains in the MODE 1, the Shift Manager will enter this procedure at the appropriate step in Section 5.0 depending on the present power level.

2.0 REFERENCES

2.1 Technical Specifications

Section 3.1.3, Control Rod Operability.

Section 3.1.7, Standby Liquid Control (SLC) System.

Section 3.1.8, Scram Discharge Volume (SDV) Vent and Drain Valves.

Section 3.2, Power Distribution Limits.

Section 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR).

Section 3.2.2, Minimum Critical Power Ratio (MCPR).

Section 3.2.3, Linear Heat Generation Rate (LHGR).

Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation.

Section 3.3.2.1, Control Rod Block Instrumentation.

Section 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation.

Section 3.4.1, Recirculation Loops Operating.

Section 3.4.2, Jet Pumps.

Section 3.4.3, Safety/Relief Valves (S/RVs).

Section 3.4.6, RCS Specific Activity.

Section 3.4.9, RCS Pressure and Temperature (P/T) Limits.

Section 3.5.1, ECCS Operating.

Section 3.5.3, RCIC System.

Section 3.6.1.1, Primary Containment.

2.1 Technical Specifications (continued)

Section 3.6.4.1, Secondary Containment.

Section 3.6.4.3, Standby gas Treatment (SGT) System.

Section 3.7.1, Residual Heat Removal Service Water (RHRSW) System.

Section 3.7.2, Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS).

Section 3.8, Electrical Power Systems.

2.2 Technical Requirements Manual

Section 3.1, Reactivity Control.

Section 3.3.1, Reactor Protection System (RPS) Instrumentation.

Section 3.3.3, ECCS Instrumentation.

Section 3.3.5, Surveillance Instrumentation.

Section 3.4.1, Coolant Chemistry.

Section 3.5, Emergency Core Cooling Systems.

Section 3.6, Containment Systems.

Section 3.6.1, Primary Containment Purge System.

Section 3.8, Auxiliary Electrical System.

2.3 Final Safety Analysis Report

Chapter 3.0, Reactor.

Chapter 4.0, Reactor Coolant System.

Chapter 5.0, Containment.

Chapter 6.0, Core Standby Cooling Systems.

Chapter 7.0, Control And Instrumentation.

Chapter 8.0, Electrical Power Systems.

Chapter 10.0, Auxiliary Systems.

Chapter 11.0, Power Conversion Systems.

Chapter 13.0, Conduct of Operations.

2.4 Plant Instructions

2.4.1 Operating Instructions.

3-AOI-100-1, Reactor Scram.

3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.

0-GOI-300-4, Switchyard Manual.

3-OI-1, Main Steam System.

3-OI-2, Condensate System.

3-OI-2A, Condensate Demineralizers System.

0-OI-2B, Condensate Storage and Transfer System.

0-OI-2C, Demineralized Water System.

3-OI-3, Reactor Feedwater System.

3-OI-4, Hydrogen Water Chemistry System.

3-OI-6, Feedwater Heating and Misc Drains System.

0-OI-12, Auxiliary Boilers.

0-OI-18, Fuel Oil System.

2.4.1 Operating Instructions. (continued)

0-OI-20, Central Lubricating Oil System.

0-OI-23, Residual Heat Removal Service Water System.

3-OI-24, Raw Cooling Water System.

0-OI-25, Raw Service Water System.

0-OI-26, High Pressure Fire Protection System.

3-OI-27, Condenser Circulating Water System.

3-OI-27A, Screen Wash System.

3-OI-27B, Amertap Condenser Tube Cleaning System.

0-OI-27C, Cooling Tower System.

0-OI-29, Potable Water System.

3-OI-30A, Refueling Zone Ventilation System.

3-OI-30B, Reactor Zone Ventilation System.

3-OI-30C, Turbine Building Ventilation System.

0-OI-30D, Radwaste Building Ventilation System.

0-OI-30E, Service and Office Building Ventilation System.

0-OI-30F, Common and DG Building Ventilation.

0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System.

0-OI-32, Control Air System.

3-OI-32A, Drywell Control Air System.

0-OI-33, Service Air System.

0-OI-34, Vacuum Priming System.

3-OI-35, Generator Hydrogen Cooling System.

3-OI-35A, Stator Cooling System.

3-OI-35B, Generator Hydrogen Seal Oil System.

3-OI-35C, Generator Circuit Breakers.

2.4.1 Operating Instructions. (continued)

0-OI-39, CO_2 System.

0-OI-40, Station Drainage System.

0-OI-44, Building Heating System.

3-OI-47, Turbine-Generator System.

3-OI-47A, EHC System.

3-OI-47B, Main Turbine Lube Oil System.

3-OI-47C, Seal Steam System.

0-OI-48, Integrated Computer System.

0-OI-53, Demineralizer Backwash Air System.

0-OI-57A, Switchyard and 4160V Electrical System.

0-OI-57B, 480V/240V AC Electrical System.

0-OI-57C, 208V/120V AC Electrical System.

0-OI-57D, DC Electrical System.

3-OI-63, Standby Liquid Control System.

3-OI-64, Primary Containment System.

0-OI-65, Standby Gas Treatment System.

3-OI-66, Off-Gas System.

0-OI-67, Emergency Equipment Cooling Water System.

3-OI-68, Reactor Recirculation System.

3-OI-69, Reactor Water Cleanup.

3-OI-70, Reactor Building Closed Cooling Water System.

3-OI-71, Reactor Core Isolation Cooling System.

3-OI-73, High Pressure Coolant Injection System.

3-OI-74, Residual Heat Removal System.

2-OI-74, Residual Heat Removal System.

2.4.1 Operating Instructions. (continued)

3-OI-75, Core Spray System.

3-OI-76, Containment Inerting System.

0-OI-77A, Waste Collector/Surge System Processing.

0-OI-77B, Floor Drain Collector System Processing.

0-OI-77C, Radwaste Filter and Demineralizer System.

0-OI-77D, Backwash Receivers and Phase Separators System.

3-OI-78, Fuel Pool Cooling and Cleanup System.

0-OI-82, Standby Diesel Generator System.

3-OI-82, Standby Diesel Generator System.

3-OI-84, Containment Atmosphere Dilution System.

3-OI-85, Control Rod Drive System.

3-OI-90, Radiation Monitoring System.

3-OI-92, Source Range Monitors.

3-OI-92A, Intermediate Range Monitors.

3-OI-92B, Average Power Range Monitoring.

3-OI-92C, Rod Block Monitor.

3-OI-94, Traversing Incore Probe System.

3-OI-99, Reactor Protection System.

2.4.2 Surveillance Instructions

3-SR-3.1.1.1, Reactivity Margin Test.

3-SR-3.1.2.1, Reactivity Anomaly and Exposure.

3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check.

3-SR-3.1.3.5(B), CRD Coupling Integrity Check After Refueling or Maintenance.

3-SR-3.1.4.1, Scram Insertion Times.

3-SR-3.1.6.1, BPWS Compliance Verification.

3-SR-3.1.8.1, Scram Discharge Volume Valve Open.

3-SR-3.1.8.2, Scram Discharge Volume Valve Operability.

3-SR-3.3.1.1.2, APRM Output Signal Adjustment.

3-SR-3.3.1.1.3(IRMs), Intermediate Range Monitor Functional Test With Reactor Mode Switch NOT In Run Position.

3-SR-3.3.1.1.5, Source Range Monitors (SRMs) and Intermediate Range Monitors (IRMs) Overlap Verification.

3-SR-3.3.1.1.6(IRMs), IRM Gain Adjustment and IRM/APRM Overlap Verification.

3-SR-3.3.1.1.9(IRM A-H), Intermediate Range Monitor (IRM) Channel A-H Calibration.

3-SR-3.3.1.1.13(APRM-1-4), Average Power Range Monitor Calibration-APRM-1-4.

3-SR-3.3.1.1.14(2e), Average Power Range Monitor (APRM) 2-OUT-OF-4 Voter Logic Functional Test.

3-SR-3.3.1.1.16(APRM-1-4), Average Power Range Monitor Functional Test-APRM-1-4.

3-SR-3.3.1.2.5&6, Instrumentation That Initiates Rod Block/Scrams Source Range Monitor (SRM) Functional Test With Reactor Mode Switch NOT in RUN Position.

3-SR-3.3.1.2.7(SRM A-D), Source Range Monitor (SRM) Calibration and Functional Test.

3-SR-3.3.2.1.1, Rod Block Monitor(RBM) Functional Test.

3-SR-3.3.2.1.2, RWM Functional Test For Startup.

3-SR-3.3.2.1.4(A), Rod Block Monitor (RBM) Calibration and Functional Test.

2.4.2 Surveillance Instructions (continued)

3-SR-3.3.2.1.4(B), Rod Block Monitor (RBM) Calibration and Functional Test.

3-SR-3.3.2.1.5, Verification Of RWM Automatic Bypass Setpoint.

3-SR-3.3.2.1.7, RWM Program Verification.

3-SR-3.3.5.1.6(ADS A), ADS Logic System Functional Test - Bus A, Time Delay Relay Calibration, and Bus Power Monitor Test.

3-SR-3.3.5.1.6(ADS B), ADS Logic System Functional Test - Bus B, Time Delay Relay Calibration, and Bus Power Monitor Test.

3-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation

3-SR-3.4.1(DLO), Reactor Recirculation System Dual Loop Operation

3-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test.

3-SR-3.4.9.1(1), Reactor Heatup or Cooldown Rate Monitoring.

3-SR-3.4.9.5-7, RPV Temperature Monitoring with Head Tensioned.

3-SR-3.5.1.5, Reactor Recirculation Pump Discharge Valves Cycling.

3-SR-3.5.1.8, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150 psig Reactor Pressure.

3-SR-3.5.1.10, Automatic Depressurization System Simulated Automatic Actuation Test.

3-SR-3.5.3.4, RCIC System Rated Flow at 150 psig.

3-SR-3.6.1.2.1, Primary Containment Airlock Local Leak Rate Test.

3-SR-3.6.1.3.3, Primary Containment Isolation Manual Valves and Blind Flanges Inside Containment Position Verification.

3-SR-3.6.1.3.5(SD), Valves Cycled During Cold Shutdown.

3-SR-3.3.1.1.I, Core Thermal Hydraulic Stability.

3-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping (ASME Section III, Class 1).

3-SI-4.7.A.5.c, Control Air/Drywell Control Air Isolation Verification.

3-SI-4.6.B.1-4, Reactor Coolant Startup Chemistry.

2.4.3 Other Instructions

0-TI-248, Station Reactor Engineer.

MCI-0-064-HLT001, Drywell Personnel Airlock Doors.

MSI-0-001-VSL001, Reactor Vessel and Cavity Disassembly and Reassembly.

MSI-0-064-HLT002, Opening and Closing of Primary Containment Hatches.

RCI-17, Surveillance and Door Control of Prohibitive High Radiation Areas.

2.4.4 Administration Procedures

SPP-5.3, Chemistry Control.

SPP-10.3, Verification Program.

SPP-10.4, Reactivity Management Program

OPDP-1, Conduct of Operations

SPP-7.2, Outage Management

2.5 Miscellaneous Documents

BP-250, Corrective Action Program Handbook.

BWROG-94078, BWR Owner's Group Guidelines for Stability Interim Corrective Action.

GE SIL 316, Reduced Notch Worth Procedure.

GE SIL 380, BWR Core Thermal Hydraulic Stability.

GE SIL 498, SB-1 and SB-9 Switch Lockup.

Incident Investigation II-B-91-129, Unit 2 manual scram due to high torus temperature - caused by temperature stratification from extended RCIC run exhaust steam discharge.

INPO SER 89-006, Withdrawal of Safety Rod Group Out of Sequence.

INPO SER 89-022, Intermittent Failure of Westinghouse Type DS and DSL Breakers to Close.

INPO SOER 88-002, Premature Criticality Events During Reactor Startup.

INPO SOER 90-003, Nuclear Instrumentation Miscalibration.

2.5 Miscellaneous Documents (continued)

INPO SER 92-19, Power Oscillations at Boiling Water Reactors.

SOER 01-1, Unplanned Radiation Exposures

Licensee Event Report 260/94009, Missed Technical Specification (Tech Specs) Surveillance before Reactor Startup as a Result of a Misunderstanding of Tech Specs.

NRC Generic Letter 94-02, Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors.

NRC IE Bulletin 79-12, Short Period Scrams at BWR Facilities.

NRC IE Bulletin 88-07, Supplement 1, Power Oscillations in Boiling Water Reactors.

NRC Inspection Report 84-45 response, RIMS L44850329806, Identify required steps in startup procedures.

NRC Inspection Report 85-15, item 4.a, NDT Curves, Technical Specification Figure 3.6-1, Out of Date.

NRC IE Notice 89-030, High Temperature Environments at Nuclear Power Plants.

NRC Information Notice 92-740, Power Oscillations at Washington Nuclear Power Unit 2.

NSRB Item A258-4, Review procedures to preclude an event similar to SER 24-91, inadequate control of reactivity changes during plant shutdown results in unwanted transient.

Q13958, issued 09/05/90.

Q16997B - Doors, Hatches, and Penetrations Required to be Closed to Maintain EQ Boundaries.

S17557B, Combined Zone Secondary Containment.

Scram Frequency Reduction Committee (SFRC) Recommendations 17, G-20-1 and G-20-2 concerning additional SRO assisting during startup or shutdown.

Technical Specifications Assessment Report, Item D89, Clearly specify the expectations for satisfying Tech Specs LCOs prior to changing operation conditions.

TVA-BFE-052, Extended Load Line Limit Analysis.

Jerry Robertson Memorandum to G.C. Campbell, Use of Increased Core Flow (L32 890302 901).

2.5 Miscellaneous Documents (continued)

W.N. Hannum (NSRB) Memorandum to R.R. Calabro and G.G. Campbell (L42 890927 800).

Letter from O. D. Kingsley to W. J. Museler, DOWNPOWERING OF NUCLEAR UNITS UNDER LOW SYSTEM LOAD CONDITIONS, March 1, 1996 (A00 960226 150)

T.A. Keys Memorandum to K.L. Welch, Use of Increased Core Flow (ICF) at Browns Ferry Nuclear Plant (L32 920709 801).

TOE 0-97-064-0823 Parallel purging of the Torus and Drywell.

Drawing 0-48N954, R003 - Miscellaneous Steel Refueling Facilities General Plan & Elevation.

TVA-BFN-TS-384, Technical Specification (Tech Specs) Change TS-384 Request for License Amendment for Power Uprate Operation and NEDC-32751P, Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 and 3 (R08-980316-888).

GE-NE-B13-01866-39, Task Report 39 Summary of System Evaluations and Proposed Changes to Design Criteria Documents (W79-980427-005).

BFN-IPIP-TASK35, Computer Process Alarm Limits (W79-980319-002).

GE-NE-B13-01866-2, Task Report 2 Power Uprate Evaluation Report for Power/Flow Operating Map (RIMS W79-971023-002).

ED-N0003-980030, BFN Setpoint and Scaling Calculation (R14-980422-104)

ND-Q0068-980011, Power/Flow Map (RIMS R14-980423-102).

GE-NE-B13-01866-05, Power Uprate Evaluation Task Report for Browns Ferry Units 1, 2 and 3 Transient Analysis (RIMS W79-971004-005).

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General

A. The Critical Steps warning represents a step, or series of steps, for an activity that requires additional focus, attention, and increased awareness. The Operator performing these steps for the activity needs to verify the Unit Supervisor and other Control Room staff are aware of the evolution. PEER checks are required for this activity and short briefs need to be made prior to performing the evolution. Included in the briefs are worst case scenario and contingencies.

B. [SFRC/C] Scram Reduction Recommendations, G-20-1 and G-20-2, require an additional SRO licensed operator to assist with BOP operations when performing power maneuvers during unit startup or shut down with Reactor power less than 60%. This individual is not limited to the Control Room. [SFRC-17, G-20-1 & -2]

C. Unit Supervisor's permission is required to reject water to main condenser from the Reactor Water Cleanup (RWCU) System without a RWCU filter in service.

D. [TSAR/C] Coolant Leakage Detection Systems is required to be in service prior to reaching 212°F. [Item D45]

E. The bottom layer of Reactor Well shield blocks are required to be in place prior to exiting Mode 4 (cold shutdown). The top layers of Reactor Well Shield Blocks are required to be installed BEFORE exceeding 10 days of power operation. This requirement is established based on evaluation of Source Term effects to Operations Personnel. [BFN PER 02-005145-000]

F. [TSAR/C] Any time a unit, system, or plant mode or operational condition change is required, the Unit Supervisor, Unit Operators, Shift Manager, and STA if manned, are required to review all applicable LCOs prior to the mode change (and as soon as practical during an emergency) to ensure compliance with Tech Specs. [Item D89]

G. To ensure that all 3-SR-2 instrumentation meets the Instrument Checks for the required modes. The 3-SR-2 readings will be taken prior to the Mode or Condition changes. The STA will verify that the readings will allow the Reactor Mode or Condition Changes.

H. The maximum rate of temperature change (rise or fall) in Wheeler Reservoir is limited to 10°F per hour as measured at the downstream temperature control point.

I. Reactor Core Isolation Cooling or High Pressure Coolant Injection Systems are not normally used for level control during Reactor startup.

3.1 General (continued)

J. Chemistry parameters are specified and corrective actions for any out-of-limit chemistry parameter are delineated in SPP-5.3. Special attention is required for chemistry parameters in Tech Specs 3.4.6 and Technical Requirements Manual 3.4.1, Coolant Chemistry.

K. Noble Metal Injection will cause a higher radiation level than Normal throughout the plant. Following a startup from Noble Metals injection this condition will diminish over time up to 6 weeks.

Therefore, if this startup is being performed following Noble Metal Injection or during the time period where radiation levels are still higher than normal from a Noble Metal Injection Shutdown, then, to minimize radiation levels, the Hydrogen Water Chemistry System should not be aligned during the startup. The Duty Engineer will make recommendations in determining if the Hydrogen Water Chemistry System should be placed in service.

L. All MSIVs should be OPEN prior to 25% Reactor power.

3.2 Coolant and Metal Temperatures

A. Lowering Reactor head flange and/or head temperature below the temperature of fully tensioning Reactor head bolts may result in bolt relaxation and potential leakage when Reactor vessel is pressurized during startup.

B. [TSAR/C] Monitoring coolant temperature when in MODE 4 with the vessel head tensioned is performed using 3-SR-3.4.9.5-7. [Item D41]

C. The following limitations apply to Reactor heatup and/or cooldown:

1. When Reactor coolant temperature is less than 215°F, a maximum heatup rate limit of 50°F/hr will reduce the 0_2 and Hydrogen Peroxide content of the coolant.

2. During Reactor Heatup with Reactor coolant temperature greater than or equal to 215°F, and during Reactor Cooldown, the optimum rate of temperature change is 20°F every 15 minutes. This will ensure the administrative limit of 90°F/HR is not exceeded. Do not Attempt to "makeup" for time intervals which fall short of 20°F. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines will assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is not exceeded.

3.2 Coolant and Metal Temperatures (continued)

3. During Reactor heatup, operators should use metal temperatures as a reminder that as metal heats up, the moderator HEATUP RATE will rise with the same amount of heat input.

D. Minimizing operation with low feedwater flow and temperature or cold feedwater flow cycling limits thermal duty on feedwater nozzles (REFER TO 3-OI-3).

3.3 Primary Containment

A. [II/F] Prior to initiating any event which adds, or has the potential to add, heat energy to the suppression chamber, the Unit Supervisor or Shift Manager will evaluate the necessity of placing suppression pool cooling in service. This is due to the potential of developing thermal stagnation during sustained heat additions. [II-B-91-129]

B. When containment integrity is required, airlock door seals should be tested within seven days after each containment access per 0-TI-360 App A.

3.4 Control Rods, Reactivity Control and Relative Instrumentation

A. [NRC/C] Startups are performed using 3-SR-3.1.3.5(A) to incorporate Reduced Notch Worth Procedure (RNWP) and Banked Position Withdrawal Sequence (BPWS) recommended by G.E. [IE Bulletin 79-12, LER 260/84004]

B. [NER/C] Periodic pauses during control rod withdrawal are necessary to allow for stabilization of neutron level and collection of data for estimating proximity to critically. [SER 89-006, SOER 88-002]

C. [INPO/C] Adjustment of Nuclear Instrumentation readings downward to match other indications without a full investigation and comparison with all available methods to measure power level may result in non-conservative power readings and protective setpoints. [SOER 90-003, SOER-88-002]

D. [NER/C] If SRMs or IRMs exhibit noise spikes during startup, control rod withdrawal should be suspended and an assessment of SRM or IRM operability performed in accordance with 3-OI-92 or 3-OI-92A, as applicable. [SOER 88-002]

E. [NER/C] Activities that can directly affect core reactivity are of a critical nature and require strict procedural compliance, along with conservative actions. [INPO SER 89-006, SOER 88-002]

F. [NSRB/C] Reactivity can be added without moving control rods due to changing plant conditions (such as lowering moderator temperature, lowering xenon concentration, rising Reactor pressure, and rising feedwater flow) especially at low power. Awareness of these conditions and monitoring core instrumentation for these changes is required. [A258-4]

3.4 Control Rods, Reactivity Control and Relative Instrumentation (continued)

G. In the event of an unexplained change in reactivity during an approach to criticality, the approach to criticality must cease and the reactor core be made sufficiently subcritical to prevent an inadvertent criticality. Approval of the Plant Manager or his designee is required to resume the approach to criticality.

H. Reactor Engineering is required to be contacted to monitor flux shaping prior to all power ascensions.

I. During the initial startup from MODE 4 following a refueling outage, 3-SR-3.1.1.1, Reactivity Margin Test, is required to be performed in conjunction with the performance of 3-GOI-100-1A.

J. [NRC/C] Core Thermal-Hydraulic Stability, the reactor is required to be verified outside Regions I, II & III. When OPRM's are INOP, REFER TO 3-SR-3.3.1.1.I. [NCO 940245010]]

K. For Unit 3 Middle of Core Life to End of Core Life, the moderator temperature coefficient of reactivity becomes positive as control rods are withdrawn for startup when moderator temperature is below 350°F. The resulting effect will be for Reactor power to rise until the moderator begins boiling and inserting negative void reactivity. Exercise additional caution when withdrawing control rods under this condition.

L. [QA/C] SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with the following reactivity control equipment bypassed unless bypassing of this equipment is specifically allowed within approved procedures:

1. Rod Worth Minimizer
2. Rod Block Monitor
3. Source Range Monitors
4. Intermediate Range Monitors
5. Average Power Range Monitors
6. Refueling Interlocks
7. Integrated Computer System [ISE-NPS-92-R01]
8. OPRM Trip Function

3.5 Thermal Limits

A. Expected changes for Core Power and CPR during transients.

Raise In Variables	CP	CPR	Reason / Transient
Subcooling	↑	↓*	Loss of Feed water heating. Cold Water Injection.
Core Flow	↑	↓*	Runaway Recirc Pump.
Pressure	↓	↓	Turbine Trip W/O Bypass. MSIV Closure
Local Power Factor	↓	↓	Control Rod Drop. Xenon Shift
Axial Flux Shape	↓	↓	Core Age

* Bundle Power Raise is greater than critical power.

B. Operating in Single Loop Operation requires Safety Limit adjustments by performing 3-SR-3.4.1(SLO). Per Tech Specs 3.4.1, a completion time of 24 hours is allowed from the time of the Recirc Pump trip. However, these actions should be performed as soon as possible.

C. Per Unit 3 TRM COLR the Thermal limits and off-rated corrections are provided for Recirculation Pump Trip out-of-service and/or Turbine Bypass out-of-service conditions. These events are analyzed for separate and/or concurrent inoperability. The Shift Manager is required to make determination if startup with the EOC-RPT will be disabled for startup.

D. [TSAR/C] Steady-state power operation is not permitted at Reactor vessel pressure of greater than or equal to 1055 psia. MCPR analyses are not valid above 1055 psia Reactor pressure. [Item D94]

3.6 EHC and Main Turbine

A. If hotwell pressure drops below -7"Hg with EHC pressure set less than 50 psi above Reactor pressure and bypass jack above zero, bypass valve operation could result.

B. Hotwell pressure above -25 inches Hg could result in low pressure turbine last stage bucket failure.

C. Abnormal vibration in the main turbine during startup could result in turbine damage.

3.6 EHC and Main Turbine (continued)

D. If Turbine seals have been in service with the Turbine Turning Gear secured and the unit is to be returned to operation, the following restrictions apply:

1. Turbine placed on the turning gear for 10 times as long as the period it was stopped, up to 4 hours, then check the eccentricity,

   AND

2. If the eccentricity is higher than normal, the turbine is required to be left on the turning gear until the eccentricity indication has reached and maintained its normal minimum width for at least one hour. REFER TO 3-OI-47.

E. During a Reactor startup, with an initial pressure greater than 150 psig and EHC being unavailable prior to the startup, the EHC system should be placed in service when it becomes available. The Main Turbine Shell and Chest warming may begin when the conditions are met. Due to main turbine shell and chest warming requirements, the EHC System should be placed in service prior to 950 psig.

F. The EHC Control System can be used in either Reactor Pressure control or Header Pressure control. While in Header Pressure control, a single header pressure input failing high could cause the bypass valves to open. While in Reactor Pressure control, a single Reactor Pressure input failing high will not affect the bypass valves. For this reason, Reactor Pressure control is the preferred mode of operation for the EHC Control System.

G. Swapping pressure control sources ("HEADER PRESSURE CONTROL" to "REACTOR PRESSURE CONTROL" or "REACTOR PRESSURE CONTROL" to "HEADER PRESSURE CONTROL") may cause the turbine bypass valves to open, depending on actual plant conditions.

H. When the pressure control swaps from "HEADER PRESSURE CONTROL" to "REACTOR PRESSURE CONTROL" the pressure set will be actual Reactor pressure at the time the swap is done, regardless of any previously raised Reactor pressure set done during a Reactor startup.

3.7 Electrical Alignments and Load Considerations

A. Downpowering of Nuclear Units Under Low System Load Conditions:

   Due to having five nuclear units in an operating status, the frequency of downpowering units under low system load conditions is expected to rise. The following communications process will be used to coordinate downpowering a unit at BFN under low load conditions:

3.7 Electrical Alignments and Load Considerations (continued)

1. The Electrical System Operator (ESO) will anticipate the potential need to downpower nuclear units as far in advance as reasonable, normally one to two days. The ESO will inform the Operations Duty Specialist (ODS) of this potential need.

2. The ODS will notify the Browns Ferry Shift Manager that a potential need to downpower exists.

3. The Shift Manager will notify the Operations Superintendent who will notify the Operations Manager and Duty Plant Manager.

4. BFN will initiate a telecon with other operating nuclear units and senior nuclear corporate management (normally, Senior Vice President, Nuclear Operations, or, President, TVA Nuclear and Chief Nuclear Officer) to formulate a contingency plan. The plan will address which units are to be downpowered based on existing plant conditions, the reduction capability of each unit, time to reach reduced power as well as return to full power, and the preferred order for downpowering.[SFRC/C] Scram Reduction Recommendations G-20-1 and G-20-2 require an additional SRO licensed operator to assist with BOP operations when performing power maneuvers during unit startup or shut down with Reactor power less than 60%. This individual is not limited to the Control Room. [SFRC-17, G-20-1 & -2]

5. The contingency plan will be communicated to the appropriate site management and Shift Manager for the impacted units as well as the transmission/power supply organization.

6. The ESO will notify the designated Shift Managers approximately two to four hours before the need to actually downpower. The Shift Manager will notify the Operations Superintendent of any actual downpower.

7. Any change to unit status that would impact the agreed upon contingency plan will cause the telecon to be reconvened with all affected parties and a revised contingency plan developed. This will be initiated by the site management who identifies the need to revise the plan.

B. Electrical alignments and Bus loading are to be made in accordance with 0-OI-57A, Switchyard and 4160V Electrical System and 0-OI-57B, 480V/240V AC Electrical System.

3.8 Condensate and Feedwater

A. Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally shall be in direct communication with the Control Room. Evolutions resulting in changes in condensate/feedwater flow (condensate/booster pump start, feedwater pump start, changes in Reactor power, feedwater flow, steam flow, etc) will affect flow rates through 3-FCV-002-0190, steam-jet air-ejector condenser(s), steam packing exhauster condenser, and Off-Gas condenser. SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 should be maintained between 2 X 10^6 lbm/hr and 3 X 10^6 lbm/hr.

B. The following are the limitations on the Condensate system and the Reactor Feedwater Pumps during normal, steady-state operations:

1. Condensate System:

   a. [II/C] Condensate flow should always be maintained within the following limits, using 3-FC-2-29 in BAL if possible, to prevent Condensate Pump damage:

      (1) One Condensate Pump operation, greater than 1.5 X 10^6 lbm/hr but less than 6.25 x 10^6 lbm/hr.

      (2) Two Condensate Pump operation, greater than 3.0 X 10^6 lbm/hr but less than 12.5 x 10^6 lbm/hr.

      (3) Three Condensate Pump operation, greater than 4.5 X 10^6 lbm/hr but less than 15.0 x 10^6 lbm/hr. [II-B-91-158]

   b. Normal maximum line current to Condensate Pump Motors should not exceed 118 amps steady-state operations.

   c. Normal maximum line current to Condensate Booster Pump Motors should not exceed 225 amps steady-state operations.

2. Reactor Feedwater Pumps:

   a. Individual Reactor Feedpump speed should be less than 5050 RPM.

C. RFW START-UP LCV, 3-LCV-3-53, does not have a hole in the disc allowing flow at low pressures. The valve does have relief ports that may allow a small amount of water to pass. The flow should not be of significant amount, but 3-FCV-3-53 may be isolated at the Unit Supervisors discretion.

3.9 Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs) [SOER 01-1, BFN PER 126211, PER 961778, PER 116666]

A. This General Operating Instruction initiates processes that are likely to cause increased radiation levels that raise the risk of unintended radiological exposures and also radiation levels that warrant High Radiation Area or Locked High Radiation Area Controls.

This GOI relies heavily on a multitude of System Operating Instructions (support procedures) for system alignments required for the various process systems. Many of these alignments can and do result in raising the radiological impacts for the areas affected by the alignments. Therefore, there are increased probabilities of unintended radiation exposures to personnel that may be occupying these areas when alignments take place, and when reactor power increases occur.

B. To reduce the probability of unintended radiation exposures, the following controls are established by this procedure:

1. Radiological Protection Hold Points (RPHPs) are pre-established at appropriate locations in this GOI and in the support procedures. The function of RPHPs is to allow Radiation Protection to help ensure no unintended radiological exposures occur as the result of a system configuration change or raising reactor power. This may require holding actions for a step (actions typically identified with a BEFORE conditional step) until verifying personnel are not in the area before continuing in the procedure. These RPHPs also allow a determination as to whether actions are required to implement RCI-17, Control of High Radiation Areas and Very High Radiation Areas, controls.

2. The Radiation Protection notification steps have an (R) placed in the step initial line, which means these steps can <u>NOT</u> be omitted unless the action associated with the step is not performed, or the step allows the notification to be N/A'd as determined by the Unit Supervisor.

3. An Appendix (Appendix A, Radiation Protection Notification Record) is provided to record Radiation Protection notifications, RPHPs, and release of RPHPs, as necessary. The instructions for Appendix A is used to identify the appropriate required logging of Radiological Protection entries. The primary function of the appendix is to ensure proper communication with Radiation Protection personnel and that they are allowed sufficient opportunity to implement needed radiological controls.

4. Radiation Protection notification steps that require a RPHP are clearly worded that an RPHP is in effect. For these steps, it should be made clear to Radiation Protection that an RPHP is in effect so that they understand that a signature on Appendix A will be necessary.

3.9 Radiation Protection Notifications and Radiological Protection Hold Points (RPHPs) [SOER 01-1, BFN PER 126211, PER 961778, PER 116666] (continued)

Radiation Protection notification steps that are not identified as RPHP steps are considered courtesy notification steps to Radiation Protection. These steps serve the purpose of informing Radiation Protection of evolutions that are about to be implemented that may impact plant radiological conditions and allow them to respond or "get their ducks in a row". None of these steps imply that a hold in the procedure is necessary unless Radiation Protection identifies one may be necessary at some point after the notification is made. In many cases, the courtesy notifications are related to an RPHP notification that will be reached later in the procedure.

These courtesy steps may also inform Radiation Protection that a system has been returned to normal, has been shutdown, or a pump that was previously started, is now shutdown. This information may be useful to Radiation Protection for determining if area surveys should be performed due to changing radiological conditions in an area. The courtesy notification steps generally require an entry of the notification in the NOMS narrative log, but may or may not require Appendix A entry by operations, depending upon expected radiological impact of the associated evolution(s).

C. If, at any time while performing this procedure, or while performing a support procedure, Radiation Protection personnel, Unit Operator, Unit Supervisor, or other knowledgeable shift member identifies the need for a RPHP, then the following is performed:

1. "RPHP" is written to the left of the affected procedure step number (this GOI or the support procedure). If the RPHP is identified for a support procedure, then RPHP is placed to the left of the step in this GOI that initiates the support procedure.

2. The appropriate notifications made to Radiation Protection personnel, as necessary.

3. The instructions for Appendix A are to be used to identify the appropriate required logging of Radiation Protection entries.

D. Removal of any <u>Radiation Control Notification</u> from this procedure requires Operations Management and Radiation Protection Management approval unless the action(s) related to the notification is also removed.

Removal or addition of any <u>procedure actions</u> that require Radiation Protection notification requires that Radiation Protection be notified.

4.0 PREREQUISITES

4.1 Prestartup Checklist

NOTES

1) [NEC/C] The steps in Section 4.0 are not required to be performed in sequence.
2) Those steps preceded by an (R) are required for all startups and can not be omitted unless provided for in the step.
3) Those steps not preceded by an (R) may be signed off as NA and initialed by the Unit Supervisor, as appropriate. [NRC IR 84-45]
4) For return to full power from power reduction, provided that the Reactor remains in RUN, it is not necessary to sign off any steps prior to where power reduction ceased and power escalation begins. Under these conditions, Section 4.0 may be N/A in part or all, at the Unit Supervisor discretion

[1] **VERIFY** REACTOR MODE SWITCH, 3-HS-99-5A-S1 in SHUTDOWN or REFUEL, key removed, and under Shift Manager control.
REFER TO Tech Spec 3.3.1.1 and 3.10.2.

(R) ____________ ____________ ____________
Initials Time Date

[1.1] [NER/C] **CHECK** REACTOR MODE SWITCH, 3-HS-99-5A-S1 for "LOOSENESS". (There should be **NO** movement between the handle lever casting and the lock cylinder.) [GE SIL 498]

(R) ____________ ____________ ____________
Initials Time Date

NOTES

1) The bottom layer of Reactor Well shield blocks must be in place prior to exiting mode 4 (cold shutdown).
2) Both Reactor Well Shield Blocks must be in place prior to entering Mode 2. [BFNPER 01-005145-000]

[2] **VERIFY** bottom layer of Reactor Well shield blocks are installed, or preparations in progress for installing the bottom layer of Reactor Well shield blocks. (N/A if shield blocks were not removed during shutdown)

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[3] **VERIFY** Main Steam System in Prestartup/Standby Readiness. REFER TO 3-OI-1.

(R) ____________ ____________ ____________
Initials Time Date

[4] **VERIFY** Condensate System in service in accordance with 3-OI-2.

____________ ____________ ____________
Initials Date Time

[5] **VERIFY** Condensate Demineralizer System in service with a minimum of three demineralizers in service. REFER TO 3-OI-2A.[PER 113186]

____________ ____________ ____________
Initials Date Time

[6] **VERIFY** CDE ammonia ≤ 0.5 ppb or actions in progress to bring it within limits (i.e., backwash/ precoat condensate demins with CG-12H).

____________ ____________ ____________
Initials Date Time
Chemistry

[7] **VERIFY** Condensate Storage and Transfer System in service and not Cross tied. REFER TO 0-OI-2B.[PER 02-004990-00]

____________ ____________ ____________
Initials Date Time

[8] **VERIFY** Demineralized Water System in service. REFER TO 0-OI-2C.

____________ ____________ ____________
Initials Date Time

[9] **VERIFY** Feedwater System in Prestartup/Standby Readiness, or in a configuration to support initial plant startup, with available Reactor Feedpumps on their turning gear, if turning gear available. REFER TO 3-OI-3.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[10] **VERIFY** Feedwater Heating and Misc Drains System in Prestartup/Standby Readiness. REFER TO 3-OI-6.

_______ Initials _______ Date _______ Time

NOTE
Due to the higher radiation levels for a startup following a shutdown where Noble Metals was injected, Hydrogen Water Chemistry should be placed in service as determined by analysis.

[11] **VERIFY** Hydrogen Water Chemistry System in Prestartup/Standby Readiness. REFER TO 3-OI-4. (N/A if System is unavailable or not required)

_______ Initials _______ Date _______ Time

[12] **VERIFY** Auxiliary Boilers in Prestartup/Standby Readiness. REFER TO 0-OI-12.

_______ Initials _______ Date _______ Time

[13] **VERIFY** Building Heating System in Prestartup/Standby Readiness. REFER TO 0-OI-44. (May be in service as weather conditions require.)

_______ Initials _______ Date _______ Time

[14] **VERIFY** Fuel Oil System in Prestartup/Standby Readiness. REFER TO 0-OI-18.

_______ Initials _______ Date _______ Time

[15] **VERIFY** Central Lubricating Oil System in Prestartup/Standby Readiness. REFER TO 0-OI-20.

_______ Initials _______ Date _______ Time

4.1 Prestartup Checklist (continued)

[16] **VERIFY** RHRSW System in Prestartup/Standby Readiness. REFER TO 0-OI-23. (May be in service as required for Shutdown Cooling or Torus Cooling.)

(R) ______ Initials ______ Time ______ Date

[17] **VERIFY** Raw Cooling Water System in service. REFER TO 3-OI-24.

(R) ______ Initials ______ Time ______ Date

[18] **VERIFY** Raw Service Water System in service with pumps in AUTO or with High Pressure Fire Pump(s) in service. REFER TO 0-OI-25.

(R) ______ Initials ______ Time ______ Date

[19] **VERIFY** High Pressure Fire Protection System in service. REFER TO 0-OI-26.

(R) ______ Initials ______ Time ______ Date

[20] **VERIFY** Condenser Circulating Water System in service with at least two pumps running. REFER TO 3-OI-27.

______ Initials ______ Date ______ Time

[21] **VERIFY** Screen Wash System in Prestartup/Standby Readiness. REFER TO 3-OI-27A.

______ Initials ______ Date ______ Time

[22] **VERIFY** Amertap System in Prestartup/Standby Readiness. REFER TO 3-OI-27B.

______ Initials ______ Date ______ Time

4.1 Prestartup Checklist (continued)

[23] **VERIFY** Cooling Towers in Prestartup/Standby Readiness. REFER TO 0-OI-27C. (N/A if not required.)

_________ _________ _________
Initials Date Time

[24] **VERIFY** Refueling Floor Ventilation in service. REFER TO 3-OI-30A.

(R) _________ _________ _________
Initials Time Date

[25] **VERIFY** Reactor Building Ventilation in service. REFER TO 3-OI-3OB.

(R) _________ _________ _________
Initials Time Date

[26] **VERIFY** Turbine Building Ventilation in service. REFER TO 3-OI-30C.

(R) _________ _________ _________
Initials Time Date

[27] **VERIFY** Radwaste Building Ventilation in service. REFER TO 0-OI-30D.

(R) _________ _________ _________
Initials Time Date

[28] **VERIFY** Service Building Ventilation in service. REFER TO 0-OI-30E.

_________ _________ _________
Initials Date Time

[29] **VERIFY** Common and DG Building Ventilation in service. REFER TO 0-OI-30F.

(R) _________ _________ _________
Initials Time Date

[30] **VERIFY** Control Bay Ventilation in service. REFER TO 0-OI-31.

(R) _________ _________ _________
Initials Time Date

[31] **VERIFY** Control Air System in service. REFER TO 0-OI-32.

(R) _________ _________ _________
Initials Time Date

4.1 Prestartup Checklist (continued)

[32] **VERIFY** Drywell Control Air in service. REFER TO 3-OI-32A.

(R) ____________ ____________ ____________
Initials Time Date

[33] **VERIFY** Service Air System in service. REFER TO 0-OI-33.

____________ ____________ ____________
Initials Date Time

[34] **VERIFY** Vacuum Priming System in service. REFER TO 0-OI-34.

____________ ____________ ____________
Initials Date Time

[35] **VERIFY** Generator Hydrogen System in service with hydrogen concentration greater than 90% and pressure between 30 and 60 psig, or in a configuration to support initial plant startup. REFER TO 3-OI-35.

____________ ____________ ____________
Initials Date Time

[36] **VERIFY** Stator Cooling System in service, or in a configuration to support initial plant startup. REFER TO 3-OI-35A.

____________ ____________ ____________
Initials Date Time

[37] **VERIFY** Seal Oil System in service, or in a configuration to support initial plant startup. REFER TO 3-OI-35B.

____________ ____________ ____________
Initials Date Time

[38] **VERIFY** Generator Circuit Breaker cycled, if required. REFER TO 3-OI-35C.

____________ ____________ ____________
Initials Date Time

[39] **VERIFY** CO_2 System in Prestartup/Standby Readiness. REFER TO 0-OI-39.

(R) ____________ ____________ ____________
Initials Time Date

4.1 **Prestartup Checklist (continued)**

[40] **VERIFY** Station Drainage System in service. REFER TO 0-OI-40.

Initials ________ Date ________ Time ________

[41] **VERIFY** EHC System in service. REFER TO 3-OI-47A.

Initials ________ Date ________ Time ________

[42] **VERIFY** Integrated Computer System in service. REFER TO 0-OI-48.

(R) Initials ________ Time ________ Date ________

[43] **VERIFY** Demineralizer Backwash Air System in Prestartup/Standby Readiness. REFER TO 0-OI-53.

Initials ________ Date ________ Time ________

[44] **VERIFY** The Common Station Service Transformers (CSST) A and B Load Tap Changers are in AUTO.

Initials ________ Date ________ Time ________

[45] **VERIFY** The 161kV Switchyard Capacitor Banks are aligned as required. REFER TO 0-OI-57A.

Initials ________ Date ________ Time ________

[46] **VERIFY** Switchyard and 4160V AC Electrical System in service. REFER TO 0-OI-57A.

(R) Initials ________ Time ________ Date ________

[47] **VERIFY** 480V/240V AC Electrical System in service. REFER TO 0-OI-57B.

(R) Initials ________ Time ________ Date ________

4.1 Prestartup Checklist (continued)

[48] **VERIFY** 208V/120V AC Electrical System in service. REFER TO 0-OI-57C.

(R) ____________ ____________ ____________
Initials Time Date

[49] **VERIFY** Auxiliary Electrical DC Distribution in service. REFER TO 0-OI-57D.

(R) ____________ ____________ ____________
Initials Time Date

[50] **VERIFY** Standby Liquid Control System in Prestartup/Standby Readiness. REFER TO 3-OI-63.

(R) ____________ ____________ ____________
Initials Time Date

[51] **VERIFY** RHR Loops I & II and Core Spray Loops I & II charged above TRM 3.5.4 Limits. REFER TO 3-OI-74 and 3-OI-75.

(R) ____________ ____________ ____________
Initials Time Date

NOTE
In order to prevent having to resample primary containment for a subsequent entry, Primary Containment Purge and/or Ventilation should remain in service until secured by the inerting process.

[52] [TSAR/C] **VERIFY** Primary Containment System in Prestartup/Standby Readiness with Drywell Coolers in service, except as specified in the note above. REFER TO 3-OI-64. [Item D48]

(R) ____________ ____________ ____________
Initials Time Date

[53] [TSAR/C] **VERIFY** Standby Gas Treatment System in Prestartup/Standby Readiness. REFER TO 0-OI-65. [Item D48]

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[54] **VERIFY** Off Gas System in Prestartup/Standby Readiness, or in a configuration to support initial plant startup. REFER TO 3-OI-66.

_____________ _____________ _____________
Initials Date Time

[55] **VERIFY** Emergency Equipment Cooling Water in Prestartup/Standby Readiness, or running. REFER TO 0-OI-67.

(R) _____________ _____________ _____________
Initials Time Date

[56] **VERIFY** Reactor Recirculation System in Prestartup/Standby Readiness or available pumps running as desired. REFER TO 3-OI-68.

(R) _____________ _____________ _____________
Initials Time Date

[57] **IF** Recirc System is in Single Loop Operation, **THEN**

NOTIFY Reactor Engineer to **VERIFY** 3-SR-3.4.1(SLO) Reactor Recirculation System Single Loop Operation is performed. (Otherwise N/A)

(R) _____________ _____________ _____________
Initials Time Date
Reactor Engineer

[58] **IF** a Recirculation Pump is in service, **THEN**

COMMENCE lowering Reactor water level to obtain normal water level band.

_____________ _____________ _____________
Initials Date Time

[59] **VERIFY** Reactor Water Cleanup System in service with at least one pump and one demineralizer in operation. REFER TO 3-OI-69.

(R) _____________ _____________ _____________
Initials Time Date

[60] **VERIFY** Reactor Building Closed Cooling Water System in operation. REFER TO 3-OI-70.

(R) _____________ _____________ _____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[61] **VERIFY** Residual Heat Removal System in Prestartup/Standby Readiness, except that one loop of RHR may be in Torus Cooling or Shutdown Cooling Mode. REFER TO 3-OI-74.

(R) ____________ ____________ ____________
Initials Time Date

[62] **VERIFY** Unit 2 Residual Heat Removal System, RHR Loop II prerequisites for Unit 3 restart, in Prestartup/Standby Readiness. REFER TO 2-OI-74.

(R) ____________ ____________ ____________
Initials Time Date

[63] **VERIFY** Core Spray System in Prestartup/Standby Readiness. REFER TO 3-OI-75.

(R) ____________ ____________ ____________
Initials Time Date

NOTE

Step 4.1[64] may be marked N/A if Drywell entry at pressure is planned or the H_2O_2 Analyzers are to be kept in Standby.

[64] **VERIFY** Containment Inerting System in Prestartup/Standby Readiness with H_2O_2 Analyzers in service. REFER TO 3-OI-76.

(R) ____________ ____________ ____________
Initials Time Date

[65] **VERIFY** Radwaste System in Prestartup/Standby Readiness and ready to receive water. REFER TO 0-OI-77A through 0-OI-77D.

(R) ____________ ____________ ____________
Initials Time Date

[66] **VERIFY** Fuel Pool Cooling System in service. REFER TO 3-OI-78.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[67] **VERIFY** Units 1/2 Standby Diesel Generators A,B,C and D in Prestartup/Standby Readiness. REFER TO 0-OI-82.

(R) ____________ ____________ ____________
Initials Time Date

[68] **VERIFY** Unit 3 Standby Diesel Generators 3A, 3B, 3C, and 3D in Prestartup/Standby Readiness. REFER TO 3-OI-82.

(R) ____________ ____________ ____________
Initials Time Date

NOTE
Step 4.1[69] may be marked N/A if Drywell entry at power is planned.

[69] **VERIFY** Containment Atmospheric Dilution System in Prestartup/Standby Readiness. REFER TO 3-OI-84.

(R) ____________ ____________ ____________
Initials Time Date

[70] **VERIFY** Control Rod Drive System in service with all rods inserted and suction from the normal source (CSTs) REFER TO 3-OI-85.[PER 02-004990-000]

(R) ____________ ____________ ____________
Initials Time Date

[71] **VERIFY** Rod Worth Minimizer in service except as allowed by Tech Specs. REFER TO 3-OI-85.

(R) ____________ ____________ ____________
Initials Time Date

[72] **VERIFY** Radiation Monitoring Systems in service. REFER TO 3-OI-90.

(R) ____________ ____________ ____________
Initials Time Date

[73] **VERIFY** Source Range Monitoring System in service. REFER TO 3-OI-92.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[74] **VERIFY** Intermediate Range Monitoring System in service. REFER TO 3-OI-92A.

(R) ____________ ____________ ____________
Initials Time Date

[75] **VERIFY** Average Power Range Monitoring System in service. REFER TO 3-OI-92B.

(R) ____________ ____________ ____________
Initials Time Date

NOTE
Step 4.1[76] may be marked N/A if Drywell entry at power is planned.

[76] **VERIFY** Traversing Incore Probe System in Prestartup/Standby Readiness. REFER TO 3-OI-94.

(R) ____________ ____________ ____________
Initials Time Date

[77] **VERIFY** Reactor Protection System in service. REFER TO 3-OI-99.

(R) ____________ ____________ ____________
Initials Time Date

[78] **VERIFY** Unit 2 Standby Liquid Control System in Prestartup/Standby Readiness (Storage Tank available to be aligned as an alternate source of injection for Unit 3). REFER TO 2-OI-63.

____________ ____________ ____________
Initials Date Time

4.1 Prestartup Checklist (continued)

[79] [NRC/C] **VERIFY** Reactor Coolant Temperature to the right of Curve #3 of Tech Specs Figure 3.4.9-1 for the following instruments: [IR 85-15]

A. Either of the following Recirc Pump 3A Temperatures: (N/A if OOS and Recirc System in Single Loop Operation.)

- RECIRC PUMPS DISCH TEMP PMP-3A (red pen), 3-TR-68-2 on Panel 3-9-4 or ICS (N/A if pump is OOS and Recirc System in Single Loop Operation.).
- RECIRC PMP A SUCT TEMP 68-6A on ICS.

(R) ____________ ____________ ____________
Initials Time Date

B. Either of the following Recirc Pump 3B Temperatures: (N/A if OOS and Recirc System in Single Loop Operation.)

- RECIRC PUMPS DISCH TEMP PMP-3B (green pen), 3-TR-68-2 on Panel 3-9-4 or ICS.
- RECIRC PMP B SUCT TEMP 68-83A on ICS.

(R) ____________ ____________ ____________
Initials Time Date

C. REACTOR VESSEL METAL TEMPERATURE, 3-TR-56-4, on Panel 3-9-47.

- RX VESSEL FLANGE, TE-56-7. ☐
- RX VESSEL FLANGE DR LINE, TE-56-8. ☐
- RX VESSEL BOTTOM HEAD, TE-56-29. ☐

(R) ____________ ____________ ____________
Initials Time Date

[80] **VERIFY** Reactor vessel head in place and bolts torqued in accordance with MSI-0-001-VSL001. (N/A if Reactor vessel head was not removed during shutdown.)

(R) ____________ ____________ ____________
Initials Time Date
Mech Maintenance

4.1 Prestartup Checklist (continued)

NOTE
For a short outage where hatches or air lock have not been opened, Step 4.1[81] through Step 4.1[84] may be marked N/A by Shift Manager or Maintenance Foreman.

[81] **VERIFY** Control Rod Drive Housing Support System locked in place and inspected by Maintenance Foreman prior to exceeding 1% RTP, **OR** prior to Reactor pressure greater than atmospheric pressure, per TRM 3.1.1.

(R) ____________ ____________ ____________
Initials Time Date
Mech Maintenance

[82] **VERIFY** All equipment hatches installed with trolley cranks chained and locked in accordance with MSI-0-064-HLT002.

(R) ____________ ____________ ____________
Initials Time Date
Mech Maintenance

NOTE
Step 4.1[83] may be marked N/A by Shift Manager or responsible Section if the Access door seal has not been broken (i.e., doors opened) and the 50.6 psig test is within its periodicity. The 50.6 psig test is required once every 30 months.

[83] **CHECK** the following prior to Drywell Close out: (Otherwise N/A)

- DWCA FLOW ELEMENT HEADER A, 3-FIQ -032-0092 (Rx Bldg El 565') reads less than 1.7 CFM.

____________ ____________ ____________
Initials Date Time

- DWCA FLOW ELEMENT HEADER B, 3-FIQ -032-0075 (Rx Bldg El 565') reads less than 1.7 CFM.

____________ ____________ ____________
Initials Date Time

4.1 Prestartup Checklist (continued)

[83.1] **IF** either Flow Meter reads above 1.7 CFM, **THEN**:

INITIATE Work Orders to identify and repair source of leakage.

________ Initials ________ Date ________ Time

[83.2] **VERIFY** Drywell personnel air lock closed and interlocks have been re-established, and tested per 3-SR-3.6.1.2.2. [BFPER 03-012038-000]

(R) ________ Initials ________ Time ________ Date

NOTES

1) [NRC/C] Work Control is the required organization for Surveillance completion signoffs. [LER 259/93001]
2) When Containment integrity is required, airlock door seals should be tested within 7 days after each containment access (0-TI-360, Appendix A may be referenced).

[84] **VERIFY** Drywell personnel air lock has been leak tested in accordance with 3-SR-3.6.1.2.1 as required by the Containment Leak Rate Program. [BFPER 03-012038-000]

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

[85] **VERIFY** Drywell integrity established in accordance with MSI-0-001-VSL001. (N/A if not initial startup following a refueling outage.)

(R) ________ Initials ________ Time ________ Date
Mech Maintenance

[86] **VERIFY** Pressure Suppression Chamber water level between -2 inches and -5.5 inches on SUPPR POOL WATER LEVEL, 3-LI-64-66 and/or SUPPR POOL WATER LEVEL, 3-LI-64-54A, on Panel 3-9-3.

(R) ________ Initials ________ Time ________ Date

4.1 Prestartup Checklist (continued)

[87] **VERIFY** Drywell Floor and Equipment Drain sumps pumped down from Panel 3-9-4.

(R) ______ ______ ______
Initials Time Date

[88] [NRC/C] **VERIFY** Refuel Floor equipment hatch cover has at least one Panel removed. [LER 259/85018]

(R) ______ ______ ______
Initials Time Date

[89] **VERIFY** The following surveillance's completed or current (Surveillances not required, may be marked N/A):

A. 3-SI-3.3.1.A.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

B. 3-SI-4.7.A.5.c.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

C. 3-SR-3.1.3.5(B).

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

D. 3-SR-3.1.4.1.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

E. 3-SR-3.1.8.1.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

4.1 Prestartup Checklist (continued)

F. 3-SR-3.1.8.2.

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

G. 3-SR-3.3.1.1.3(IRMs).

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

H. 3-SR-3.3.1.1.9(IRM A-H).

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

I. 3-SR-3.3.1.1.12.

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

J. 3-SR-3.3.1.1.13(APRM-1) or 3-SR-3.3.1.1.16(APRM-1).

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

K. 3-SR-3.3.1.1.13(APRM-2) or 3-SR-3.3.1.1.16(APRM-2).

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

L. 3-SR-3.3.1.1.13(APRM-3) or 3-SR-3.3.1.1.16(APRM-3).

(R) ________ Initials ________ Time ________ Date
[NRC/C] Work Control

4.1 Prestartup Checklist (continued)

M. 3-SR-3.3.1.1.13(APRM-4) or 3-SR-3.3.1.1.16(APRM-4).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

N. 3-SR-3.3.1.1.14(2e).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

O. 3-SR-3.3.2.2.4. (N/A step if surveillance will be performed with unit on line prior to reaching 25% power)

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

P. 3-SR-3.3.5.1.6(ADS A).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

Q. 3-SR-3.3.5.1.6(ADS B).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

R. 3-SR-3.4.1(SLO).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

S. 3-SR-3.4.1(DLO).

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

4.1 Prestartup Checklist (continued)

T. 3-SR-3.5.1.2(RHR I) Completed for Modes 1-5.

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

U. 3-SR-3.5.1.2(RHR II) Completed for Modes 1-5.

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

V. 3-SR-3.5.1.5.

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

W. 3-SR-3.5.1.10.

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

X. 3-SR-3.6.1.3.3.

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

Y. 3-SR-3.6.1.3.5(SD).

(R) ________ ________ ________
Initials Time Date
[NRC/C] Work Control

4.1 Prestartup Checklist (continued)

Z. 3-SR-3.7.5.2.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

AA. 3-SR-3.7.5.3.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

[90] **VERIFY** Primary and Secondary Containment Integrity established REFER TO Technical Specifications 3.6.1.1 and 3.6.4.1.

(R) ______ ______ ______
Initials Time Date

[91] **VERIFY** a minimum of 15 feet of water on CST 3 LEVEL, 3-LI-2-165A, on Panel 3-9-6.

(R) ______ ______ ______
Initials Time Date

[92] **When** either of the following has been performed:

- Maintenance on the TIP System,

OR

- Work under the Reactor vessel,

THEN

VERIFY Exercising TIP Drives for Reactor startup complete. REFER TO 3-OI-94. (N/A if not a maintenance outage.)

______ ______ ______
Initials Date Time

[93] **VERIFY** Preparing Source Range Monitors for Reactor startup complete. REFER TO 3-OI-92.

(R) ______ ______ ______
Initials Time Date

4.1 Prestartup Checklist (continued)

[94] **VERIFY** Preparing Intermediate Range Monitors for Reactor startup complete. REFER TO 3-OI-92A.

(R) ____________ ____________ ____________
Initials Time Date

[95] **VERIFY** IRM recorders high alarm setpoint programmed ON with setpoint at 75.

(R) ____________ ____________ ____________
Initials Time Date
IM

[96] **VERIFY** Offsite power available in accordance with indications available on electrical switchboard Panel 3-9-23. REFER TO Tech Specs 3.8.

(R) ____________ ____________ ____________
Initials Time Date

[97] **CHECK** the following on Panel 3-9-5:

- High Reactor Water Level Trip Channels A and B are energized and reset by observing red lights extinguished and green lights illuminated: (N/A, if Shutdown Cooling is in service).

  RX WTR LVL CH A HI RFPT/MT TRIP RESET, 3-HS-3-208A. ☐

  RX WTR LVL CHB HI RFPT/MT TRIP RESET, 3-HS-3-208B. ☐

(R) ____________ ____________ ____________
Initials Time Date

- Reactor Pressure, Level, Steam Flow, and Feed Flow instrument failures (indicated by yellow instrument readout) are not present or associated instrument inputs are inhibited on Panel 3-9-5 or locally at the computer.

(R) ____________ ____________ ____________
Initials Time Date

- Backlight for SINGLE ELEMENT push-button, 3-HS-46-6/1, on Panel 3-9-5, is illuminated and backlight for THREE ELEMENT push-button, 3-HS-46-6/3 is extinguished.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[98] **CHECK** BOTH Standby Liquid Control System SQUIB VALVE A and B CONTINUITY blue lights illuminated on Panel 3-9-5:

- 3-ZI-63-8A. ☐
- 3-ZI-63-8B. ☐

(R) ________ ________ ________
Initials Time Date

[99] **VERIFY** the following red lights illuminated to ensure Main Steam Isolation Logic reset on Panel 3-9-4:

- MSIV GROUP A1, 3-IL-64-A1. ☐
- MSIV GROUP B1, 3-IL-64-B1. ☐
- MSIV GROUP A2, 3-IL-64-A2. ☐
- MSIV GROUP B2, 3-IL-64-B2. ☐

(R) ________ ________ ________
Initials Time Date

[100] **VERIFY** SCRAM SOLENOID GROUP A and B LOGIC RESET lights illuminated on Panel 3-9-5.

(R) ________ ________ ________
Initials Time Date

[101] **VERIFY** the following BACKUP SCRAM VALVE lights illuminated on Panel 3-9-5:

- SYSTEM A BACKUP SCRAM VALVE, 3-IL-99-5A/AB. ☐
- SYSTEM B BACKUP SCRAM VALVE, 3-IL-99-5A/CD. ☐

(R) ________ ________ ________
Initials Time Date

[102] **OBTAIN** proper withdrawal sequence from Reactor Engineer.

(R) ________ ________ ________
Initials Time Date
Reactor Engineer

4.1 Prestartup Checklist (continued)

NOTES
1) Step 4.1[104] through Step 4.1[107]are performed on Panel 3-9-3.
2) Only one (1) Main Steam Line may have its Isolation valve(s) closed in Step 4.1[103].

[103] **IF** a Main Steam Line Isolation Valve is INOP and cannot be opened. **THEN**

MARK the associated Main Steam Isolation Valve and the Inline valve as N/A on the following steps: (Otherwise N/A)

- Step 4.1[104]

_______ _______ _______
Initials Date Time

- Step 4.1[105]

_______ _______ _______
Initials Date Time

- Step 5.0[42.4]

_______ _______ _______
Initials Date Time

[104] **IF** Reactor Coolant Temperature indicates ≤ 215°F, **AND**, Reactor pressure indicates ≤ 0 psig,**THEN**

VERIFY the following Outboard Main Steam Isolation Valves indicate CLOSED: (Otherwise N/A)

- 3-FCV-1-15 using MSIV LINE A OUTBOARD, 3-HS-1-15A.

(R) _______ _______ _______
Initials Time Date

4.1 Prestartup Checklist (continued)

- 3-FCV-1-27 using MSIV LINE B OUTBOARD, 3-HS-1-27A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-38 using MSIV LINE C OUTBOARD,
3-HS-1-38A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-52 using MSIV LINE D OUTBOARD,
3-HS-1-52A.

(R) ____________ ____________ ____________
Initials Time Date

[105] **VERIFY** the following Inboard Main Steam Isolation Valves indicate OPEN: (N/A, if desired to establish Hot Standby conditions during power ascension.)

- 3-FCV-1-14 using MSIV LINE A INBOARD, 3-HS-1-14A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-26 using MSIV LINE B INBOARD, 3-HS-1-26A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-37 using MSIV LINE C INBOARD, 3-HS-1-37A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-51 using MSIV LINE D INBOARD, 3-HS-1-51A.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[106] **IF** Reactor Coolant Temperature indicates ≤ 210°F, **THEN**

VERIFY the following Reactor Head Vents indicate OPEN: (Otherwise N/A)

- 3-FCV-3-98 using RPV HEAD VENT INBD VALVE, 3-HS-3-98A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-3-99 using RPV HEAD VENT OUTBD VALVE, 3-HS-3-99A.

(R) ____________ ____________ ____________
Initials Time Date

[107] **IF** Reactor Coolant Temperature indicates ≤ 210°F, **THEN**

VERIFY the following Main Steam Line drain valves indicate closed: (Otherwise N/A)

- 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A.

(R) ____________ ____________ ____________
Initials Time Date

- 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A.

(R) ____________ ____________ ____________
Initials Time Date

[108] **IF** Reactor is in MODE 4, **THEN**

VERIFY EHC SETPOINT, 3-PI-47-162 is set at 150 psig. (N/A if not in MODE 4)

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[109] **VERIFY** REACTOR MODE SWITCH, 1-HS-99-5A-S1 position:

SHUTDOWN ________ ($\sqrt{}$) REFUEL ________ ($\sqrt{}$)

(R) ________ Initials ________ Time ________ Date

[110] **VERIFY** All control rods full-in as indicated by "00", green backlights ("full-in") illuminated, or Control Rod Position Log.

(R) ________ Initials ________ Time ________ Date

[111] **VERIFY** CRD POWER, 3-HS-85-46, in ON.

(R) ________ Initials ________ Time ________ Date

NOTE
If more than 24 hours elapse between performance of Step 4.1[112] through Step 4.1[127] and the beginning of Section 5.0, another review of these steps is advised to ensure they are still current.

[112] **VERIFY** All Mechanical maintenance necessary to initiate unit startup complete.

(R) ________ Initials ________ Time ________ Date
Mech Maintenance

[113] **VERIFY** All Electrical maintenance necessary to initiate unit startup complete.

(R) ________ Initials ________ Time ________ Date
Elec Maintenance

[114] **VERIFY** All I&C maintenance necessary to initiate unit startup complete.

(R) ________ Initials ________ Time ________ Date
I&C Maintenance

4.1 Prestartup Checklist (continued)

[115] **VERIFY** All Technical Support procedures necessary for unit startup complete.

(R) ______ ______ ______
Initials Time Date
Site Engineering

[116] **VERIFY** Temporary Shielding removed. REFER TO RCI-15.2, Temporary Shielding.

(R) ______ ______ ______
Initials Time Date
Radiation Protection

[117] **VERIFY** All Operations surveillances necessary for unit startup complete.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

(R) ______ ______ ______
Initials Time Date
Unit Supervisor/SRO

[118] **VERIFY** All surveillance's necessary for unit startup complete.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

[119] **VERIFY** applicable portions of 3-SR-3.3.1.2.5&6 complete if not performed within the last 7 days.

(R) ______ ______ ______
Initials Time Date
[NRC/C] Work Control

[120] **VERIFY** CRD TEST HOIST EQUIPMENT HANDLING PLATFORM OUTLETS, breaker 1C is OFF on 480V Reactor MOV Board 3C.

(R) ______ ______ ______
Initials Time Date

4.1 Prestartup Checklist (continued)

[121] **VERIFY** no clearance or temporary alterations in effect that would prevent a unit startup.

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

[122] **VERIFY** no Tech Specs LCOs or Technical Requirements Manual LCO's in effect that would prevent unit startup or mode change.

(R) ____________ ____________ ____________
Initials Time Date
STA

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

[123] **IF** startup is following a Reactor scram, **THEN**

VERIFY complete BP-250, Restart Approval. (Otherwise N/A)

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

[124] **COMPLETE** Attachment 1 prior to exceeding 200°F to verify EQ doors in proper position.

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

[125] **VERIFY** Fire Protection Report, Volume 1, Appendix R Safe Shutdown Program, Section III reviewed for operability of required safe shutdown equipment or applicable compensatory measures implemented.

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

4.1 Prestartup Checklist (continued)

[126] **VERIFY** RPS shorting links installed. **REFER TO** 0-GOI-100-3A, Attachment 6 or 0-GOI-100-3C, Attachment 2. (N/A if removed following a refueling outage.)

_____________ _____________ _____________
Initials Time Date
Unit Supervisor

[127] **PERFORM** LAMP TEST for EHC Control System. **REFER TO** EHC Control System Lamp Test section in 3-OI-47.

_____________ _____________ _____________
Initials Time Date
Unit Supervisor

[128] **IF** desired by the Unit Supervisor, **THEN**

DISABLE the Feedwater Heater alarms from the AW-51 station and **LOG** in NOMS Narrative logs as a carryover item. (Otherwise N/A).

_____________ _____________ _____________
Initials Time Date
Unit Supervisor

[129] **IF** a mode change is anticipated in the next 12 hours, **THEN**

PERFORM the following; (Otherwise N/A)

[129.1] **OBTAIN** the required 3-SR-2 section for the mode being changed and **LOG** in the NOMS Narrative LOG. (i.e. Mode 1,2&3)

(R) _____________ _____________ _____________
Initials Time Date

[129.2] **VERIFY** the current mode 3-SR-2 Data is obtained. (i.e. Mode 4&5)

(R) _____________ _____________ _____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[130] **VERIFY** the following indicate between 40°F and 95°F at each point on Panel 3-9-3:

- SUPPRESSION POOL WATER TEMPERATURE indicator, 3-TR-64-161.

(R) ____________ ____________ ____________
Initials Time Date

- SUPPRESSION POOL WATER TEMPERATURE indicator, 3-TR-64-162.

(R) ____________ ____________ ____________
Initials Time Date

[131] **PERFORM** the following for IRMs on Panel 3-9-5:

[131.1] **VERIFY** the following IRM Range Switches are on range 1:

- CHANNEL A IRM RANGE SWITCH, 3-XS-92-7/42A. ☐
- CHANNEL C IRM RANGE SWITCH, 3-XS-92-7/42C. ☐
- CHANNEL E IRM RANGE SWITCH, 3-XS-92-7/42E. ☐
- CHANNEL G IRM RANGE SWITCH, 3-XS-92-7/42G. ☐
- CHANNEL B IRM RANGE SWITCH, 3-XS-92-7/42B. ☐
- CHANNEL D IRM RANGE SWITCH, 3-XS-92-7/42D. ☐
- CHANNEL F IRM RANGE SWITCH, 3-XS-92-7/42F. ☐
- CHANNEL H IRM RANGE SWITCH, 3-XS-92-7/42H. ☐

(R) ____________ ____________ ____________
Initials Date Time

4.1 Prestartup Checklist (continued)

[131.2] **VERIFY** all eight SELECT switches selected to IRM and recorders inking.

(R) ___________ ___________ ___________
Initials Date Time

[131.3] **RECORD** both IRM BYPASS, joystick positions. (N/A if not bypassed.)

- 3-HS-92-7A/S4A

___________ Channel(s) bypassed

- 3-HS-92-7A/S4B

___________ Channel(s) bypassed

(R) ___________ ___________ ___________
Initials Date Time

[131.4] **REQUEST** Reactor Engineering to initiate 3-SR-3.3.1.1.5, SRM and IRM Overlap Verification.

(R) ___________ ___________ ___________
Initials Time Date
Reactor Engineer

[131.5] **IF** control rod withdrawal for startup is expected to occur on the current shift, **THEN**

CONDUCT a pre-evolution briefing on Reactivity Management in accordance with SPP-10.4. (Otherwise N/A)

(R) ___________ ___________ ___________
Initials Time Date

4.1 Prestartup Checklist (continued)

NOTES

1) It may be necessary to momentarily place IRM range switches in Position 2 or 3 to verify downscale light illuminated.
2) If an IRM is in BYPASS its associated DNSCL light will not be lit.

[131.6] **VERIFY** all IRMs that are NOT bypassed, DNSCL lights illuminated.

(R) ____________ ____________ ____________
Initials Time Date

[131.7] **VERIFY** the following display lights for all eight IRMs are extinguished:

- HIGH HIGH OR INOP. ☐
- HIGH. ☐
- BYPASSED (Will be illuminated in bypassed channel.) ☐

(R) ____________ ____________ ____________
Initials Time Date

[132] **PERFORM** the following for APRMs on Panel 3-9-5:

[132.1] **RECORD** APRM BYPASS , 3-HS-92-7B/S3 joystick position. (N/A if not bypassed.)

Channel ____________ bypassed

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

[132.2] **VERIFY** the following display lights for all four APRMs are as follows:

- HIGH OR INOP lights extinguished. ☐
- UPSCALE lights extinguished. ☐
- DNSCL lights illuminated. ☐
- BYPASSED lights extinguished. (Will be illuminated in bypassed channel.) ☐

(R) ____________ ____________ ____________
Initials Time Date

[133] **PERFORM** the following for RBMs on Panel 3-9-5:

[133.1] **RECORD** RBM BYPASS, 3-HS-92-7B/S2 joystick position. (N/A if not bypassed.)

____________ Channel bypassed

(R) ____________ ____________ ____________
Initials Time Date

[133.2] **VERIFY** all RBM display lights extinguished. (BYPASS light will be illuminated < 25% power or in bypassed channel.)

(R) ____________ ____________ ____________
Initials Time Date

NOTE
Tech Specs limits plant to one startup per calendar year from all rods in with RWM inoperable.

[134] **VERIFY** RWM set to allow two insert errors (N/A if RWM not operable).

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[135] **CHECK** SRM count rate greater than 3 cps on at least three operable SRM channels.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

NOTE
The emergency rod insert function of the CRD NOTCH OVERRIDE, 3-HS-85-47 switch is considered operable if'00' indication is lost.

[136] [NRC/C] **VERIFY** operability of emergency rod insert function of CRD NOTCH OVERRIDE switch, 3-HS-85-47, by performing the following: [IE Bulletin 79-12]

[136.1] **SELECT** control rod.

(R) ____________ ____________ ____________
Initials Time Date

[136.2] **PLACE** and **HOLD** CRD NOTCH OVERRIDE, 3-HS-85-47 switch to EMERG ROD IN until SELECTED ROD position 00 display extinguishes, then **RELEASE**.

(R) ____________ ____________ ____________
Initials Time Date

[137] **PERFORM** Rod Drift Alarm Test using an insertion signal. REFER TO 3-OI-85.

(R) ____________ ____________ ____________
Initials Time Date

4.1 Prestartup Checklist (continued)

NOTES

1) When using auxiliary boilers to supply steam loads, the preferred method of inventory control is to blow down to Radwaste to preclude overfilling the CST's.
2) Steam Seal Header pressure may be reduced to as low as 1/2 psig provided the Turbine Generator is off line with auxiliary steam supplying the steam seals.
3) If Steam Seal pressure is being maintained at 1/2 psig to minimize water use during start up, prior to shifting Steam Seals to Main Steam ensure 3-PCV-1-147 is in Auto and Steam Seal Header pressure is between 2 1/2 psig and 5 1/2 psig.

[138] **IF** it is desired to:

- Establish Steam Seals to the Main Turbine and Reactor Feedpump Turbines,

AND

- Establish vacuum in Main Condenser using auxiliary steam,

THEN

PERFORM the following: (Otherwise N/A):

[138.1] **START** Auxiliary Boilers. REFER TO 0-OI-12.

________ Initials ________ Date ________ Time

[138.2] **ESTABLISH** sealing steam to Main Turbine and Reactor Feedpump Turbines. REFER TO 3-OI-47C.

________ Initials ________ Date ________ Time

CAUTION

Time to criticality should be carefully evaluated. The time SJAE's are on Aux Steam should be minimized to prevent filling the CST during startup.

[138.3] **ESTABLISH** Condenser vacuum. REFER TO 3-OI-66.

________ Initials ________ Date ________ Time

4.1 Prestartup Checklist (continued)

NOTE
Tech Specs requires, when less than 10% RTP, control rod pattern is verified to be in compliance with the BPWS by performing 3-SR-3.1.6.1 every 24 hours prior to entry into Mode 2.

[139] **VERIFY** Control Rod Pattern in Compliance with the BPWS per 3-SR-3.1.6.1 (N/A if greater than 10% RTP).

(R) ____________ ____________ ____________
Initials Time Date

NOTES
1) The bottom layer of Reactor Well shield blocks must be in place prior to exiting mode 4 (cold shutdown). 2) Both Reactor Well Shield Blocks must be in place prior to entering Mode 2. [BFNPER 01-005145-000]

[140] **VERIFY** both Reactor Well Shield Block layers installed.

____________ ____________ ____________
Initials Date Time

[141] **VERIFY** Control Rod Drive Housing Support System in place prior to exceeding 1% RTP or prior to Reactor pressure greater than atmospheric pressure. REFER TO TRM 3.1.1.

____________ ____________ ____________
Initials Date Time

4.1 Prestartup Checklist (continued)

	NAME (print)	INITIALS
Performed by:	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
	______________________	________
Reviewed by:	______________________	________
	Shift Manager	Date

4.1 Prestartup Checklist (continued)

REMARKS:

5.0 INSTRUCTION STEPS

NOTES
1) For return to full power from power reduction, provided that the Reactor remains in RUN, it is not necessary to sign off any steps prior to where power reduction ceased and power escalation will begin, EXCEPT for Step 5.0[2].
2) [NRC/C] Sequential completion is preferred in Section 5.0 unless the Unit Supervisor approves otherwise.
3) Steps 5.0[1] thru Step 5.0[9] must be completed as appropriate.
4) Steps beyond Step 5.0[9] may not be signed off until all steps proceeding Step 5.0[9] are signed or addressed as noted in the steps.
5) All steps and conditions shall be verified prior to any Mode or Condition changes, to ensure all tech specs are met.
6) Those steps preceded by an (R) are required for all startups and can not be omitted unless provided for in the step. [NRC IR 84-45]
7) Sections other than Operations have signoff responsibilities in this section. Early contact by Operations will minimize unnecessary delays.

[1] [NRC/C] **VERIFY** all Prerequisites listed in Section 4.0 are satisfied OR Actions are in progress to complete those steps prior to Step 5.0[9]. [IR 84-45]

(R) ____________ ____________ ____________
Initials Time Date

[2] **REVIEW** all Precautions and Limitations listed in Section 3.0.

(R) ____________ ____________ ____________
Initials Time Date

[3] **VERIFY** 0-TI-270, Refueling Test Program has been initiated and all appropriate signatures for Reactor startup have been obtained.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTION

1) When Reactor coolant temperature is less than 215°F, a maximum heatup rate limit of 50°F/hr will reduce the 0_2 and Hydrogen Peroxide content of the coolant.

2) During Reactor Heatup/Cooldown, the optimum rate is 20°F every 15 minutes. This will ensure the administrative limit of 90°F/Hr is not exceeded. Attempts to "makeup" for time intervals which fall short of 20°F SHALL not be made.. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines will assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is not exceeded.

3) During Reactor heatup, operators should use metal temperatures as a reminder that as metal heats up, the moderator HEATUP RATE will rise with the same amount of heat input.

NOTES

1) If RHR Shutdown Cooling is not in service, Step 5.0[4] sign-off signifies verification of Standby Readiness.

2) Attachment 2, Temperature Verifications From Cold Shutdown to 212°F, has requirement to be performed prior to reaching 210°F and 212°F. DECAY HEAT may cause Reactor coolant temperature rise above 212°F prior to reaching the Point of Adding Heat.

[4] **STOP** RHR Shutdown Cooling and **REALIGN** RHR System for Standby Readiness. REFER TO 3-OI-74.

(R) ______ ______ ______
Initials Time Date

[5] **MONITOR** Reactor temperature.

And

PERFORM Attachment 2, Temperature Verifications From Cold Shutdown to 210°F, while continuing in this procedure for Reactor startup.

______ ______ ______
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

NOTE

LEVEL A, 3-LI-3-58A and LEVEL B, 3-LI-3-58B normally indicate greater than +60 inches when Reactor temperature is less than 212°F.

[6] **CHECK** Reactor vessel water level between 28 inches and 38 inches on all the following level instruments on Panel 3-9-5:

A. LEVEL A, 3-LI-3-53.

(R) ____________ ____________ ____________
Initials Time Date

B. LEVEL B, 3-LI-3-60.

(R) ____________ ____________ ____________
Initials Time Date

C. LEVEL C, 3-LI-3-206.

(R) ____________ ____________ ____________
Initials Time Date

D. LEVEL D, 3-LI-3-253.

(R) ____________ ____________ ____________
Initials Time Date

E. RW LVL, 3-LT-3-53-60 (Red Pen) on RX VESSEL LEVEL/TOTAL FW FLOW, 3-XR-3-53.

(R) ____________ ____________ ____________
Initials Time Date

NOTE

If Reactor is started up in Single Loop Operation and the second Recirc Pump is started 3-SR-3.4.1(DLO) should be performed.

[7] **VERIFY** RUNNING or **START** Reactor Recirc Pump(s). REFER TO 3-OI-68.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[8] **VERIFY** the following in preparation for startup:

- Reactor Engineer is present in Control Room. ☐
- **IF** performing initial startup after a refueling outage, **THEN**

  PERFORM 3-SR-3.1.1.1, Reactivity Margin Test, prior to withdrawing control rods. ☐

(R) ____________ ____________ ____________
Initials Time Date

NOTES

1) Steps 5.0[9.2] and 5.0[9.3] are performed to ensure transition from mode 4 and 5 section of 3-SR-2 to modes 1, 2,3 section of 3-SR-2 and to ensure that all required data is obtained prior to mode change per LCO 3.0.4 and SR-3.0.4.
2) The previous shift data may be used if the data has been obtained within 12 hours and all data is verified, to allow mode changes.
3) Use of the previous shift's data to minimize startup delays does not preclude the shift from obtaining the required SR-2 data for the current shift following Mode Change. This should be performed soon as possible.

[9] **PERFORM** the following prior to entering Mode 2.

[9.1] **IF** any RPHPs were initiated by procedures used in Section 4.0 and are still in effect, **THEN**

VERIFY the RPHPs are closed out, OR Radiation Protection authorizes entering Mode 2 with the RPHP in place.

(R) ____________ ____________ ____________
Initials Date Time

[9.2] **VERIFY ALL OBTAINABLE** data for 3-SR-2 Modes 1, 2 and 3 sections is obtained.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

NOTE
The STA will perform Step 5.0[9.3].

[9.3] **VERIFY** the following:

- All obtainable 3-SR-2 modes 1,2 and 3 section data has been obtained. ☐
- All 3-SR-2 data meets the requirements for the Reactor to be placed in Mode 2 per LCO 3.0.4 and SR-3.0.4. ☐

______	______	______
Initials STA	Date	Time

NOTE
The Shift Manager/Unit Supervisor will perform Step 5.0[9.4].

[9.4] **REVIEW** the Configuration Log (SPP-10.1), TACFs, LCO Tracking Log, and Clearance Books for System Operability Impact for MODE 2.

	______	______	______
(R)	Initials	Time SM/US	Date

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

NOTE

Prior to Mode change, verification of 3-SR-3.4.1(SLO) is completed if operating in Single Loop Operation, to satisfy Tech Specs and SR-3.0.4.

[10] **OBTAIN** Reactor mode switch key from Shift Manager.

And

PLACE REACTOR MODE SWITCH, 3-HS-99-5A-S1 in START/HOT STBY position.

(R) ____________ ____________ ____________
Initials Time Date

[11] **VERIFY** proper RWM sequence selected, as compared to 3-SR-3.1.3.5(A), CONTROL ROD COUPLING INTEGRITY CHECK (N/A if RWM inoperable.)

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[12] [NER/C] **ESTIMATE** the critical rod configuration per 0-TI-248. [SOER 88-002]

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[13] **VERIFY** RWM is latched to the correct group. (N/A if RWM inoperable.)

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE

1) Normal CRD drive water differential pressure is between 250 psid and 270 psid for all control rods designated for rod notch withdrawal. 3-OI-85 provides instruction for a higher pressure if required to move a control rod off of "00"
2) Operations Management expectations are that 3-SR-3.3.2.1.2, RWM FUNCTIONAL TEST FOR START-UP, will be performed in Step 5.0[14] prior to pulling control rods for the purpose of making the Reactor critical. (Reference Tech Specs 3.3.2.1)

[14] **PERFORM** the Following:

- 3-SR-3.3.2.1.2, RWM FUNCTIONAL TEST FOR STARTUP.

(R) ____________ ____________ ____________
Initials Time Date

- 3-SR-3.3.2.1.7, RWM Program Verification.

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[15] **IF** Rod Worth Minimizer is not operable, **THEN**

PERFORM 3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check. (N/A if operable.)

(R) ____________ ____________ ____________
Initials Time Date

[16] [TSAR/C] **VERIFY** moderator temperature is greater than temperature required by Tech Specs 3.4.9-1 Figure 3.4.9-1, Curve #3. (Tech Specs requires SR-3.4.9.2 be performed within 15 minutes prior to Control Rod withdrawal to achieve criticality.) [Item C5] [3-SR-3.4.9.1(1)].

(R) ____________ ____________ ____________
Initials Time Date

[17] **VERIFY** Condensate System in short-cycle cleanup mode.
REFER TO 3-OI-2.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[18] **VERIFY** all preceding steps requiring signoff (R) have been signed prior to proceeding to the next step.

(R) ____________ ____________ ____________
Initials Time Date
Shift Manager

[19] **NOTIFY** Chemistry that Unit 3 is ready for startup.

____________ ____________ ____________
Initials Time Date

[20] **NOTIFY** Radiation Protection that UNIT 3 is ready for startup. **RECORD** time Radiation Protection notified in NOMS Narrative Log.

(R) ____________ ____________ ____________
Initials Date Time

[20.1] **VERIFY** appropriate data recorded on Appendix A in accordance with Appendix A instructions.

(R) ____________ ____________ ____________
Initials Date Time

[21] **NOTIFY** Chattanooga Load Coordinator and Wilson Load Dispatcher of impending Reactor startup.

____________ ____________ ____________
Initials Date Time

[22] **ANNOUNCE** over plant PA system that “Unit 3 Reactor startup is commencing”.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Control rods must not be withdrawn unless the applicable portions of 3-SR-3.3.2.1.2, RWM FUNCTIONAL TEST FOR STARTUP, have been satisfactorily completed within the last 8 hours. (NA if RWM is inoperable and Technical Specification 3.3.2.1.C is met.)

2) [NER/C] The Unit Operator withdrawing control rods is responsible for controlling reactivity and is charged with monitoring nuclear instrumentation. Any actions that affect reactivity (including recirculation control, feedwater addition, use of nuclear steam for auxiliaries, or SRV/HPCI/RCIC testing) should be clearly announced, coordinated, and monitored for correct response subsequent to the reactivity change. [SOER 88-002]

3) During a hot startup following a scram from high power, the condition of peak Xenon with no moderator voids could exist at time of startup. Under these conditions extremely high rod notch worth can be encountered.

4) [INPO/C] All activities that can distract the operator and supervisors involved with the Reactor startup (such as shift turnover, surveillance testing, and excessive personnel in the Control Room) should be avoided during the approach to criticality. [INPO SOER 88-002]

[23] **PERFORM** the following:

[23.1] **VERIFY** that an SRO is present in the Control Room who is designated by the Shift Manager to oversee the approach to criticality and ensure reactivity is added in a controlled and cautious manner.

(R) ____________ ____________ ____________
Initials Time Date

[23.2] **VERIFY** completion of pre-evolution briefing on reactivity management (SPP-10.4) prior to the approach to criticality.

(R) ____________ ____________ ____________
Initials Time Date

[23.3] **VERIFY** applicable portions of 3-SR-3.3.2.1.2 have been satisfactorily completed within the last 8 hours. (N/A, if RWM is inoperable and Tech Specs 3.3.2.1.C is met.)

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[23.4] **OBTAIN** permission from the Operations Superintendent and the Plant Operations Manager, or their alternates, in concurrence with the Plant Manager, to proceed with unit startup.

(R) ____________ ____________ ____________
Initials Time Date
Shift Manager

NOTE

Source Range Data should be taken just prior to pulling control rods for startup. This will minimize a difference in source range counts caused by a change in plant conditions.

[24] **PERFORM** the following to startup the Reactor:

[24.1] **PERFORM** the following for SRMs on Panel 3-9-5:

RECORD SOURCE RANGE MONITORS reading:

CHANNEL A LEVEL________cps

CHANNEL C LEVEL________cps

CHANNEL B LEVEL________cps

CHANNEL D LEVEL________cps

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE

[NER/C] A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. This requirement along with close monitoring of neutron monitoring instrumentation should assure a slow controlled approach to criticality. Criticality should be expected at all times. [SOER 88-002]

[24.2] **CALCULATE** SRM count rate at which notch withdrawal limitations will be imposed by multiplying pre-startup count rate, recorded in Step 5.0[24.1], by a factor of 16. **RECORD** results below and at Step 5.0[26]:

CHANNEL A LEVEL________cps

CHANNEL C LEVEL________cps

CHANNEL B LEVEL________cps

CHANNEL D LEVEL________cps

(R) ____________ ____________ ____________
Initials Time Date
1st

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[24.3] **RECORD** channels selected and pen inking on SRM LEVEL recorder (select highest-reading channels):

RED pen ________________

GREEN pen ______________

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[24.4] **RECORD** SRM BYPASS, 3-HS-92-7A/S3 joystick position. (N/A if not bypassed.)

Channel ________ bypassed

(R) ________ ________ ________
Initials Time Date

[24.5] **VERIFY** the following Panel 3-9-5 SRM display lights extinguished:

- HIGH HIGH. ☐
- HIGH OR INOP. ☐
- DNSCL. ☐
- BYPASSED (Will be illuminated if channel bypassed.) ☐
- RETRACT PERMIT (NA if above setpoint.) ☐
- PERIOD. ☐

(R) ________ ________ ________
Initials Time Date

CAUTION
Criticality should be expected at all times.

[24.6] **COMMENCE** rod withdrawal. REFER TO 3-OI-85 and 3-SR-3.1.3.5(A).

(R) ________ ________ ________
Initials Time Date

[24.7] **CHECK** coupling integrity by performing 3-SR-3.1.3.5(A) as each control rod is withdrawn.

(R) ________ ________ ________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[24.8] [INPO/C] **MONITOR** SRM/IRM instrumentation closely during rod pulls while approaching criticality, pausing between rod pulls as needed for neutron level stabilization. [INPO SER 89-006]

(R) ____________ ____________ ____________
Initials Time Date

[24.9] **CONTINUE** withdrawing control rods in accordance with 3-SR-3.1.3.5(A).

(R) ____________ ____________ ____________
Initials Time Date

NOTE
The following steps apply for all Control Rod Withdrawals and does not require a operator signoff for the steps. The actions should be reviewed by all personnel involved with withdrawing control rods.

[25] **MONITOR** Reactor power during rod withdrawals and perform the following for the associated conditions.

[25.1] **IF** single-notch withdrawals result in a Reactor period of less than 60 seconds, **THEN**

PERFORM the following:

[25.1.1] **REINSERT** the last control rod pulled to obtain a stable period greater than 60 seconds.

[25.1.2] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[25.2] **IF** a Reactor period of less than 30 seconds is observed, **THEN**

PERFORM the following:

[25.2.1] **INSERT** control rods in accordance with 3-SR-3.1.3.5(A).

[25.2.2] **VERIFY** Reactor subcritical.

[25.2.3] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

5.0 INSTRUCTION STEPS (continued)

[25.3] **IF** a Reactor period of less than 5 seconds is observed, **THEN**

SHUT DOWN the Reactor until a thorough assessment has been performed. REFER TO 3-GOI-100-12A.

CAUTION

1) Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and buildup of plutonium.
2) [NER/C] When rod movement is restricted to notch withdrawal, failure to stop at each notch position may result in high notch worth. [GE SIL 316]

NOTE

Once required, Control rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

[26] **WHEN** SRMs indicate the calculated values recorded below:

CHANNEL A LEVEL________cps

CHANNEL C LEVEL________cps

CHANNEL B LEVEL________cps

CHANNEL D LEVEL________cps,

THEN

START single-notch withdrawal of control rods.

(R) ____________ ____________ ____________
Initials Time Date
1st

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Criticality should be expected at all times.
2) Extended operation close to the point of criticality could result in inadvertent criticality and must be avoided.

[27] **WHEN** in a configuration that is expected to be near critical, **AND** Nuclear Instrument response is **NOT** as expected, **THEN**

NOTIFY Reactor Engineer and Shift Manager.

_____________ _____________ _____________
Initials Date Time

[28] **IF** operation is to be suspended for greater than one hour near the point of criticality, **THEN**

PLACE the Reactor core sufficiently subcritical as directed by the Shift Manager and as advised by the Reactor Engineer, to avoid an inadvertent criticality. (Otherwise N/A)

_____________ _____________ _____________
Initials Date Time

[29] **WITHDRAW** control rods to maintain a period of 100 seconds or greater as indicated on the following indicators on Panel 3-9-5:

- CHANNEL A PERIOD, 3-XI-92-7/44A. ☐
- CHANNEL B PERIOD, 3-XI-92-7/44B. ☐
- CHANNEL C PERIOD, 3-XI-92-7/44C. ☐
- CHANNEL D PERIOD, 3-XI-92-7/44D. ☐

(R) _____________ _____________ _____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE
Steps 5.0[30.1] through 5.0[30.3] may be signed off after completion of Step 5.0[30.3] when the Reactor is stable.

[30] **WHEN** Reactor is critical and desired period is obtained, as indicated by a rising neutron flux on a constant period with no rod motion, **THEN**

PERFORM the following:

[30.1] **PERFORM** verification of criticality

(R) __________ __________ __________
Initials Time Date
1st

(R) __________ __________ __________
Initials Time Date
2nd Party

5.0 INSTRUCTION STEPS (continued)

NOTE

Period is measured directly from IRMs, using one of the following methods:

1) MULTIPLY time for 10% power rise by 10.5.
2) MULTIPLY doubling time by 1.445.
3) DIVIDE time for decade rise by 2.3.
4) Directly, time for power to rise from 25 to 68.

[30.2] **RECORD** the following in the Narrative Log:

- Period __________

  (R) __________ Initials __________ Time __________ Date

- Time __________

  (R) __________ Initials __________ Time __________ Date

- Rod Group __________

  (R) __________ Initials __________ Time __________ Date

- Rod Number __________

  (R) __________ Initials __________ Time __________ Date

- Rod Notch __________

  (R) __________ Initials __________ Time __________ Date

5.0 INSTRUCTION STEPS (continued)

- Recirc Pump 3A and 3B Temperatures using either of the following: (N/A indication for a pump that is OOS and in Single Loop Operation.)
  1. RECIRC PUMPS DISCH TEMP PMP-3A (PMP-3B), red pen (green pen) on 3-TR-68-2 on Panel 3-9-4.
  2. RECIRC PMP A (B) SUCT TEMP 68-6A (68-83A) on ICS.
  3. RECIRC PMP A (B) DISCHARGE TEMP 68-2 (68-78) on ICS.

________ / ________
3A LOOP °F / 3B LOOP °F

(R) ________ ________ ________
Initials Time Date

[30.3] **VERIFY** Reactor Engineer records applicable criticality data in 3-SR-3.1.1.1.

________ ________ ________
Initials Date Time

[31] **VERIFY** Reactor period greater than 30 seconds.

(R) ________ ________ ________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE

1) Completing paper closure of 3-SR-3.3.1.1.5 is not required prior to performing Step 5.0[32]. HOWEVER, ALL AC steps must be VERIFIED COMPLETED SATISFACTORY prior to withdrawing SRMs.
2) Tech Spec Bases states that overlap between SRMs and IRMs exists when IRM downscale indications have cleared and IRM readings are on-scale and trending higher prior to SRMs reaching 10^5 cps.

[32] **VERIFY** SRM/IRM overlap by obtaining data and completing 3-SR-3.3.1.1.5 SRM and IRMs Overlap Verification.

(R) __________ __________ __________

Initials Time Date

Reactor Engineer

NOTES

1) SRMs are fully withdrawn when IRMs are on Range 3 or above and indicating above their downscale trip point.
2) If a shutdown margin test has been performed using a different rod sequence, 3-SR-3.1.3.5(A) will provide required actions to insert all control rods, establish normal sequence and perform the subsequent start up with re-entry at Step 5.0[23].

[33] **WITHDRAW** SRMs as necessary, to maintain them on scale between 10^2 cps and 10^5 cps.

__________ __________ __________

Initials Date Time

[34] **MAINTAIN** IRMs on scale between approximately 25 and 75 using IRM range switches.

__________ __________ __________

Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[35] **ENSURE** 3-SI-4.6.B.1-4 has been satisfactorily completed prior to pressurizing Reactor.

(R) ____________ ____________ ____________
Initials Time Date
Chem Shift Supv

[36] **WHEN** all operable IRMs are on Range 3 or above, **THEN**

WITHDRAW all operable SRMs.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTION

1) When Reactor coolant temperature is less than 215°F, a maximum heatup rate limit of 50°F/hr will reduce the 0_2 and Hydrogen Peroxide content of the coolant.
2) During Reactor Heatup with Reactor coolant temperature greater than or equal to 215°F, and during Reactor Cooldown, the optimum rate of temperature change is 20° every 15 minutes. This will ensure the administrative limit of 90°F/HR is not exceeded. Do not attempt to "makeup" for time intervals which fall short of 20°F. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines will assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is not exceeded.
3) During Reactor heatup, operators should use metal temperatures as a reminder that as metal heats up, the moderator HEATUP RATE will rise with the same amount of heat input.

NOTE

The Heatup/Cooldown rate graph on ICS may be monitored by typing HUR or by selecting Heatup rate from the Operations Support (OPSSUP) menu.

[37] **INITIATE** 3-SR-3.4.9.1(1), using a licensed Unit Operator, at least 15 minutes prior to heatup. Copies of Illustration 3 should be used to plot heatup rate. (N/A, if performing a startup not requiring a heatup.)

OR

VERIFY 3-SR-3.4.9.1(1), in progress per Attachment 2, Temperature Verifications From Cold Shutdown to 210°F. (N/A, if performing a startup not requiring heatup.) (N/A, if performing a startup not requiring heatup.)

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) [NRC/C] When ranging an IRM from Range 6 to Range 7, power indicated on Range 7 may not be consistent with indications observed when ranging from ranges 1 through 6. Should this occur, the Shift Manager or Reactor Engineer should determine if IRM response is acceptable, or if calibrations to ensure adequate gain settings are necessary. [LER 50-260-93006]

2) For Unit 3 Middle of Core Life to End of Core Life, the moderator temperature coefficient of reactivity becomes positive as control rods are withdrawn for startup when moderator temperature is below 350°F. The resulting effect will be for Reactor power to rise until the moderator begins boiling. Exercise additional caution when withdrawing control rods under this condition.

NOTE

If in Single Loop Operation, 3-SR-3.4.1(SLO) is required to be completed prior to Mode change to satisfy Tech Specs and SR-3.0.4.

[38] **RAISE** power level by control rod withdrawal until desired rate of heating power is reached. (Usually Range 7 on IRMs.)

________ Initials ________ Date ________ Time

[39] **PERFORM** the following for EHC system:

[39.1] **VERIFY** EHC SETPOINT, 3-PI-47-162 is set at a minimum of 150 psig. (may be set higher depending on plant conditions (actual Reactor pressure))

(R) ________ Initials ________ Time ________ Date

[39.2] **VERIFY** EHC inservice prior to 150 PSIG. (N/A IF Reactor pressure is greater than 150 psig prior to startup.)

(R) ________ Initials ________ Time ________ Date

5.0 INSTRUCTION STEPS (continued)

[40] **IF** Sealing Steam System is not in service, **THEN**

PERFORM the following (NA if previously performed):

[40.1] **ESTABLISH** sealing steam to Main Turbine and Feedpump Turbines using: REFER TO 3-OI-47C.

- Aux Boiler steam.

OR

- Nuclear steam may be used if Reactor is still pressurized (as in a hot restart).

AND

A RFP is being used to maintain Reactor water level.

Initials _______ Date _______ Time _______

[40.2] **IF** not already performed, **THEN**

ESTABLISH condenser vacuum. REFER TO 3-OI-66.

Initials _______ Date _______ Time _______

[41] **IF** the Reactor is being placed in a HOT STANDBY condition, **THEN**

PERFORM ATTACHMENT 3, Startup With MSIVs Closed. (Otherwise N/A).

Initials _______ Date _______ Time _______

5.0 INSTRUCTION STEPS (continued)

NOTE
At low Reactor pressure elevated Off Gas flow, and lower Condenser Vacuum may be noted temporarily after opening MSIVS.

[42] **WHEN** Reactor Coolant Temperature indicates above 215°F AND Reactor pressure indicates above 0 psig, **THEN**

[42.1] **PERFORM** the following on Panel 3-9-3:

- **VERIFY OPEN** 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A.

  Initials ________ Date ________ Time ________

- **VERIFY OPEN** 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A.

  Initials ________ Date ________ Time ________

- **VERIFY OPEN** 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A.

  Initials ________ Date ________ Time ________

- **VERIFY OPEN** 3-FCV-1-57 using MSIV DOWNSTREAM DRAINS SHUTOFF, 3-HS-1-57A.

  Initials ________ Date ________ Time ________

[42.2] **THROTTLE OPEN** 3-FCV-1-59 using DOWNSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-59A while maintaining appropriate heatup rate..

Initials ________ Date ________ Time ________

5.0 INSTRUCTION STEPS (continued)

[42.3] **VERIFY** RPV metal temperatures to the right of Tech Spec Curve 3.4.9-1 as referenced in 3-SR-3.4.9.1(1).

_____ Initials _____ Date _____ Time

[42.4] **WHEN** verification of RPV metal temperatures to the right of Tech Spec Curve 3.4.9-1 as referenced in 3-SR-3.4.9.1(1) is complete, **THEN**

VERIFY OPEN Outboard Main Steam Isolation valves on Panel 3-9-3:

- 3-FCV-1-15 using MSIV LINE A OUTBOARD, 3-HS-1-15A.

_____ Initials _____ Date _____ Time

- 3-FCV-1-27 using MSIV LINE B OUTBOARD, 3-HS-1-27A.

_____ Initials _____ Date _____ Time

- 3-FCV-1-38 using MSIV LINE C OUTBOARD, 3-HS-1-38A.

_____ Initials _____ Date _____ Time

- 3-FCV-1-52 using MSIV LINE D OUTBOARD, 3-HS-1-52A.

_____ Initials _____ Date _____ Time

5.0 INSTRUCTION STEPS (continued)

NOTE
When Reactor water temperature is greater than 215°F, heatup is limited to 90°F/Hr.

[43] **IF** Reactor coolant oxygen content CANNOT be maintained less than 300 ppb when coolant temperature is greater than 285°F, **THEN**

PERFORM the following: (N/A if less than 300 ppb.)

[43.1] **SHUT DOWN** Reactor. REFER TO 3-GOI-100-12A.

(R) ____________ ____________ ____________
Initials Time Date

CAUTION
During Reactor Cooldown, the optimum rate of temperature change is 20°F every 15 minutes. This will ensure the administrative limit of 90°F/Hr is not exceeded. Do not attempt to "makeup" for time intervals which fall short of 20°F. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines will assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is not exceeded.

[43.2] **COOL DOWN** Reactor at a rate not to exceed 90°F/hr.

(R) ____________ ____________ ____________
Initials Time Date

[43.3] **REQUEST** Chemistry to sample for dissolved oxygen every 4 hours until MODE 4 is achieved.

(R) ____________ ____________ ____________
Initials Time Date

[43.4] **EXIT** this procedure and **ENTER** 3-GOI-100-12A.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE
Section 5.0[44] may be N/A'd if Reactor pressure was not lowered below the RCIC low pressure isolation setpoint during the unit shutdown.

[44] **WHEN** PRESS A, 3-PI-3-54, indicates approximately 70 psig on Panel 3-9-5, **THEN**

PERFORM the following:

[44.1] **VERIFY RESET** RCIC steam line low pressure isolation. REFER TO 3-OI-71.

(R) ____________ ____________ ____________
Initials Time Date

[44.2] **WARM** and **PRESSURIZE** RCIC steam line. REFER TO 3-OI-71. (N/A if already performed.)

(R) ____________ ____________ ____________
Initials Time Date

[44.3] **VERIFY** RCIC in Prestartup/Standby Readiness. REFER TO 3-OI-71.

(R) ____________ ____________ ____________
Initials Time Date

[44.4] **VERIFY** the following fuses installed and Caution Order removed for RCIC ST LINE TRAP BYPASS VLV, 3-LCV-071-0005 (3A Elec. BD RM, 3-LPNL-925-0032, JJ Block).

- 3-FU1-071-0005A, 13AF17. ☐
- 3-FU1-071-0005B, 13AF18. ☐

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) RWCU blowdown is limited to maintain WATER TO RWCU DEMINS, 3-XS-69-6 point 3, temperature less than 130°F, as indicated by RWCU HX TEMP, 3-TI-69-6, located on Panel 3-9-4.
2) At Reactor vessel pressures less than rated, as much as a 13 inch level discrepancy can exist between the 3-LI-3-208A(B)(C)(D) uncompensated narrow range level instruments and the 3-LI-3-53(60)(206)(253) compensated narrow range level instruments. The 3-LI-3-208A(B)(C)(D) level instruments are not temperature compensated and the lower the pressure on the Reactor vessel, the higher the 3-LI-3-208A(B)(C)(D) level instruments will read. Failure to maintain the RPV level indicated on the 3-LI-3-208A(B)(C)(D) level instruments less than 48 inches can result in unnecessary turbine trips (i.e., RFPTs, HPCI, RCIC, and Main Turbine).

[45] **IF** Reactor is still pressurized as in a hot restart **AND** a RFP is in service to maintain Reactor water level, **THEN**
MAINTAIN Reactor water level between 28 inches and 38 inches as indicated by RX LVL (RED pen) on RX VESSEL LEVEL/TOTAL FW FLOW recorder, 3-XR-3-53, AND less than 48" on 3-LI-3-208A(B)(C)(D). (N/A if RFP is not being used to maintain Reactor water level)

(R) __________ __________ __________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[46] **IF** Reactor pressure is less than 750 psig AND a RFP is not being used to maintain Reactor water level, **THEN**

MAINTAIN Reactor water level between 28 inches and 50 inches as indicated by RX LVL (RED pen) on RX VESSEL LEVEL/TOTAL FW FLOW recorder, 3-XR-3-53, AND less than 48" on 3-LI-3-208A(B)(C)(D), using the following vessel makeup and level control systems: (N/A if RFP is being used to maintain Reactor water level)

- CRD System (40 to 65 gpm). (Control Rod Drive Hydraulic System Startup section of 3-OI-85)..
- CRD System (up to 80 gpm). (CRD Pump Operation at Elevated Flow section of 3-OI-85).
- RWCU System. (3-OI-69).
- Condensate System. (3-OI-2).

(R) __________ __________ __________
Initials Time Date

NOTE
Step 5.0[47] may be marked N/A if Reactor pressure was not lowered below the HPCI low pressure isolation setpoint during the unit shutdown.

[47] **WHEN** RX PRESSURE WIDE RANGE, PRESS A, 3-PI-3-54, indicates greater than approximately 110 psig, **THEN**

PERFORM the following:

[47.1] **VERIFY RESET** HPCI steam line low pressure isolation. REFER TO 3-OI-73.

(R) __________ __________ __________
Initials Time Date

[47.2] **WARM** and **PRESSURIZE** HPCI steam line. REFER TO 3-OI-73. (N/A if previously performed.)

(R) __________ __________ __________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[47.3] **VERIFY** HPCI in Prestartup/Standby Readiness. REFER TO 3-OI-73.

(R) ______ ______ ______
Initials Time Date

[47.4] **VERIFY** the following fuses installed and Caution Order removed for HPCI ST LINE TRAP BYPASS VLV, 3-LCV-073-0005 (3-PNLA-009-0003, REAR BAY 6, BB BLOCK GREEN).

- 3-FU2-073-0005. ☐
- 3-FU2-073-23AF9. ☐
- 3-FU2-073-23AF10. ☐

(R) ______ ______ ______
Initials Time Date

[47.5] **BEGIN** warming Reactor Feedpump to be placed in service. REFER TO 3-OI-3. (N/A if previously performed,)

______ ______ ______
Initials Date Time

CAUTION

1) If proper care is not exercised while placing the Startup Level Control Valve in service, over filling the Reactor vessel or quick charging the high pressure feedwater heaters may occur.
2) Failure to verify feedwater alignment (i.e., Feedwater Heaters and piping are filled and vented prior to opening the RFP Discharge Valve) per 3-OI-3, Placing the Startup Level Control Valve in Service section, may cause water hammer. [BFNPER 01-004201-000]

[47.6] **VERIFY** Feedwater System aligned for injection to Reactor vessel with Startup Level Control Valve available for service. REFER TO 3-OI-3.

______ ______ ______
Initials Date Time

[48] **VENT** the drywell, as necessary, to maintain drywell pressure less than 1.33 psig. REFER TO 3-OI-64.

______ ______ ______
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

NOTES

1) CRD flow of approximately 80 gpm with all steam line drains closed, may be sufficient to test the EHC system with the equivalent steam flow of one Turbine Bypass Valve less than or equal 50% open.
2) Steam drains should not be closed until RPV pressure is approximately 100 psig (338 F) to allow purging the lines of condensation, minimizing chances of water hammer.
3) The following steps will isolate the Reactor and Reactor water level should be closely monitored during pressurization.

[49] **VERIFY** the following prior to exceeding 125 psig Reactor pressure. (N/A if Hot Startup is being performed.)

- MAIN STEAM LINE DRAIN VALVES, 3-FCV-1-55, 3-FCV-1-56, 3-FCV-1-58, and 3-FCV-1-59 CLOSED.

(R) ____________ ____________ ____________
Initials Time Date

- STOP VLV BEFORE SEAT DRAINS, 3-FCV-6-100, 3-FCV-6-101, 3-FCV-6-102, and 3-FCV-6-103 CLOSED.

(R) ____________ ____________ ____________
Initials Time Date

- All RFP turbine warming drains closed on RFP not being warmed.

(R) ____________ ____________ ____________
Initials Time Date

- Turbine steam seals isolated from the Reactor steam supply.

(R) ____________ ____________ ____________
Initials Time Date

- Off-Gas Preheaters isolated from Reactor steam supply.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

- SJAEs isolated from the Reactor steam supply.

(R) ____________ ____________ ____________
Initials Time Date

- Turbine bypass valves closed.

(R) ____________ ____________ ____________
Initials Time Date

- RFW START-UP LEVEL CONTROL, 3-LIC-3-53 is available and aligned to the RPV as needed. (If CRD system cannot maintain Reactor water level RFP SU Bypass valve should be utilized.)

(R) ____________ ____________ ____________
Initials Time Date

NOTE
Backfilling of Moisture Separator Reservoir level control sensing lines should be completed prior to initiation of Main Turbine shell or chest warming.

[50] **NOTIFY** Instrument Maintenance to backfill the MSLCR level control system sensing lines. (N/A if recovering from a load reduction and the turbine remained on line).

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

CAUTION

If not adjusted accordingly, turbine first stage pressure will rise as Reactor pressure rises while in shell warming or chest warming. Extreme caution must be exercised to ensure turbine first stage pressure is maintained in the pressure band dictated by 3-OI-47 to prevent a Reactor scram.

NOTES

1) Main turbine shell warming or chest warming may be performed concurrently with pressurizing the reactor provided it is accomplished prior to exceeding 350 psig. If additional shell warming or chest warming is desired after exceeding 350 psig, it may only be conducted parallel to raising reactor pressure to rated, with the approval of OPS Superintendent/OPS Manager. If the CRD system cannot maintain inventory, then shell warming or chest warming is resumed after placing the first Reactor Feedpump in service.

2) Backfilling of Moisture Separator Reservoir level control sensing lines should be completed prior to initiation of Main Turbine shell or chest warming.

[51] **IF** EHC is available, **THEN**

INITIATE shell warming high pressure turbine at the Unit Supervisor's discretion. REFER TO 3-OI-47. (N/A if not performed at this time.)

__________ Initials   __________ Date   __________ Time

[52] **IF** EHC is available, shell warming is complete, and chest warming is required, **THEN**

INITIATE Chest warming at Unit Supervisor's discretion. REFER TO 3-OI-47. (N/A if not performed at this time.)

__________ Initials   __________ Date   __________ Time

5.0 INSTRUCTION STEPS (continued)

[53] **WHEN** Reactor pressure is approximately 150 psig, **THEN**

PERFORM the following:

[53.1] **VERIFY** operability of EHC Control System by allowing a bypass valve to throttle OPEN. (N/A if Reactor is still pressurized as in hot restart).

__________ Initials __________ Date __________ Time

> **NOTE**
>
> The following steps will ensure the CRD is aligned for level control and the capabilities for level control are not overrun.

[53.2] **STOP** control rod withdrawal and subsequent Turbine Bypass Valve opening.

__________ Initials __________ Date __________ Time

[53.3] **VERIFY** RFW START-UP LEVEL CONTROL, 3-LIC-3-53 is **NOT** being used to augment the CRD SYSTEM for level control. (i.e., not injecting feedwater)

__________ Initials __________ Date __________ Time

5.0 INSTRUCTION STEPS (continued)

CAUTION

1) If not previously performed, RCIC and HPCI must be proven operable within 12 hours after reaching 150 psig, but prior to exceeding 165 psig Reactor pressure.
2) When the pressure control swaps from "HEADER PRESSURE CONTROL" to "REACTOR PRESSURE CONTROL" the pressure set will be actual Reactor pressure at the time the swap is done, regardless of any previously raised Reactor pressure set done during a Reactor startup.

NOTES

1) To provide adequate steam flow for RCIC, 3-SR-3.5.3.4 requires at least one bypass valve to be > 50% open.
2) To provide adequate steam flow for HPCI, 3-SR-3.5.1.8, at least two turbine bypass valves must be open.

[54] **PERFORM** the following to support RCIC and/or HPCI operability:

[54.1] **REFER TO** Tech Specs 3.5.3 and 3.5.1, respectively, to determine RCIC and/or HPCI operability.

_____________ Initials _____________ Date _____________ Time

[54.2] **RAISE** EHC Pressure setpoint as directed by Unit Supervisor using Pressure Setpoint RAISE Pushbutton, 3-HS-47-162B, on Panel 3-9-7, but **NOT** to exceed 165 psi prior to HPCI and RCIC being operable. (N/A if not required).

_____________ Initials _____________ Date _____________ Time

[54.3] **IF** 3-SR-3.5.3.4 and 3-SR-3.5.1.8 are required to be performed for the current operating cycle, **THEN**

VERIFY 3-SR-3.5.3.4 and 3-SR-3.5.1.8 are complete with Reactor pressure greater than 150 psig and prior to exceeding 165 psig. (N/A if not Required.)

(R) _____________ Initials _____________ Time _____________ Date

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

CAUTION

1) [II/F] Prior to initiating any event which adds, or has the potential to add, heat energy to the suppression chamber, the Unit Supervisor will evaluate the necessity of placing suppression pool cooling in service. This is due to the potential of developing thermal stagnation during sustained heat additions. [II-B-91-129]
2) If not previously performed, RCIC and HPCI must be proven operable within 12 hours of reaching 150 psig Reactor pressure.

NOTE

Step 5.0[55] is performed to ensure RCIC and HPCI are proven operable prior to exceeding shutoff head of RHR and Core Spray pumps.

[55] **WHEN** Reactor pressure is greater than 150 psig, but less that 165 psig, **THEN**

[55.1] **RECORD** Time LCO entered. (N/A, if no LCO entry is required.)

Date ___________ Time ___________

(R) ___________ ___________ ___________
Initials Time Date

[55.2] **VERIFY** RCIC and HPCI are operable prior to exceeding 165 psig and within 12 hours of entering LCO in Step 5.0[55.1].
REFER TO Tech Specs 3.5.3 and 3.5.1, respectively AND **ENTER** in NOMS Narrative Log. (N/A if no LCO entered).

(R) ___________ ___________ ___________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Failure to monitor Reactor water level closely while performing the next step may result in loss of water level due to exceeding CRD makeup capacity.
2) RFW START-UP LEVEL CONTROL, 3-LIC-3-53 must be closed prior to exceeding shutoff head (350 psig). If the Start-Up Level Control valve is being used to augment level control, then the CRD system, which is the only readily High Pressure makeup source, cannot maintain Reactor water level above 350 psig. Therefore the CRD system needs to be the only high pressure makeup source.
3) When the pressure control swaps from "HEADER PRESSURE CONTROL" to "REACTOR PRESSURE CONTROL" the pressure set will be actual Reactor pressure at the time the swap is done, regardless of any previously raised Reactor pressure set done during a Reactor startup.
4) At Reactor vessel pressures less than rated, as much as a 13 inch level discrepancy can exist between the 3-LI-3-208A(B)(C)(D) uncompensated narrow range level instruments and the 3-LI-3-53(60)(206)(253) compensated narrow range level instruments. The 3-LI-3-208A(B)(C)(D) level instruments are not temperature compensated and the lower the pressure on the Reactor vessel, the higher the 3-LI-3-208A(B)(C)(D) level instruments will read. Failure to maintain the RPV level indicated on the 3-LI-3-208A(B)(C)(D) level instruments less than 48 inches can result in unnecessary turbine trips (i.e., RFPTs, HPCI, RCIC, and Main Turbine).

[56] **CONCURRENTLY PERFORM** the following:

[56.1] **MAINTAIN** Reactor water level between +12 and +50 inches, AND less than 48 inches on 3-LI-3-208A-D.

(R) ____________ ____________ ____________
Initials Time Date

[56.2] **DEPRESS** Pressure Setpoint RAISE push-button, 3-HS-47-162B, on Panel 3-9-7, as necessary to maintain EHC SETPOINT, 3-PI-47-162 above Reactor pressure until reaching approximately 955 psig (N/A if a Hot Startup is being performed and a RFP is maintaining level).

(R) ____________ ____________ ____________
Initials Time Date

[57] **VERIFY** EHC SETPOINT, 3-PI-47-162 set at 955 psig on Panel 3-9-7.

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[58] **CONTINUE** withdrawing Control Rods at the Unit Supervisor discretion.

Initials ______ Date ______ Time ______

[59] **IF** shell warming or chest warming are **NOT** to be performed in parallel with Reactor pressurization, **THEN**

STOP shell warming and chest warming the high pressure turbine prior to exceeding 350 psig. REFER TO 3-OI-47. (N/A if warming is not in progress or is to be performed in parallel with Reactor pressurization.)

Initials ______ Date ______ Time ______

CAUTIONS
1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[60] **WHEN** Reactor pressure is approximately 450 psig to 500 psig, **THEN**

PERFORM the following:

[60.1] **VERIFY** two Condensate and two Condensate Booster pumps running. REFER TO 3-OI-2.

Initials ______ Date ______ Time ______

[60.2] **VERIFY** Condensate System flow being maintained within the limits of 3-OI-2 using CNDS FLOW CONTROL SHORT CYCLE, 3-FC-2-29, on Panel 3-9-6, in AUTO/BAL.

Initials ______ Date ______ Time ______

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) If proper care is not exercised while placing a feedpump in service, over filling the Reactor vessel or quick charging the high pressure feedwater heaters may occur.
2) Failure to verify feedwater alignment (i.e., Feedwater Heaters and piping are filled and vented prior to opening the RFP Discharge Valve) per 3-OI-3, Placing the First RFP/RFPT In Service section, may cause water hammer. [BFNPER 01-004201-000]

NOTES

1) If required to maintain Reactor water level the Reactor Feed Pump may be used to add water in Step 5.0[60.3]. But, when no longer required, maintain discharge pressure approximately 100 psig below Reactor Pressure until required to be used.
2) The first Reactor Feed Pump will be placed fully in service when the first Turbine Bypass Valve is between 10% and 50 % open.

Start of Critical Step(s)

[60.3] **WHEN** Reactor pressure is approximately 750 psig, **THEN**

RAISE the first Reactor Feed Pump speed in manual control to approximately 100 psig below Reactor Pressure. REFER TO 3-OI-3.

Initials Date Time

End of Critical Step(s)

[60.4] **MAINTAIN** the Reactor Feed Pump in Step 5.0[60.3] approximately 100 psig below Reactor pressure unless required to be used to maintain Reactor water level.

Initials Date Time

[61] **IF** additional shell warming is required, **THEN**

REESTABLISH shell warming of high pressure turbine. REFER TO 3-OI-47. (N/A if not required.)

Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[62] **VERIFY** the following annunciators on Panel 3-9-5 are reset at approximately 850 psig:

- MAIN STEAM LINE CH A PRESS LOW (3-XA-55-5B, window 25). ☐
- MAIN STEAM LINE CH B PRESS LOW (3-XA-55-5B, window 26). ☐

(R) ____________ ____________ ____________
Initials Time Date

[63] **VERIFY** all surveillances required prior to going into MODE 1 are current.

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

(R) ____________ ____________ ____________
Initials Time Date
Unit Supervisor

[64] **VERIFY** the following in standby readiness:

- Rod Block Monitor. (3-OI-92C).

(R) ____________ ____________ ____________
Initials Time Date

- CAD System. (3-OI-84).

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

NOTE

Drywell to Torus differential pressure must be established within 24 hours after reaching 15% RTP per Tech Specs Section 3.6.2.6 as referenced in 3-OI-64.

[65] **PRIOR** to exceeding 950 psig, **PERFORM** the following:

[65.1] **VERIFY** EHC system in service. REFER TO 3-OI-47A.

(R) ____________ Initials ____________ Time ____________ Date

[65.2] **VERIFY** EHC Pressure Control is selected to REACTOR PRESSURE control prior to opening bypass valves.

(R) ____________ Initials ____________ Time ____________ Date

[65.3] **BEGIN** shell warming high pressure turbine at the Unit Supervisor's discretion. REFER TO 3-OI-47. (N/A if previously performed).

____________ Initials ____________ Date ____________ Time

[65.4] **WHEN** shell warming is complete, **THEN**

BEGIN Chest warming at the Unit Supervisor's discretion. REFER TO 3-OI-47. (N/A if previously performed).

____________ Initials ____________ Date ____________ Time

[65.5] **RECORD** the time 935 psig was obtained in the NOMS Narrative Log.

____________ Initials ____________ Time ____________ Date

5.0 INSTRUCTION STEPS (continued)

NOTES

1) Prior to entering Mode 1, the 150 psig test for both HPCI and RCIC must be completed and both declared operable. The 150 psig test may be completed by using either Nuclear Steam or Aux Boiler Steam.
2) RCIC must be proven operable at high Pressure within 12 hours from Reactor Steam Pressure reaching 950 psig and at leas one turbine bypass valve is full open.
3) HPCI must be proven operable at high pressure within 12 hours from Reactor Steam Pressure reaching 950 psig and at least two turbine bypass valves are full open).
4) Failure of the HPCI or RCIC High Pressure (950 psig) surveillance while in Mode 2, will preclude Mode 1 entry.
5) Failure of the HPCI or RCIC High Pressure (950 psig) surveillance in Mode 1, results in a 14 day LCO.
6) It is preferred to perform the HPCI and RCIC 950 psig surveillances in MODE 1 if the 12 hour LCO clock permits.

[66] **VERIFY** RCIC operable within 12 hours after Reactor pressure is greater than or equal to 950 psig, but less than or equal to 1040 psig, AND at least one turbine bypass valve is full open. **COMPLETE** 3-SR-3.5.3.3 OR **VERIFY** current (N/A if RCIC surveillance is going to be performed in Mode 1).

(R) ____________ ____________ ____________
Initials Time Date

[67] **VERIFY** HPCI operable within 12 hours after Reactor pressure is greater than or equal to 950 psig, but less than or equal to 1040 psig, AND at least two turbine bypass valves are full open. **COMPLETE** 3-SR-3.5.1.7 OR **VERIFY** current (N/A if HPCI surveillance is going to be performed in Mode 1).

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE
3-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test, is performed once per operating cycle. Tech Specs SR 3.4.3.2 requires that each S/RV opens when manually actuated, however it is not required to be performed until 12 hours after Reactor steam pressure and flow are adequate to perform the test. Adequate pressure at which this test is to be performed is greater than 935 psig. Adequate steam flow is represented by at least 3 main turbine bypass valves full open. A check with Work Control will determine whether this SR should be performed at this time.

[68] **WHEN** Reactor pressure is greater than or equal to 935 psig **AND** three (3) Turbine bypass valves are fully open, **THEN**

PERFORM the following:

- **ENTER** 12 hour LCO for Main Steam Relief Valve Operability. (Tech Specs LCO 3.4.3). (N/A, if 3-SR-3.4.3.2 is not required)

  (R) ____________ ____________ ____________
  Initials Time Date

- **RECORD** Time LCO entered. (N/A if LCO entry not required.)

  Date ____________ Time ____________

  (R) ____________ ____________ ____________
  Initials Time Date

- **IF** 3-SR-3.4.3.2 is required to be performed and Reactor pressure is greater than or equal to 935 psig with 3 turbine bypass valves full open, **THEN**

  PERFORM 3-SR-3.4.3.2. (Otherwise N/A)

  (R) ____________ ____________ ____________
  Initials Time Date

5.0 INSTRUCTION STEPS (continued)

[69] **WHEN** Reactor pressure reaches approximately 950 psig **AND** the first bypass valve 10% to 50% open, **THEN**

PERFORM the following:

[69.1] **VERIFY** the first RFP is in service maintaining Reactor water level.

__________ Initials __________ Date __________ Time

[69.2] **BEFORE** placing Seal Steam System, SJAE and Preheaters on nuclear steam, **PERFORM** the following: [BFN PER 126211]

[69.2.1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to transfer Seal Steam System, SJAE, and Preheaters to nuclear steam. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) __________ Initials __________ Date __________ Time

[69.2.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7] [BFN PER 126211]

(R) __________ Initials __________ Date __________ Time

NOTE

The Shift Manager/Unit Supervisor will perform Step 5.0[69.3]

[69.3] **REVIEW** the Daily Configuration Log, LCO Tracking Log, TACFs, and Clearance Books for System Operability impact for MODE 1 OPERATION.

(R) __________ Initials __________ Time __________ Date

Shift Mgr. / Unit Supv

5.0 INSTRUCTION STEPS (continued)

NOTE
Step 5.0[69.4] is to ensure that all required data is obtained prior to mode change per LCO 3.0.4 and SR-3.0.4 and is verified by the STA.

[69.4] **VERIFY** that all 3-SR-2 data meets the requirements for the Reactor to be placed in Mode 1 per LCO 3.0.4 and SR-3.0.4.

Initials STA	Date	Time

[69.5] **IF** Steam Seal pressure is being maintained at 1/2 psig to minimize water use during startup, **THEN**

VERIFY the following on Panel 3-9-7:

- 3-PCV-1-147 is in AUTO using STEAM SEAL REGULATOR, 3-HS-1-147. ☐
- STEAM SEAL HDR PRESSURE, 3-PI-1-148A indicates between 2 1/2 psig and 5 1/2 psig. ☐

Initials	Date	Time

[69.6] **TRANSFER** Sealing Steam System from auxiliary steam to nuclear steam. REFER TO 3-OI-47C. (N/A if previously placed on Nuclear steam)

Initials	Date	Time

[69.7] **TRANSFER** SJAE and Preheaters from auxiliary steam to nuclear steam. REFER TO 3-OI-66. (N/A if previously placed on Nuclear steam)

Initials	Date	Time

5.0 INSTRUCTION STEPS (continued)

[69.8] **BEGIN** warm-up of a second RFP. REFER TO 3-OI-3.

_______ Initials _______ Date _______ Time

[69.9] **NOTIFY** Electrical Maintenance, to **INSTALL** Main Generator and Exciter field brushes.

_______ Initials _______ Date _______ Time

[70] **IF** additional chest warming is required, **THEN**

ESTABLISH Turbine chest warming. REFER TO 3-OI-47.

_______ Initials _______ Date _______ Time

[71] **VERIFY** IRM/APRM overlap by operator visual observation before exceeding 5% power.

(R) _______ Initials _______ Time _______ Date

[72] **IF** leakage walkdowns are being performed, **THEN** (Otherwise N/A)

BEFORE exceeding 5% power, **PERFORM** the following:

[72.1.1] **NOTIFY** Radiation Protection that an RPHP is in effect for reactor power approaching 5%. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) _______ Initials _______ Date _______ Time

[72.1.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7, SOER 01-1, BFN PER 126211]

(R) _______ Initials _______ Date _______ Time

5.0 INSTRUCTION STEPS (continued)

[73] **CONTINUE** to withdraw control rods to raise Reactor power to approximately 8%. REFER TO 3-OI-85 and 3-SR-3.1.3.5(A).

(R) ______ ______ ______
Initials Time Date

[74] **VERIFY** all operable APRM downscale alarms are reset and no rod blocks exist.

(R) ______ ______ ______
Initials Time Date

[75] **VERIFY** the following:

- Hotwell Pressure is below -24" Hg. ☐
- CONDENSER A, B OR C VACUUM LOW annunciator, (3-XA-55-7B, window 17) is reset on Panel 3-9-7. ☐

(R) ______ ______ ______
Initials Time Date

[76] **VERIFY** all operable MSIVs are open on Panel 3-9-3.

(R) ______ ______ ______
Initials Time Date

[77] **IF** primary containment purge and/or Primary Containment Ventilation is in service, **THEN**

PLACE the following switches in the BYPASS position (Panel 3-9-3):

- PC PURGE DIV I RUN MODE BYPASS, 3-HS-64-24. ☐
- PC PURGE DIV II RUN MODE BYPASS, 3-HS-64-25. ☐

______ ______ ______
Initials Date Time

[78] **IF** Recirculation System is in Single Loop Operation, **THEN**

VERIFY that 3-SR-3.4.1(SLO) is completed to satisfy Tech Specs and SR-3.0.4. (Otherwise N/A)

______ ______ ______
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[79] **VERIFY** HPCI and RCIC OPERABLE for the 150 psig test, prior to entering MODE 1.

(R) __________ Initials __________ Time __________ Date

[80] **PERFORM** the following to go to MODE 1:

[80.1] **OBTAIN** Shift Manager permission to go to MODE 1.

Permission Granted to go to MODE 1:

____
Shift Manager Signature

(R) __________ Initials __________ Time __________ Date

MODE/CONDITION CHANGE

[80.2] **PRIOR** to exceeding 12% power, **PLACE** REACTOR MODE SWITCH to RUN.

AND

LEAVE the REACTOR MODE SWITCH key installed.

(R) __________ Initials __________ Time __________ Date

[81] **WHEN** REACTOR MODE SWITCH is placed in RUN, **THEN**

PERFORM the following:

[81.1] **RECORD** time in the NOMS Narrative Log.

(R) __________ Initials __________ Time __________ Date

[81.2] **VERIFY** RCIC operable. 3-SR-3.5.3.3 completed or current.

(R) __________ Initials __________ Time __________ Date

5.0 INSTRUCTION STEPS (continued)

[81.3] **VERIFY** HPCI operable. 3-SR-3.5.1.7 completed or current.

(R) ____________ ____________ ____________
Initials Time Date

[82] **IF** personnel are in the drywell, **THEN** (Otherwise N/A)

BEFORE exceeding 12% power, **PERFORM** the following:

[82.1.1] **NOTIFY** Radiation Protection that an RPHP is in effect for reactor power approaching 12% **AND** to evacuate all personnel from the drywell. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

[82.1.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7, SOER 01-1, BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

[83] **IF** requested by Reactor Engineer, **THEN**

PERFORM the following: (Otherwise N/A)

[83.1] **OBTAIN** Shift Manager's concurrence to bypass RWM.

____________ ____________ ____________
Initials Date Time

[83.2] **BYPASS** RWM in accordance with 3-OI-85.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

NOTES

1) Drywell to Torus differential pressure must be established within 24 hours after reaching 15% RTP per Tech Specs Section 3.6.2.6. (3-OI-64).

2) Primary Containment must be inerted within 24 hours of reaching 15% RTP per Tech Specs Section 3.6.3.2. (3-OI-76).

[84] **WHEN** Reactor is at 15% RTP, **THEN**

- **RECORD** the time 15% RTP was obtained in the NOMS Narrative Log.

(R) ____________ ____________ ____________
Initials Time Date

- **ENTER** 24 hour LCO for Drywell to Suppression Pool Differential Pressure. REFER TO Tech Specs LCO 3.6.2.6. (N/A if Drywell to Suppression Pool Differential Pressure already established)

(R) ____________ ____________ ____________
Initials Time Date

- **ENTER** 24 hour LCO for Primary Containment Oxygen Concentration. REFER TO Tech Specs LCO 3.6.3.2. (N/A if Primary Containment is already inerted)

(R) ____________ ____________ ____________
Initials Time Date

- **RECORD** Time LCO entered. (N/A if no LCO entry is required.)

Date ____________ Time ____________

(R) ____________ ____________ ____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[85] **WHEN** stable operation can be maintained, **THEN**

PLACE operating RFPT in automatic operation. REFER TO 3-OI-3.

_____ Initials _____ Date _____ Time

[86] **TRANSFER** IRM/APRM recorders to APRM.

(R) _____ Initials _____ Time _____ Date

[87] **TRANSFER** IRM/RBM recorders to RBM.

(R) _____ Initials _____ Time _____ Date

[88] **PERFORM** the following for IRMs:

[88.1] **WITHDRAW** all operable IRMs.

(R) _____ Initials _____ Time _____ Date

[88.2] **PLACE** all range switches to a position such that associated alarms are reset.

(R) _____ Initials _____ Time _____ Date

[88.3] **VERIFY** all IRM upscale or downscale alarms are reset.

(R) _____ Initials _____ Time _____ Date

5.0 INSTRUCTION STEPS (continued)

[88.4] **VERIFY** IRM recorder High Alarm setpoint programmed OFF.

_____________ _____________ _____________
Initials Date Time
IM

[89] **IF** Drywell Personnel Air Lock has been opened since startup began, **THEN**

VERIFY the following: (N/A if not opened since startup began.)

A. Drywell Personnel Air Lock interlocks have been re-established and tested per 3-SR-3.6.1.2.2. [BFPER 03-012038-000]

(R) _____________ _____________ _____________
Initials Time Date

B. Drywell Personnel Air Lock has been leak tested in accordance with 3-SR-3.6.1.2.1 as required by the Containment Leak Rate Program. [BFPER 03-012038-000]

(R) _____________ _____________ _____________
Initials Time Date
[NRC/C] Work Control

[90] **VERIFY** N_2 inerting of Drywell and Torus in progress or complete, to ensure inerting is complete within 24 hours of entering LCO in Step 5.0[84]. REFER TO 3-OI-76.

(R) _____________ _____________ _____________
Initials Time Date

[91] **VERIFY** nitrogen purge to TIP system operating. REFER TO 3-OI-94.

(R) _____________ _____________ _____________
Initials Time Date

[92] **IF** DWCA is aligned to Plant Control Air, **THEN** (Otherwise N/A)

ALIGN DWCA to Containment Inerting Nitrogen source.
REFER TO 3-OI-32A.

(R) _____________ _____________ _____________
Initials Time Date

5.0 INSTRUCTION STEPS (continued)

NOTE
Due to time constraints, the Generator Core Condition Monitor should be placed in service prior to the Purity Meter.

[93] **VERIFY** the following:

- Generator Core monitor placed in service. (3-OI-35).

  ________ Initials ________ Date ________ Time

- Main Turbine on turning gear or rolling greater than or equal to 2 RPM. (3-OI-47).

  ________ Initials ________ Date ________ Time

- Main Generator and exciter field brushes installed.

  ________ Initials ________ Date ________ Time
  Electrical Maint.

- GENERATOR 3 STOP VALVE AND LS TCO BLOCK (BT-31) in relay room on Panel RB34 for Unit 3 is installed.

  ________ Initials ________ Date ________ Time

- GEN HYDROGEN PRESSURE, 3-PI-35-17A, greater than 30 psig on Panel 3-9-8.

  ________ Initials ________ Date ________ Time

- LP steam supply valves to available RFPTs open.

  ________ Initials ________ Date ________ Time

5.0 INSTRUCTION STEPS (continued)

- Appropriate personnel on the turbine deck to sound out the turbine during rolling.

______ ______ ______
Initials Date Time

- PCB 234 AIR ABN, 3-XA-55-8D, Window 57 reset.

______ ______ ______
Initials Date Time

- GEN HYDROGEN PURITY, 3-H2I-35-12A greater than 90 percent on Panel 3-9-8.

______ ______ ______
Initials Date Time

[94] **REMOVE** Shift Manager Hold Order and **CLOSE** knife blade switches CS-1, CS-2, and CS-3, prior to rolling Main Turbine.

(R) ______ ______ ______
Initials Time Date

[95] **VERIFY** all outage work activities are dispositioned per SPP 7.2, Outage Management.

(R) ______ ______ ______
Initials Time Date
Outage and Site Scheduling Manager or Designee

(R) ______ ______ ______
Initials Time Date
Maintenance Mods Manager or Designee

5.0 INSTRUCTION STEPS (continued)

NOTE
With the feedwater heaters not in service each bypass valve is worth approximately 4%. Therefore, with 7 bypass valves open, Reactor power has the potential to exceed 25%.

[96] **WHEN** 5 to 6 turbine bypass valves are open (being careful NOT to exceed 25% Reactor power), **THEN**

[96.1] **ROLL** Turbine-Generator REFER TO 3-OI-47.

Initials ____ Date ____ Time ____

[96.2] **RAISE** speed to rated while observing Main Turbine loading limitations. REFER TO 3-OI-47.

Initials ____ Date ____ Time ____

[97] **VERIFY** MAIN TURBINE SHUTDOWN, 3-XA-55-8A, window 11, is reset.

Initials ____ Date ____ Time ____

[98] **SYNCHRONIZE** Turbine-Generator to grid and **APPLY** initial load. REFER TO 3-OI-47.

Initials ____ Date ____ Time ____

5.0 INSTRUCTION STEPS (continued)

NOTE

Steps 5.0[99]and 5.0[100] may be N/A'd during a load reduction, when the Main Turbine is removed from service and the following conditions are met:

- Main Turbine was removed from service for a short time.
- The Unit Supervisor evaluated and determines that no work has occurred on any systems or components affecting Steps 5.0[99] and 5.0[100].

[99] **COORDINATE** with Mechanical Engineering Support to **INSPECT** the Moisture Separator Room for steam leaks that would **NOT** have been detected prior to the Turbine Roll. (N/A if recovering from a load reduction and the turbine remained on line.)

Initials	Date	Time

NOTE

Step 5.0[100] may be accomplished by placing a 2 inch by 2 inch thin piece of metal or similar device over the vent hole and verifying that it is not held in place by in-leakage. Other methods may be used as directed by 3-POI-2-1.

[100] **COORDINATE** with Mechanical Engineering Support to **CHECK** steam seal regulator relief valves for in-leakage. (N/A if recovering from a load reduction and the turbine remained on line.)

Initials	Date	Time

[101] **PLACE** Feedwater Heaters and Moisture Separator Drain System in Warm-up Mode. REFER TO 3-OI-6.

Initials	Date	Time

5.0 INSTRUCTION STEPS (continued)

[102] **VERIFY** alignment of steamline drain valves for normal operation. REFER TO 3-OI-1.

Initials ______ Date ______ Time ______

CAUTION

Exceeding 150 MVARS incoming reactive load may result in slipping a generator pole during periods of low excitation.

[103] [INPO/C] **CHECK** the following parameters to ensure proper operation and **COMPARE** to indicated APRM power for agreement during any power change:

- Reactor Pressure. ☐
- Feedwater Flow. ☐
- Reactor Water Level. ☐
- Steam Flow. ☐
- Reactor Power. ☐
- Generator MW. ☐
- Core Flow. ☐
- Bypass Valve Position. [INPO SOER 90-003] ☐

Initials ______ Date ______ Time ______

[104] **MAINTAIN** Reactor power and core flow within limits of Unit 3 Power/Flow Map. **REFER TO** ICS and/or 0-TI-248, Station Reactor Engineer.

(R) Initials ______ Time ______ Date ______

5.0 INSTRUCTION STEPS (continued)

[105] **WHEN** Reactor power as indicated on APRMs is greater than 15%, but less than 25%, **THEN**

PERFORM or **VERIFY** within required periodicity

- 3-SR-3.3.2.1.1, Rod Block Monitor (RBM) Functional Test.

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

- 3-SR-3.3.2.1.4(A), Rod Block Monitor (RBM) Calibration and Functional Test.

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

- 3-SR-3.3.2.1.4(B), Rod Block Monitor (RBM) Calibration and Functional Test.

(R) ____________ ____________ ____________
Initials Time Date
[NRC/C] Work Control

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally is required to be in direct communication with the Control Room.

[106] **WHEN** total steam flow exceeds 19%, **THEN**

PERFORM the following:

[106.1] **VERIFY** Condensate System flow being maintained within the limits of 3-OI-2 using CNDS FLOW CONTROL SHORT CYCLE, 3-FC-2-29, on Panel 3-9-6, in AUTO/BAL.

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[106.2] **VERIFY** charcoal adsorbers are in service. REFER TO 3-OI-66.

(R) ____________ ____________ ____________
Initials Time Date

[106.3] Prior to reaching 25% power, **VERIFY** 3-SR-3.3.2.2.4, Reactor Feedwater and Main Turbine High Water Level Trip Logic System Functional Test is completed.

(R) ____________ ____________ ____________
Initials Time Date

[107] **BEFORE** exceeding 25% power (0-TI-248, Station Reactor Engineer, and Illustration 1), **PERFORM** the following:

[107.1] **NOTIFY** Radiation Protection that an RPHP is in effect for reactor power approaching 25%. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

[107.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7, SOER 01-1, BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

[107.3] **VERIFY** the following:

[107.3.1] All operable Main Steam Isolation Valves open.

(R) ____________ ____________ ____________
Initials Time Date

[107.3.2] Reactor Feedwater Temperature greater than 160°F.

(R) ____________ ____________ ____________
Initials Time Date

[107.3.3] Restart of Core Monitoring System on ICS.

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

5.0 INSTRUCTION STEPS (continued)

[107.3.4] Core Monitoring Software available.

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

NOTE
Steps 5.0[107.3.5]and 5.0[107.3.6]are performed to ensure that all required data is obtained prior to mode change per LCO 3.0.4 and SR-3.0.4.

[107.3.5] 3-SR-2 data for 25% RTP has been performed.

____________ ____________ ____________
Initials Date Time

NOTE
Step 5.0[107.3.6] SHALL be verified by the STA.

[107.3.6] All 3-SR-2 data meets the requirements for exceeding 25% Reactor power per LCO 3.0.4 and SR-3.0.4.

____________ ____________ ____________
Initials Date Time
STA

[108] **PERFORM** 3-SR-3.3.2.1.5, Verification of RWM Automatic Bypass Setpoint. (N/A if not required)

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

[109] **WHEN** Reactor power is greater than 25%, **THEN**

[109.1] **PERFORM** 3-SR-3.3.1.1.2, APRM Output Signal Adjustment. (not required to be performed until 12 hours after thermal power greater than or equal to 25% RTP.)

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[109.2] **VERIFY** Thermal Limits are set to meet the following requirements:

- COLR
- Tech Specs 3.2
- Administrative limits as required for Feedwater Temperature

(R) ____________ ____________ ____________
Initials Time Date
Reactor Engineer

[109.3] **VERIFY** the Main Turbine Bypass system operable per Tech Specs 3.7.5.

(R) ____________ ____________ ____________
Initials Time Date

[109.4] **VERIFY** RFPT and Main Turbine High Water Level Trip OPERABLE per Tech Specs 3.3.2.2.

(R) ____________ ____________ ____________
Initials Time Date

[110] **PLACE** Hydrogen Water Chemistry System in service. REFER TO 3-OI-4. (N/A if system is unavailable or not required to be in service.)

____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

NOTES

1) Verification of control rod following the drive by observing a response in the nuclear instrumentation is required each time a control rod is moved.
2) Thermal power changes of 15% of rated power or more occurring within one hour requires Chemistry be notified to determine if sampling in accordance with Tech Specs 3.4.6 and Technical Requirements Manual 3.4.1 is required.

[111] **VERIFY CLOSED** all TURBINE BYPASS valves prior to exceeding 30% Reactor power.

(R) ___________ ___________ ___________
Initials Time Date

[112] **CONTINUE** control rod withdrawals in combination with core flow changes, as recommended by Reactor Engineer, until approximately 30% Reactor power.

___________ ___________ ___________
Initials Date Time

[113] **PERFORM** a Recombiner performance evaluation. REFER TO 3-OI-66.

___________ ___________ ___________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

MODE/CONDITION CHANGE

NOTE

Per Unit 3 TRM COLR the CPR limits and off-rated corrections are provided for Recirculation Pump Trip out-of-service and/or Turbine Bypass out-of-service conditions. These events are analyzed for separate and concurrent for operability.

[114] **WHEN** Reactor power exceeds 30%, **THEN**

[114.1] **TRANSFER** Reactor Feedwater Control System to three-element control. REFER TO 3-OI-3.

_______ _______ _______
Initials Date Time

[114.2] **VERIFY** the EOC-RPT Trips are operable per Tech Specs 3.3.4.1. (N/A if disabled per 3-OI-68)

(R) _______ _______ _______
Initials Time Date

[114.3] **VERIFY** annunciator TURB CV FAST CLOSURE TURB SV CLOSURE SCRAM/RPT TRIP LOGIC BYPASS, 3-XA-55-5B, window 16 resets.

(R) _______ _______ _______
Initials Time Date

[114.4] **VERIFY** that the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low scrams are OPERABLE per Tech Specs 3.3.1.1

(R) _______ _______ _______
Initials Time Date

[114.5] **VERIFY** Turbine First Stage Pressure Permissive pressure switches operable. REFER TO TRM 3.3.1.

(R) _______ _______ _______
Initials Time Date

[115] **VERIFY** Generator Hydrogen purity greater than 97%. REFER TO 3-OI-35.

_______ _______ _______
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[116] **VERIFY** Generator Hydrogen pressure in the pressure band required in 3-OI-35.

Initials ______ Date ______ Time ______

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[117] **PLACE** additional condensate demineralizers in service as required to support starting a second Reactor Feedpump. REFER TO 3-OI-2A.

Initials ______ Date ______ Time ______

CAUTION

Placing a second Reactor Feed pump in service prior to 30% power or 4×10^{6} lbm/hr feedwater flow may cause fluctuations in Feedwater Level Control System.

[118] **PLACE** a second Reactor Feedpump in service. REFER TO 3-OI-3.

Initials ______ Date ______ Time ______

[119] **BEGIN** warming the third Reactor Feedpump. REFER TO 3-OI-3.

Initials ______ Date ______ Time ______

[120] **VERIFY** Condensate System flow being maintained within the limits of 3-OI-2 using CNDS FLOW CONTROL SHORT CYCLE, 3-FC-2-29, on Panel 3-9-6, in AUTO/BAL.

Initials ______ Date ______ Time ______

5.0 INSTRUCTION STEPS (continued)

NOTE
Step 5.0[121] is to be performed prior to placing feedwater heaters in service.

[121] **VERIFY** all outage work activities are dispositioned in accordance with SPP-7.2, Outage Management.

(R) ____________ ____________ ____________
Initials Time Date
Outage and Site Scheduling Manager or Designee

(R) ____________ ____________ ____________
Initials Time Date
Maintenance Mods Manager or Designee

[122] **BEFORE** placing feedwater heaters and moisture separators in service, **PERFORM** the following:

[122.1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to place Feedwater Heaters and Moisture Separators in service. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

[122.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7, SOER 01-1, BFN PER 126211]

(R) ____________ ____________ ____________
Initials Date Time

5.0 INSTRUCTION STEPS (continued)

[123] **WHEN** reactor power is approximately 40%, **THEN**

BEGIN placing Feedwater Heaters and Moisture Separator Drain System in service. REFER TO 3-OI-6.

[123.1] **WHEN** all Feedwater Heaters and Moisture Separator Drain System are in service, **THEN**

At AW-51, **VERIFY** that the Feedwater Heater alarms are **NOT** bypassed. REFER TO 3-OI-6.

__________ Initials __________ Date __________ Time

[124] **WHEN** reactor power is approximately 45%, but **BEFORE** 50% reactor power, **THEN**

[124.1] **NOTIFY** Radiation Protection that an RPHP is in effect for reactor power approaching 50%. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) __________ Initials __________ Date __________ Time

[124.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7, SOER 01-1, BFN PER 126211]

(R) __________ Initials __________ Date __________ Time

[125] **REQUEST** the Unit Operator to frequently Monitor the Power to Flow Map on ICS and/or 0-TI-248, Station Reactor Engineer, during power ascension.

AND

TAKE actions as appropriate.

__________ Initials __________ Date __________ Time

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[126] **VERIFY** ALL high radiation areas required to be locked are locked or posted. REFER TO RCI-17.

Initials _______ Date _______ Time _______

[127] **PLACE** additional condensate demineralizers in service to support starting third Condensate Pump, Condensate Booster Pump, and Reactor Feedpump. REFER TO 3-OI-2A.

Initials _______ Date _______ Time _______

[128] **START** third Condensate and Condensate Booster Pump. REFER TO 3-OI-2.

Initials _______ Date _______ Time _______

[129] **VERIFY** Condensate System flow is being maintained within the limits of 3-OI-2 using CNDS FLOW CONTROL SHORT CYCLE, 3-FC-2-29, on Panel 3-9-6, in AUTO/BAL.

Initials _______ Date _______ Time _______

[130] **PLACE** third Reactor Feedpump in service. REFER TO 3-OI-3.

Initials _______ Date _______ Time _______

5.0 INSTRUCTION STEPS (continued)

NOTES
1) Average thermal power for an 8-hour period is limited to 3458 MWt.
2) Exceeding a thermal power of 3526 MWt under any conditions is unacceptable.

[131] **IF** heat balance indicates a thermal power greater than 3458 MWt, **THEN** **PERFORM** the following:

- **REDUCE** Reactor power to 3458 MWt or less using Reactor Recirc flow. ☐
- **CHECK** average CMWT. ☐
- **NOTIFY** Reactor Engineer at Shift Manager direction. ☐

_______ Initials _______ Date _______ Time

[132] **VERIFY** Drywell and Torus are N_2 inerted within 24 hours of entering LCO in Step 5.0[84] (15% RTP)

_______ Initials _______ Date _______ Time

[133] **VERIFY** Drywell to Torus differential pressure is greater than or equal to 1.1 psig within 24 hours of entering LCO in Step 5.0[84]. (15% RTP)

_______ Initials _______ Date _______ Time

[134] **VERIFY** 3-SI-4.7.A.2.a, Primary Containment Nitrogen Consumption and Leakage has been commenced.

_______ Initials _______ Date _______ Time

5.0 INSTRUCTION STEPS (continued)

[135] **CONTINUE** control rod withdrawal in combination with core flow changes, as recommended by the Reactor Engineer, while monitoring Core Thermal Limits (Illustration 1), until desired power level is reached.

(R) __________ __________ __________

Initials Time Date

[136] [NRC/C] **WHEN** the plant is operating at rated thermal power or Maximum Obtainable Load, **THEN**

VERIFY CV POSITION LIMIT, 3-XI-47-157, is set at approximately 66. REFER TO 3-OI-47. [GE SIL 589]

__________ __________ __________

Initials Date Time

5.0 INSTRUCTION STEPS (continued)

	NAME (print)	INITIALS
Performed by:	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
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	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
	______________________	__________
Reviewed by:	______________________	__________
	Shift Manager	Date

5.0 INSTRUCTION STEPS (continued)

REMARKS:

Illustration 1
(Page 1 of 1)

Reactor Thermal Limits

Administrative Reactor Thermal Limits for MFLPD, MFLCPR, MAPRAT, and CTP (MWt) are listed in 0-TI-248, Appendix for Administrative Limits. These limits should be reviewed with Reactor Engineer.

Monitoring of core thermal limits at the following frequencies is recommended:

A. During startups as recommended by the Reactor Engineer using 0-TI-248, Appendix for Core Thermal Limits Monitoring.

B. Following completion of planned power rise with control rods or recirc flow.

C. Following any unexpected power change.

D. Once every two hours during steady state operation.

If core monitoring software becomes unavailable, the Shift Manager and Reactor Engineer will determine the appropriate frequency for monitoring core thermal limits using the backup core monitoring computer taking into consideration current core conditions and margin to thermal limits. Power changes should not normally be made without the core monitoring software being available.

Maximum steady-state power averaged over 8 hours is 3458 MWt. However, the Reactor should not be operated such that the steady state power (as indicated by 30 min avg, 1 hr avg, or 2 hr avg) is above 3458 MWt

Minor variations in process parameter inputs to the process computer may result in individual edits or indications above 3458 MWt while true steady-state core thermal power is ≤3458. Normal variation is within 5 MWt of steady-state core thermal power. Running averages (from core thermal power summary on the nuclear heat balance display) are not as sensitive. The following guidance is provided:

RESULT (MWt)	GUIDANCE
> 3463	REDUCE power.
3458 to 3463	ALLOW time for recent perturbations to settle. EVALUATE trend. IF the trend indicates steady state core thermal power will be above 3458, THEN REDUCE power. EVALUATE trend.
> 3458 (any running avg)	REDUCE power.

Illustration 2
(Page 1 of 1)

Core Quadrants/Octants

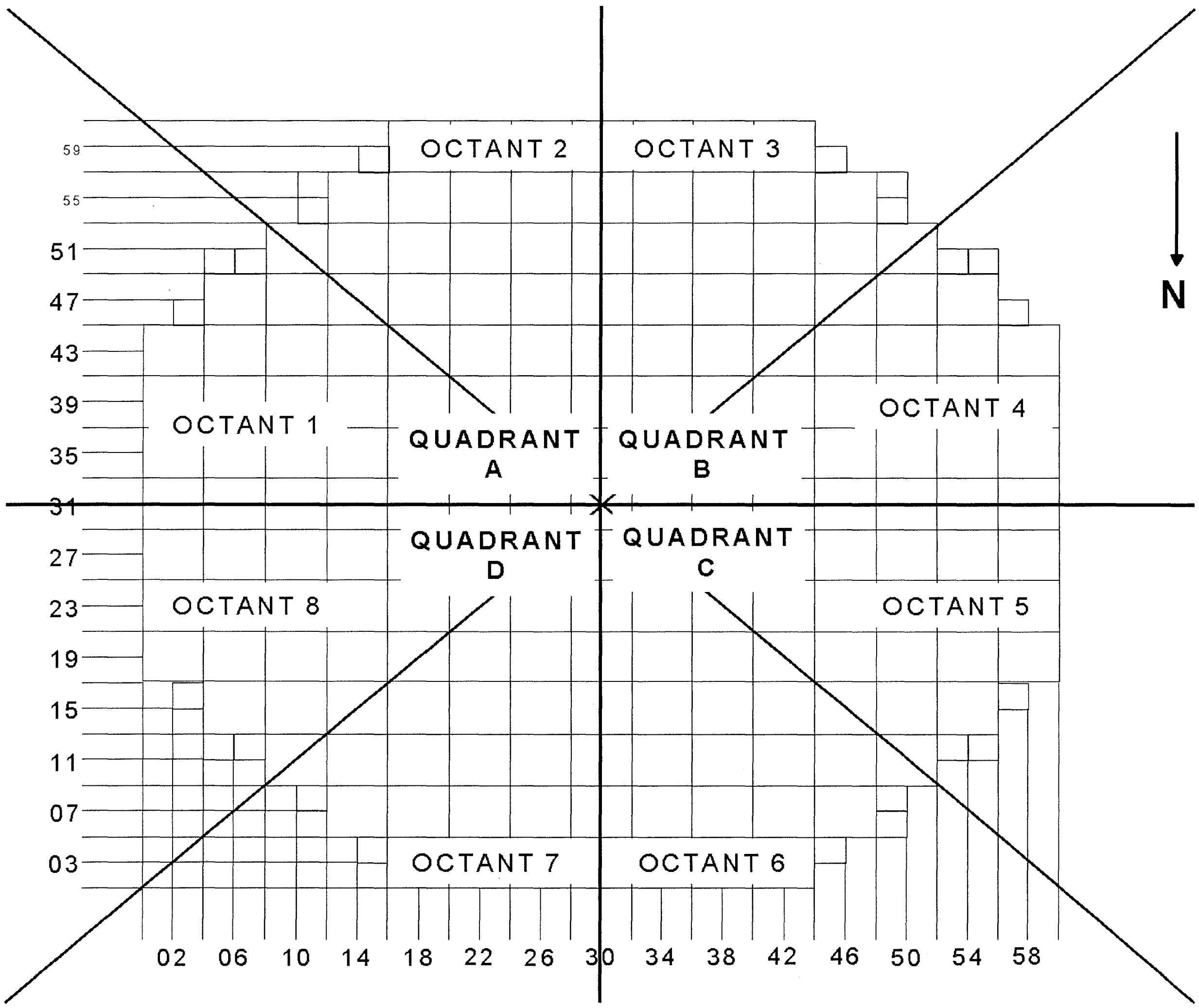


Illustration 3
(Page 1 of 1)

Reactor Vessel Heatup Graph

NOTE

The slope of the diagonal lines represents a 90 degree per hour heatup rate.

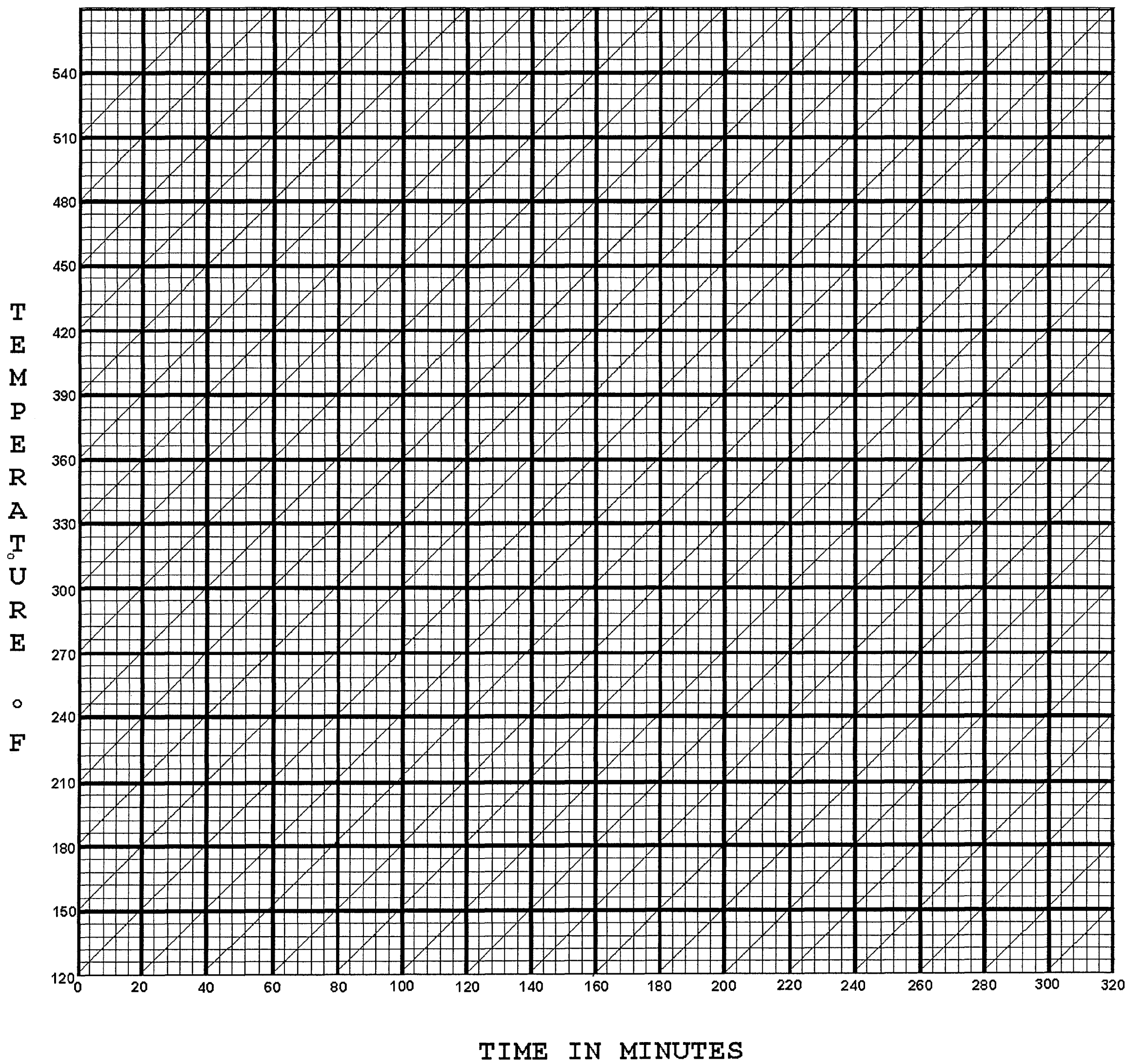


Illustration 4
(Page 1 of 2)

Percent Power vs. Time
(To obtain 4000 MWt-minutes)

IRM RANGE	SCALE FACTOR	PERCENT RATED POWER
10	0.39	0-48.75
9	0.1248	0-15.6
8	0.039	0-4.875
7	0.01248	0-1.56
6	0.0039	0-0.4875
5	0.001248	0-0.156
4	0.00039	0-0.04875
3	0.0001248	0-0.0156
2	0.000039	0-0.004875
1	0.00001248	0-0.00156

PERCENT POWER = (IRM READING)(SCALE FACTOR)

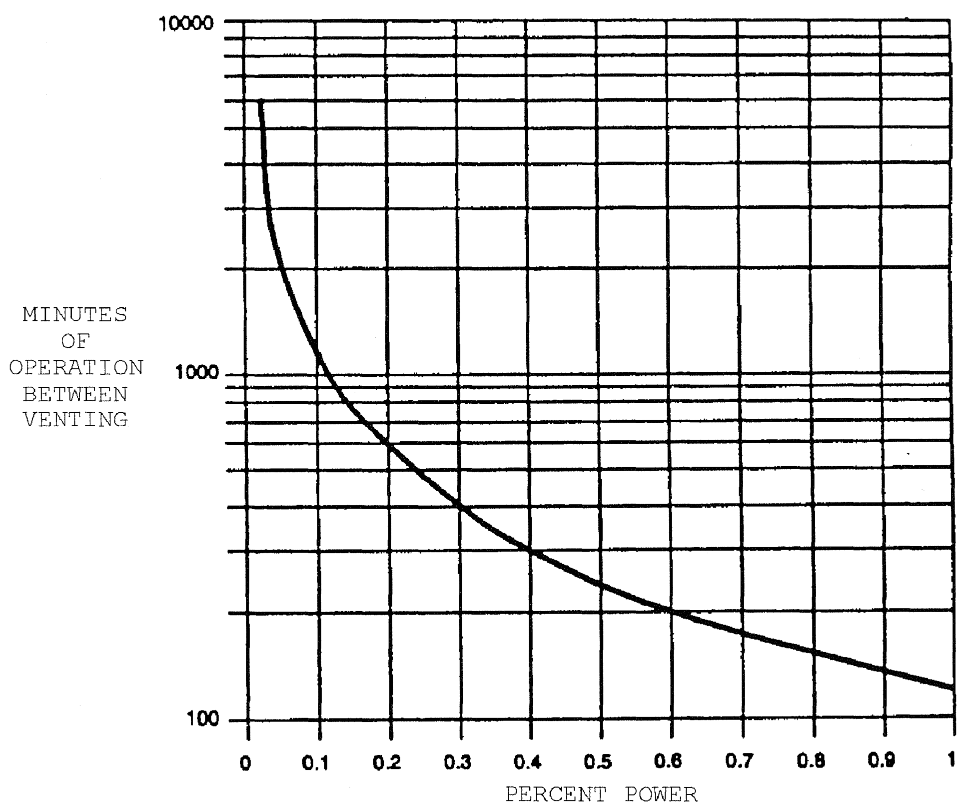
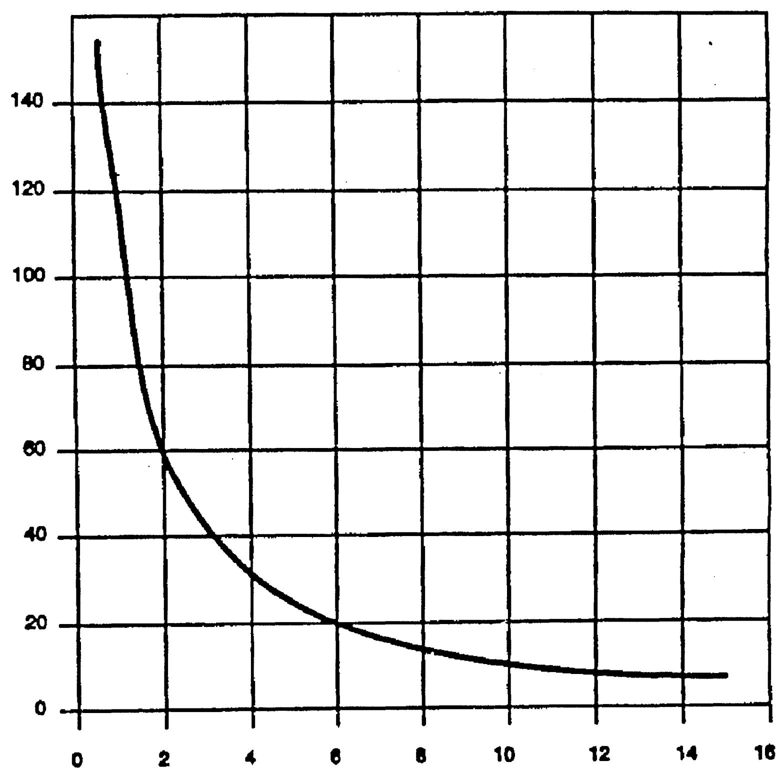


Illustration 4
(Page 2 of 2)

Percent Power vs. Time
(To obtain 4000 MWt-minutes)

IRM RANGE	SCALE FACTOR	PERCENT RATED POWER
10	0.39	0-48.75
9	0.1248	0-15.6
8	0.039	0-4.875
7	0.01248	0-1.56
6	0.0039	0-0.4875
5	0.001248	0-0.156
4	0.00039	0-0.04875
3	0.0001248	0-0.0156
2	0.000039	0-0.004875
1	0.00001248	0-0.00156

PERCENT POWER = (IRM READING)(SCALE FACTOR)



Attachment 1
(Page 1 of 1)

Environmentally Qualified Barrier Doors

VERIFY the following doors meet the requirements of 0-GOI-300-5, Environmentally Qualified Doors prior to exceeding 200°F:

DOOR NO.	LOCATION	INITIALS
36/41	El 519' between Unit 2 and Unit 3 Core Spray pump rooms	________
37/40	El 519' between Unit 2 and Unit 3 RHR pump rooms	________
44/45	El 541' between Unit 2 and Unit 3 RHR pump rooms	________
253	El 565' TIP Room door	________
505	El 593' RWCU Heat Exchanger Room (SW)	________
508	El 593' RWCU Pump Room 3A	________
509	El 593' RWCU Pump Room 3B	________
512	El 593' RWCU Heat Exchanger Room (NW)	________
513/514	El 593' Emergency exit lock - Electric boardroom 3B	________
657/658	El 621' Emergency exit lock - Electric boardroom 3A	________
250/251	El 565' Equipment Lock between Unit 3 Reactor Bldg and Turbine Bldg	________
244/248/249	El 565' Personnel access between Unit 2 and Unit 3	________

COMMENTS: ____

Attachment 2
(Page 1 of 4)

Temperature Verifications from Cold Shutdown to 212°F

NOTES

1) Lower Reactor coolant temperatures yield higher concentrations of oxygen (O_2) and hydrogen peroxide (H_2O_2). O_2 and H_2O_2 when combined with heat and stress promote intergranular stress cracking and corrosion. Therefore, heat-up is limited to less than or equal to 50°F/hr until Reactor Recirc loop water temperatures reach 215°F as indicated on RECIRC PUMPS DISCH TEMP, 3-TR-68-2.
2) This attachment can be performed in any order as long as the steps are completed prior to required temperature.
3) Attachment 1 must be performed prior to exceeding 200°F . The signoff for this is in the Prestartup Checklist Step 4.1[124]and Step 1.0[3] of this attachment.[PER 120826]

1.0 TEMPERATURE VERIFICATION FROM COLD SHUTDOWN TO 212°F

[1] **INITIATE** 3-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring, using a licensed unit operator, at least 15 minutes prior to heatup and pressurization. Copies of Illustration 3 should be used to plot heatup rate. (N/A, if performing a startup not requiring heatup.)

(R) ____________ ____________ ____________
Initials Time Date

NOTES

1) Step 1.0[2] should be performed as soon as practical. This will ensures that all required 3-SR-2 data is obtained to allow a mode/condition change per LCO 3.0.4 and SR-3.0.4.
2) Step 1.0[2] SHALL be verified by the STA.

[2] **VERIFY** all 3-SR-2 data meets the requirements for exceeding 212°F Reactor Coolant Temperature. (N/A if the Reactor Startup is being performed greater than 212°F.)

____________ ____________ ____________
Initials STA Date Time

Attachment 2
(Page 2 of 4)

Temperature Verifications from Cold Shutdown to 212°F

1.0 TEMPERATURE VERIFICATION FROM COLD SHUTDOWN TO 212°F (continued)

[3] **PERFORM** Attachment 1 prior to exceeding 200°F to verify EQ doors in proper position.

(R) ____________ ____________ ____________
Initials Time Date

NOTES

1) Valves in this step may already be in the required position due to plant conditions, e.g. MODE 3. In this case, Step 1.0[4] of this attachment verifies stated conditions.
2) When CRD is the only source of makeup, steam drains may be closed as required to maintain Reactor vessel level during heatup.

[4] **PERFORM** the following prior to reaching 210°F Reactor Coolant Temperature. (N/A Section 1.0[4] of this attachment if the Reacror Startup is being performed greater than 212°F.)

[4.1] **VERIFY OPEN** the following valves on Panel 3-9-7:

- 3-FCV-6-100 using STOP VALVE 1 BEFORE SEAT DR VLV, 3-HS-6-100A.

____________ ____________ ____________
Initials Date Time

- 3-FCV-6-101 using STOP VALVE 2 BEFORE SEAT DR VLV, 3-HS-6-101A.

____________ ____________ ____________
Initials Date Time

- 3-FCV-6-102 using STOP VALVE 3 BEFORE SEAT DR VLV, 3-HS-6-102A.

____________ ____________ ____________
Initials Date Time

Attachment 2
(Page 3 of 4)

Temperature Verifications from Cold Shutdown to 212°F

1.0 TEMPERATURE VERIFICATION FROM COLD SHUTDOWN TO 212°F (continued)

- 1-FCV-6-103 using STOP VALVE 4 BEFORE SEAT DR VLV, 3-HS-6-103A.

________ Initials ________ Date ________ Time

[4.2] **VERIFY CLOSED** the following valves on Panel 3-9-3:

- 3-FCV-3-98 using RPV HEAD VENT INBD VALVE, 3-HS-3-98A.

(R) ________ Initials ________ Time ________ Date

- 3-FCV-3-99 using RPV HEAD VENT OUTBD VALVE, 3-HS-3-99A.

(R) ________ Initials ________ Time ________ Date

[4.3] **IF** Drywell entry was performed, **THEN**:

[4.3.1] At Unit Supervisor discretion, **VERIFY** RPV HEAD VENT SHUTOFF VLV, 3-SHV-010-0100, is either LOCKED OPEN or LOCKED CLOSED (DW EL. 563' NW Side along biological shield.).

(R) ________ Initials ________ Time 1ST ________ Date

(R) ________ Initials ________ Time 2ND ________ Date

[4.3.2] **RECORD** position of RPV HEAD VENT SHUTOFF VLV, 3-SHV-010-0100, in narrative log.

(R) ________ Initials ________ Time ________ Date

Attachment 2
(Page 4 of 4)

Temperature Verifications from Cold Shutdown to 212°F

1.0 TEMPERATURE VERIFICATION FROM COLD SHUTDOWN TO 212°F (continued)

[4.4] [TOE 0-97-064-0823] **IF** Containment Inerting is in progress, **THEN**

VERIFY Drywell and Suppression Chamber are **NOT** being Inerted in parallel. (Otherwise N/A)

(R) ____________ ____________ ____________
Initials Time Date

[4.5] **IF** performing initial startup after a refueling outage, **THEN**

VERIFY 3-SR-3.1.1.1, Reactivity Margin Test, is complete. (Otherwise N/A.)

(R) ____________ ____________ ____________
Initials Time Date

[5] **WHEN** moderator temperature is approximately 212°F, **THEN**

NOTIFY Chemistry to commence startup of the Durability Monitor.

____________ ____________ ____________
Initials Date Time

Attachment 3
(Page 1 of 21)

Startup with MSIV's Closed

NOTE
When Reactor water temperature is greater than 215°F, heatup is limited to 90°F/Hr.

1.0 REACTOR STARTUP WITH MSIV'S CLOSED

[1] **IF** Reactor coolant oxygen content can not be maintained less than 300 ppb when coolant temperature is greater than 285°F, **THEN**

PERFORM the following: (N/A if less than 300 ppb.)

[1.1] **SHUT DOWN** Reactor. REFER TO 3-GOI-100-12A.

(R) ____________ ____________ ____________
Initials Time Date

CAUTION
During Reactor Cooldown, the optimum rate is 20°F every 15 minutes. This will ensure the administrative limit of 90°F/Hr is not exceeded. Do not attempt to "makeup" for time intervals which fall short of 20°.

[1.2] **COOL DOWN** Reactor at a rate not to exceed 90°F/hr.

(R) ____________ ____________ ____________
Initials Time Date

[1.3] **REQUEST** Chemistry to sample for dissolved oxygen every 4 hours until MODE 4 is achieved.

(R) ____________ ____________ ____________
Initials Time Date

[2] **IF** MSIV's will be opened prior to 215°F, **THEN** (Otherwise N/A)

RECOMMENCE this procedure at Step 5.0[42].

____________ ____________ ____________
Initials Date Time

Attachment 3
(Page 2 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

NOTES

1) If RCIC will be used for Reactor Vessel Level/Pressure Control, N/A Step 1.0[3] in this attachment.
2) Step 1.0[3] of this attachment is performed more than once if prolonged operation in Hot Standby without RCIC in operation is anticipated.

[3] **WHEN** extended operation in Hot Standby with MSIVs closed is anticipated, **THEN**

VENT Reactor every 4000 MWt minutes as follows: (**REFER TO** Illustration 4 to determine time interval between venting.)

[3.1] **VERIFY** Reactor Feedpump Turbines are on turning gear. REFER TO 3-OI-3.

Initials	Date	Time

[3.2] **VERIFY** Main Turbine on turning gear. REFER TO 3-OI-47.

Initials	Date	Time

[3.3] **VERIFY** Auxiliary Boilers in service. REFER TO 0-OI-12.

Initials	Date	Time

[3.4] **VERIFY** Steam Seals on Main Turbine and Reactor Feedpump Turbines. REFER TO 3-OI-47C.

Initials	Date	Time

[3.5] **VERIFY** condenser vacuum established. REFER TO 3-OI-66.

Initials	Date	Time

Attachment 3
(Page 3 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[3.6] **OPEN** the following valves to vent the Reactor on Panel 3-9-3:

- 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A.

_______ Initials _______ Date _______ Time

- 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A.

_______ Initials _______ Date _______ Time

- 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A.

_______ Initials _______ Date _______ Time

[3.7] **CONTINUE** venting until at least 4000 cubic feet of steam has been released. (Times are approximate for the pressures indicated.)

Reactor Pressure (psig)	Vent Time (minutes)
1000#	15
900#	16
800#	17
700#	18
600#	20
500#	22
400#	24
300#	28
200#	34
100#	48

_______ Initials _______ Date _______ Time

Attachment 3
(Page 4 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[3.8] **WHEN** venting is complete, **THEN**

PERFORM the following:

[3.8.1] **BREAK** condenser vacuum. REFER TO 3-OI-66. (N/A if desired to maintain vacuum).

_______ Initials _______ Date _______ Time

[3.8.2] **SHUT DOWN** Auxiliary Boilers. REFER TO 0-OI-12. (N/A if desired to leave Auxiliary Boilers in operation).

_______ Initials _______ Date _______ Time

[3.8.3] **CLOSE** the following valves on panel 3-9-3:

- 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A.

_______ Initials _______ Date _______ Time

- 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A.

_______ Initials _______ Date _______ Time

- 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A.

_______ Initials _______ Date _______ Time

Attachment 3
(Page 5 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

NOTE

Step 1.0[4] may be marked N/A if Reactor pressure was not lowered below the RCIC low pressure isolation setpoint during the unit shut down.

[4] **WHEN** PRESS A, 3-PI-3-54, on Panel 3-9-5, indicates approximately 70 psig, **THEN**

PERFORM the following:

[4.1] **VERIFY RESET** RCIC steam line low pressure isolation. REFER TO 3-OI-71.

(R) ____________ ____________ ____________
Initials Time Date

[4.2] **WARM** and **PRESSURIZE** RCIC steam line. (N/A if already performed.) REFER TO 3-OI-71.

(R) ____________ ____________ ____________
Initials Time Date

[4.3] **VERIFY** RCIC in Prestartup/Standby Readiness. REFER TO 3-OI-71.

(R) ____________ ____________ ____________
Initials Time Date

Attachment 3
(Page 6 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

CAUTIONS

1) RWCU blowdown is limited to maintain WATER TO RWCU DEMINS, 3-XS-69-6 point 3, temperature less than 130°F, as indicated by RWCU SYSTEM TEMPERATURES, 3-TI-69-6, located on Panel 3-9-4.

2) At Reactor vessel pressures less than rated, as much as a 13 inch level discrepancy can exist between the 3-LI-3-208A(B)(C)(D) uncompensated narrow range level instruments and the 3-LI-3-53(60)(206)(253) compensated narrow range level instruments. The 3-LI-3-208A(B)(C)(D) level instruments are not temperature compensated and the lower the pressure on the Reactor vessel, the higher the 3-LI-3-208A(B)(C)(D) level instruments will read. Failure to maintain the RPV level indicated on the 3-LI-3-208A(B)(C)(D) level instruments less than 48 inches can result in unnecessary turbine trips (i.e., RFPTs, HPCI, RCIC, and Main Turbine).

[5] **MAINTAIN** Reactor water level between 28 and 38 inches as indicated by RX LVL (Red Pen) on RX VESSEL LEVEL/TOTAL FW FLOW recorder, 3-XR-3-53, AND less than 48" on 3-LI-3-208A(B)(C)(D), using the following vessel makeup and level control systems:

- CRD System (40 to 65 gpm). (Control Rod Drive Hydraulic System Startup section of 3-OI-85).
- CRD System (up to 80 gpm). (CRD Pump Operation at Elevated Flow section of 3-OI-85).
- RWCU System. (3-OI-69).
- Condensate System. (3-OI-2).

(R) __________ __________ __________

Initials Time Date

Attachment 3
(Page 7 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

> **NOTE**
>
> Step 1.0[6] may be marked N/A if Reactor pressure was not lowered below the HPCI low Pressure isolation setpoint during the unit shut down.

[6] **WHEN** PRESS A, 3-PI-3-54, indicates greater than 110 psig, **THEN**

[6.1] **VERIFY RESET** HPCI steam line low pressure isolation. REFER TO 3-OI-73.

(R) ____________ ____________ ____________
Initials Time Date

[6.2] **WARM** and **PRESSURIZE** HPCI steam line. REFER TO 3-OI-73. (N/A if previously performed.)

(R) ____________ ____________ ____________
Initials Time Date

[6.3] **VERIFY** HPCI in Prestartup/standby Readiness. REFER TO 3-OI-73.

(R) ____________ ____________ ____________
Initials Time Date

[7] **VENT** the drywell, as necessary, to maintain drywell pressure less than 1.33 psig. REFER TO 3-OI-64.

____________ ____________ ____________
Initials Date Time

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

MODE/CONDITION CHANGE

CAUTION

[II/F] Prior to initiating any event which adds, or has the potential to add, heat energy to the suppression chamber, the Unit Supervisor will evaluate the necessity of placing suppression pool cooling in service. This is due to the potential for developing thermal stagnation during sustained heat additions. [II-B-91-129]

NOTES

1) If not previously performed, HPCI and RCIC must be proven operable within 12 hours after reaching 150 psig Reactor pressure.
2) Step 1.0[8] is performed to ensure HPCI and RCIC are proven operable prior to exceeding shutoff head of RHR and Core spray pumps.

[8] **WHEN** Reactor pressure is greater than 150 psig, **THEN**

PERFORM the following:

[8.1] **RECORD** Time LCO entered. (N/A, if no LCO entry is required.)

Date ___________ Time ___________

(R) ___________ ___________ ___________
Initials Time Date

[8.2] **VERIFY** the following are complete for the current operating cycle prior to exceeding 165 psig. (N/A if not Required:

- 3-SR-3.5.3.4, RCIC System Rated Flow at Low RPV Pressure.

(R) ___________ ___________ ___________
Initials Time Date

- 3-SR-3.5.1.8, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at 150 psig Reactor pressure.

(R) ___________ ___________ ___________
Initials Time Date

Attachment 3
(Page 9 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[8.3] **VERIFY** HPCI and RCIC are operable within 12 hours of entering LCO in Step 1.0[8.1] per Tech Specs 3.5.1 and 3.5.3, respectively and enter in NOMS Narrative Log.

(R) ____________ ____________ ____________
Initials Time Date

[8.4] **VERIFY** operability of EHC Control System by allowing a bypass valve to THROTTLE OPEN.

(R) ____________ ____________ ____________
Initials Time Date

CAUTIONS

1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[9] **ESTABLISH** level control with RFW START-UP LEVEL CONTROL, 3-LIC-3-53 on Panel 3-9-5. REFER TO 3-OI-3.

____________ ____________ ____________
Initials Date Time

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[10] **WHEN** Reactor Water Cleanup Blowdown Operation is no longer required for vessel level control, **THEN**

STOP Reactor Water Cleanup Blowdown Operation. REFER TO 3-OI-69.

_____ Initials _____ Date _____ Time

CAUTIONS

1) Operation of the RCIC Turbine below 2100 RPM may result in turbine damage.
2) [NER/C] Extended RCIC system operation may raise Suppression Chamber O_2 concentration above Tech Specs limits because of air in-leakage from RCIC Turbine Gland Seal System. [GE SIL 548]

[11] **IF** CRD flow will be inadequate when Reactor pressure is too high for Condensate System, **THEN**

PERFORM the following as necessary:

[11.1] **RAISE** CRD flow to a maximum of 80 gpm if **NOT** already performed. REFER TO 3-OI-85, CRD Pump Operation at Elevated Flow section.

_____ Initials _____ Date _____ Time

[11.2] **START** RCIC on CST-TO-CST Flow path. REFER TO 3-OI-71.

_____ Initials _____ Date _____ Time

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

CAUTIONS

1) While in Mode 2, 950 psig Reactor pressure and 1% Reactor power is the maximum limit when the unit is dependent on RCIC for level control.
2) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
3) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[12] **CONTINUE** to withdraw control rods to raise Reactor power and pressure to the levels directed by the Unit Supervisor.

Initials _______ Date _______ Time _______

[13] **WHEN** the following conditions exist:

- Condensate System cannot inject into Reactor vessel,

  AND

- CRD flow cannot maintain Reactor vessel water level.

THEN

PERFORM the following:

[13.1] **VERIFY** RCIC on CST-TO-CST Flow path. REFER TO 3-OI-71.

Initials _______ Date _______ Time _______

[13.2] **OPEN** 3-FCV-71-39 using RCIC PUMP INJECTION VALVE, 3-HS-71-39A.

Initials _______ Date _______ Time _______

Attachment 3
(Page 12 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[13.3] **SLOWLY THROTTLE CLOSE** 3-FCV-71-38 using RCIC PUMP CST TEST VLV, 3-HS-71-38A, until flow is established to Reactor vessel and water level is stable.

________	________	________
Initials	Date	Time

[14] **WHEN** desired power level or pressure is reached, **THEN**

PERFORM the following:

[14.1] **STABILIZE** Reactor vessel level by throttling 3-FCV-71-38 using RCIC PUMP CST TEST VLV, 3-HS-71-38A.

________	________	________
Initials	Date	Time

[14.2] **MANIPULATE** control rods to maintain desired power level.

________	________	________
Initials	Date	Time

[14.3] **STABILIZE** Reactor pressure using control rods and by throttling 3-FCV-71-38 using RCIC PUMP CST TEST VLV, 3-HS-71-38A.

________	________	________
Initials	Date	Time

[15] **PERFORM** any Surveillances required in MODE 2.

	________	________	________
(R)	Initials	Time	Date
		[NRC/C] Work Control	

	________	________	________
(R)	Initials	Time	Date
		Unit Supervisor	

Attachment 3
(Page 13 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[16] **WHEN** ready to continue startup to full power, **THEN**

[16.1] **IF** MSIV's are closed, **THEN**

OPEN MSIV's. REFER TO 3-OI-1.

_____________ Initials _____________ Date _____________ Time

[16.2] **CONTINUE** at Step 1.0[17] of this attachment.

_____________ Initials _____________ Date _____________ Time

NOTES

1) When Reactor water temperature is greater than 215°F, heatup is limited to 90°F/Hr.
2) Main turbine shell warming or chest warming may be performed concurrently with pressurizing the Reactor provided it is accomplished prior to exceeding 350 psig. If additional shell warming or chest warming is desired after exceeding 350 psig, it may only be conducted parallel to raising Reactor pressure to rated, with the approval of OPS Superintendent/OPS Manager. IF the CRD system cannot maintain inventory, then shell warming or chest warming is resumed after placing the first Reactor Feedpump in service.

[17] **IF** EHC is available, **THEN**

BEGIN shell warming high pressure turbine at the Unit Supervisor's discretion. REFER TO 3-OI-47. (N/A if not performed at this time.)

_____________ Initials _____________ Date _____________ Time

Attachment 3
(Page 14 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[18] **IF** EHC is available, and chest warming is required, **THEN**

COMMENCE Chest warming, if desired, when shell warming is complete. (N/A if not performed at this time.)

__________ Initials __________ Date __________ Time

CAUTIONS

1) When the pressure control swaps from "HEADER PRESSURE CONTROL" to "REACTOR PRESSURE CONTROL" the pressure set will be actual Reactor pressure at the time the swap is done, regardless of any previously raised Reactor pressure set done during a Reactor startup.
2) At Reactor vessel pressures less than rated, as much as a 13 inch level discrepancy can exist between the 3-LI-3-208A(B)(C)(D) uncompensated narrow range level instruments and the 3-LI-3-53(60)(206)(253) compensated narrow range level instruments. The 3-LI-3-208A(B)(C)(D) level instruments are not temperature compensated and the lower the pressure on the Reactor vessel, the higher the 3-LI-3-208A(B)(C)(D) level instruments will read. Failure to maintain the RPV level indicated on the 3-LI-3-208A(B)(C)(D) level instruments less than 48 inches can result in unnecessary turbine trips (i.e., RFPTs, HPCI, RCIC, and Main Turbine).

[19] **CONCURRENTLY PERFORM** the following:

[19.1] **MAINTAIN** Reactor water level between +12 and +50 inches on RX VESSEL LEVEL/TOTAL FW FLOW (Red Pen), 3-XR-3-53, AND less than +48 inches on 3-LI-3-208A(B)(C)(D).

[19.2] **DEPRESS** Pressure Setpoint RAISE, 3-HS-47-162B pushbutton on Panel 3-9-7, as necessary to maintain EHC SETPOINT, 3-PI-47-162 above Reactor pressure until reaching approximately 950 psig.

(R) __________ Initials __________ Time __________ Date

Attachment 3
(Page 15 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[20] **CLOSE** the following valves on Panel 3-9-3:

- 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A.

  ______ Initials ______ Date ______ Time

- 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A.

  ______ Initials ______ Date ______ Time

- 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A.

  ______ Initials ______ Date ______ Time

[21] **IF** shell warming or chest warming are not to be performed in parallel with Reactor pressurization, **THEN**

STOP shell warming and chest warming the high pressure turbine prior to exceeding 350 psig. REFER TO 3-OI-47. (N/A if warming is not in progress or is to be performed in parallel with Reactor pressurization).

______ Initials ______ Date ______ Time

Attachment 3
(Page 16 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

CAUTION

If not adjusted accordingly, turbine first stage pressure will rise as Reactor pressure rises while in shell warming or chest warming. Extreme caution must be exercised to ensure turbine first stage pressure is maintained in the pressure band dictated by 3-OI-47 to prevent a Reactor scram.

NOTE

Chest warming or shell warming may be conducted parallel to raising Reactor pressure to rated with the approval of the OPS Supt/OPS Manager (3-OI-47). CRD system injection to the Reactor at 60 to 80 gpm will support approximately 30,000 to 40,000 lbm/hr steam flow and maintain normal Reactor water level.

[22] **IF** EHC is available, and chest warming is to be performed parallel to Reactor pressurization, **THEN**

VERIFY the following prior to exceeding 350 psig: (Otherwise N/A)

- Main steam line drain valves CLOSED:
  - 3-FCV-1-55 using MN STM LINE DRAIN INBD ISOLATION VLV, 3-HS-1-55A. ☐
  - 3-FCV-1-56 using MN STM LINE DRAIN OUTBD ISOLATION VLV, 3-HS-1-56A. ☐
  - 3-FCV-1-58 using UPSTREAM MSL DRAIN TO CONDENSER, 3-HS-1-58A. ☐

__________ Initials __________ Date __________ Time

Attachment 3
(Page 17 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

- Stop valve before seat drains CLOSED:
  - 3-FCV-6-100 using STOP VALVE 1 BEFORE SEAT DR VLV, 3-HS-6-100A. ☐
  - 3-FCV-6-101 using STOP VALVE 2 BEFORE SEAT DR VLV, 3-HS-6-101A. ☐
  - 3-FCV-6-102 using STOP VALVE 3 BEFORE SEAT DR VLV, 3-HS-6-102A. ☐
  - 3-FCV-6-103 using STOP VALVE 4 BEFORE SEAT DR VLV, 3-HS-6-103A. ☐

________ Initials ________ Date ________ Time

- All RFP turbine warming drains CLOSED with the exception of the drains on the first RFP to be placed in service.

________ Initials ________ Date ________ Time

- Turbine steam seals ISOLATED from the Reactor steam supply.

________ Initials ________ Date ________ Time

- Offgas Preheaters ISOLATED from Reactor steam supply.

________ Initials ________ Date ________ Time

- SJAEs ISOLATED from the Reactor steam supply.

________ Initials ________ Date ________ Time

- Turbine bypass valves CLOSED.

________ Initials ________ Date ________ Time

Attachment 3
(Page 18 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

> **CAUTIONS**
>
> 1) Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
> 2) Changes in condensate system flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally must be in direct communication with the Control Room.

[23] **WHEN** Reactor pressure is greater than 450 psig, **THEN**

PERFORM the following:

[23.1] **VERIFY** two condensate and two condensate booster pumps running. REFER TO 3-OI-2.

_____________ _____________ _____________
Initials Date Time

> **CAUTION**
>
> If proper care is not exercised while placing a feedpump in service, over filling the Reactor vessel or quick charging the high pressure feedwater heaters may occur.

[23.2] **PLACE** first RFP in service. REFER TO 3-OI-3.

_____________ _____________ _____________
Initials Date Time

[23.3] **VERIFY** Condensate System flow being maintained within the limits of 3-OI-2 using CNDS FLOW CONTROL SHORT CYCLE, 3-FC-2-29, on Panel 3-9-6, in AUTO/BAL.

_____________ _____________ _____________
Initials Date Time

Attachment 3
(Page 19 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[23.4] **IF** RFP operation and/or level control is unstable, **THEN**

REMOVE RFP from service and **CONTINUE** to raise Reactor power and pressure. REFER TO 3-OI-3.

Initials ____ Date ____ Time ____

CAUTION
If proper care is not exercised while placing a feedpump in service, over filling the Reactor vessel or quick charging the high pressure feedwater heaters may occur.

[23.5] **IF** a RFP had to be removed from service due to unstable operation AND Reactor pressure is approximately 750 psig, **THEN**

PLACE RFP back in service. REFER TO 3-OI-3.

Initials ____ Date ____ Time ____

[24] **BEFORE** placing Seal Steam System, SJAE and Preheaters on nuclear steam, PERFORM the following: [BFN PER 126211]

[24.1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to transfer Seal Steam System, SJAE, and Preheaters to nuclear steam. **RECORD** time Radiation Protection notified in the NOMS Narrative Log. [BFN PER 126211]

(R) Initials ____ Date ____ Time ____

[24.2] **VERIFY** appropriate data and signatures recorded on Appendix A in accordance with Appendix A Instructions [Tech Spec 5.7] [BFN PER 126211]

(R) Initials ____ Date ____ Time ____

Attachment 3
(Page 20 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

NOTE
Prompt performance of Step 1.0[25] should stabilize Feedpump operation.

[25] **WHEN** a RFP is placed in service, **THEN**

PERFORM the following:

- **TRANSFER** Sealing Steam System from auxiliary steam to nuclear steam. REFER TO 3-OI-47C. (N/A if previously placed on Nuclear steam)

  ______ Initials ______ Date ______ Time

- **TRANSFER** SJAE and Preheaters from auxiliary steam to nuclear steam. REFER TO 3-OI-66. (N/A if previously placed on Nuclear steam)

  ______ Initials ______ Date ______ Time

- **BEGIN** warm-up of a second RFP. REFER TO 3-OI-3.

  ______ Initials ______ Date ______ Time

- **VERIFY** CRD flow 40 to 65 gpm. REFER TO 3-OI-85.

  ______ Initials ______ Date ______ Time

- **VERIFY** RCIC in standby readiness. REFER TO 3-OI-71.

  ______ Initials ______ Date ______ Time

Attachment 3
(Page 21 of 21)

Startup with MSIV's Closed

1.0 REACTOR STARTUP WITH MSIV'S CLOSED (continued)

[26] **CONTINUE** in this procedure at Step 5.0[62] and N/A Steps 5.0[42] through 5.0[61].

______	______	______
Initials	Date	Time

Appendix A
(Page 1 of 2)

Radiation Protection Notifications

INSTRUCTIONS FOR APPENDIX A DATA ENTRY

This appendix provides record of Radiation Protection notifications, RPHPs, and required signatures made during the performance of this GOI. Each notification step in this procedure, or in any referenced support procedure, that requires Appendix A be entered requires the following instructions to be used to complete the appropriate parts of the data entry page. Copies are made as needed to support this data entry.

B. Ops **ENTER** name of the Radiation Protection Representative notified with date and time of notification. Time of notification is also required in NOMS narrative log.

C. Ops **ENTER** step number (including Section number) associated with notification requirement. If the notification is directed from a support procedure, then enter the procedure number and current revision number

D. For all RPHP notifications, Radiation Protection **DETERMINE** if the RPHP is required to prevent unintended exposures and/or to implement RCI-17, Control of High Radiation Areas and Very High Radiation Area controls. **IF** RPHP is identified in a support procedure to this GOI, **THEN DETERMINE** if an RPHP is also necessary for the GOI. **CONFER** with Operations, as necessary.

E. For each identified procedure RPHP, Radiation Protection Supervisor's signature is required to release the RPHP for the action associated with affected step. This signature signifies one of two conditions: [SOER 01-1, Tech Spec 5.7, BFN PER 126211]

1. Radiation Protection actions are completed to prevent unintended exposures and/or RCI-17 requirements have been met and any personnel working within affected areas are on an appropriate RWP for the anticipated radiological conditions.

OR

2. No actions were necessary because appropriate controls were already in place.

F. **WHEN** the use of this procedure is completed, **FORWARD** copies of the completed appendix pages to the Radiation Protection Supervisor.

If, while performing this procedure, or while performing a support procedure, Radiation Protection personnel, Unit Operator, Unit Supervisor, or other knowledgeable shift member identifies the need for a RPHP, then "RPHP" is written to the left of the affected procedure step number (this GOI or the support procedure. If the RPHP is identified for a support procedure, then RPHP is also placed to the left of the step in this GOI that initiates the support procedure and then A through E above is performed, as applicable.

Appendix A
(Page 2 of 2)

Name Of Radiation Protection Person Notified: ______________________________

Date: ____ / ____ / _______ Time:________________

Step#________ Procedure: ______________ (if not this procedure) Rev:______

RPHP Required by OI? ____(Y) ____(N) RPHP Required For GOI? ____(Y) ____(N)

RCI-17 Controls Necessary? ____(Y) ____(N)

Radiation Protection Supervisor Signature for Release

______________________________ Date: ____ / ____ / _______ Time:________________

Comments:

Name Of Radiation Protection Person Notified: ______________________________

Date: ____ / ____ / _______ Time:________________

Step#________ Procedure: ______________ (if not this procedure) Rev:______

RPHP Required by OI? ____(Y) ____(N) RPHP Required For GOI? ____(Y) ____(N)

RCI-17 Controls Necessary? ____(Y) ____(N)

Radiation Protection Supervisor Signature for Release

______________________________ Date: ____ / ____ / _______ Time:________________

Comments:

FORWARD copies of completed Appendix pages to Radiation Protection Supervisor.