Draft Submittal

BRUNSWICK EXAM
50-2003-301
50-325 & 50-324

FEBRUARY 10 - 14 & 19, 2003

1. Operating Test Simulator Scenarios

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description: _The crew assumes the shift with power at 90% EOL. There is a severe demand for power due to a late heat wave. RCIC Inboard Steam Isolation valve E-51-F007 is Inoperable for breaker maintenance (valve is OPEN). TS 3.6.1.3 is satisfied with E-51-F045 closed and its breaker open. The previous shift initiated 0GP-13 "Increasing Unit Capacity at End of Core Cycle", Section 5.1 (Bypassing Feedwater Heaters #4 and #5). Night orders include direction to complete a Control Rod Operability Check on rod 42-39 per 0PT-14.1 (*all rods but 42-39 were already completed*) and then to increase power to 100%.

Time	Position	Applicant's Actions or Behavior
1		The RO completes 0PT-14.1 and commences power increase to 100%.
2		Following power increase to 100% APRM 1 fails HI (MNI031F). *This is intended to simply create an instrument malfunction for the RO, but may be interpreted by the crew as being associated with 0GP-13.* The RO is expected to bypass the APRM and reset the half scram. SRO will check TS.
3		One channel of MS Radiation fails low (MRM001F) . The crew is expected to take appropriate actions per Alarm Response Procedures and TS and may again attribute the failure to actions taken for 0GP-13.
4		At this point the fuel failure is increased and is readily detectable on the three remaining MS Rad Monitors as well as the Off Gas Monitor(s). *Note it is intended that the fuel failure be high enough to ensure the MSIVs remain closed, but not so high as to cause radiation levels in the Reactor Building to exceed "Maximum Safe" levels.*
5		The crew may attempt to reduce power, but MS radiation will continue to increase (MRM0011F through 13F). *The increase is intended to be slow enough to allow the SRO to make the decision to manually Scram and close the MSIVs before automatic action occurs. With the feedwater heaters bypassed there are restrictions on how low the power may be reduced. This may complicate the decision and may "push" the SRO to manually scram earlier rather than later.*

6		When the MSIV closure occurs a RCIC Steam Break occurs in the South RHR room upstream of E-51-F045 (MES025F). The Area Temperature Hi isolation signal is generated, but valve F008 motor trips on overload (MES026F). The SRO will direct entry into Secondary Containment Control 0EOP-03-SCCP and will, ultimately result in Emergency RPV Depressurization since all attempts to isolate the RCIC Steam Leak will be unsuccessful.

Additional work that needs to be done on Scenario # 1

1. Risk for CRD & HPCI ooc
2. TS actions for APRM
3. Water Leak vs Steam Leak
4. Turnover Notes for Scenario
5. Reactivity Plan for Scenario
6. Need 0-GP-12 Signed off to step in effect.
7. Ensure both RBM have been declared OP & on scale
8. Get Good power rods, 00-12, Rings 1, 2, 3 and 4/ 1c to get good rods
9. Steps to trip APRM Channel

Facility: Brunswick Scenario No.: 1 Op-Test No.: ______

Title:Medium Break LOCA inside containment with Loss of Offsite Power and failure of one EDG

Examiners: ______________________ Operators: ______________________

______________________ ______________________

Initial Conditions: The crew assumes the shift with the plant at 28% power, BOL. HPCI is OOC, the #1 APRM failed low and is bypassed. One CRD (22-19) is inoperable; stuck at position 48. CRD Pump "2B" is OOC for PM's. Severe weather has been reported in the area. The previous shift completed partial stroking of all MSIVs except "A" inboard

Turnover: The previous shift has completed all Required Actions for the Stuck Rod per TS 3.1.3 (Rod 22-19 is disarmed). Night orders include direction to partial stroke the "A" inboard MSIV per PT-40.2.8 and then to increase power to 40%. The Reactor Engineer recommends using control rods per Sequence A-1 (continuous withdrawal acceptable) with a maximum power increase of 1% per minute.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N(BOP)	Partial stroke "A" Inboard MSIV
2	N/A	R(RO) (SRO)	Increase power from 28% to 40%.
3	MRD018F	C (RO)	"A" CRD Pump suction filter plugged.
4	MNI037F	I (RO)	#2 APRM fails low.
5	MRC007F MRC009F	C (RO) (SRO)	"A" Recirc Pump Seal Leak.
6	MCN017F	C (BOP) (SRO)	AOG Guard Bed Fire
7	MEE032	M(ALL)	Loss of Offsite Power
8	MDG002F	C(BOP)	One EDG fails to start Note: This event is combined with Event 7 in one D-2
9	MNB009F	M (ALL)	1000 GPM leak inside drywell

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

Op-Test No.: _____ Scenario No.: 1 Event No.: 1

Event Description: Perform Partial Stroke Test for the "A" Inboard MSIV per PT-40.2.8. The surveillance has been completed for the remaining MSIV's. The simulator instructor will provide relay and light status from scripted messages when asked by the BOP.

Time	Position	Applicant's Actions or Behavior
	SRO	Direct the BOP to perform the MSIV functional surveillance for the "A" Inboard MSIV per procedure PT-40.2.8. Note: The simulator operator will function as the relay/light observer at "back" panels
		.
	BOP	1. Obtain and review Precautions and Limitations of PT-40.2.8. Request RO to monitor Rx Pressure and Main Steam Line Flow during valve stroking.
		2. Verify that MSIV DC & AC coil lights (P622/P623) lights/indications are illuminated.
		3. Verify that and RPS Group 1-4 (P609/P611) are lit.
		4. DEPRESS B21-F022 TEST pushbutton and confirm valve strokes to dual position indication and relays operate.
		CUE: *Simulator Operator will report "C72-K3A contacts 1-2 & 3-4 are open; 5-6 & 7-8 are closed at P609 and C72-K3B contacts 1-2 & 3-4 are open; 5-6 & 7-8 are closed on P611"*
		5.Release B21-F022A TEST pushbutton and confirm B21-F022A fully opens.
		CUE: *If requested, Simulator Operator will report "C72-K3A contacts 1-2 & 3-4 are closed; 5-6 & 7-8 are open at P609 and C72-K3B contacts 1-2 & 3-4 are closed; 5-6 & 7-8 are open on P611"*
	RO	Monitor Main Steam Flow and Reactor Pressure Note: At this low power no significant pressure increase or flow change is expected

Op-Test No.: ____	Scenario No.: 1	Event No.: 1	Page 4 of 20

Event Description: Perform Partial Stroke Test for the "A" Inboard MSIV per PT-40.2.8. The surveillance has been completed for the remaining MSIV's. The simulator instructor will provide relay and light status from scripted messages when asked by the BOP.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 2

Event Description: Increase power from 28% to 40%. Note: The next event (CRD filter clogged) will be initiated when the Reactivity Manipulation has been satisfied or at the Lead Examiner's discretion

Notes:1. The Reactor Engineer has specified power increase be accomplished using continuous withdrawal with Control Rod Sequence A-1 not to exceed 1% per minute.
2.The RO will be given a copy of 0GP-12 with sections 5.2.1 through 5.2.17 completed (signed off).

Time	Position	Applicant's Actions or Behavior
	SRO	1. Ensure power increase is acceptable to Load Dispatcher 2.Direct the RO to increase power in accordance with 0GP-12. He will specify using continuous withdrawal of Control Rods in sequence A-1 at <1% power/min. 3. He will direct the BOP to monitor Turbine Operation in accordance with OP-26, Figure 3.
	RO	1) Obtain current copy of 2OP-07 and 0GP-10
		2) Review Precautions and Limitations and Prerequisites in 0GP-12, and 0GP-10.
		2) Ensure steps 5.2.1 through 5.2.17 and Attachment 2 of 0GP-12 have been completed
		3) Withdraw control rods per sequence A-1 to effect <1%/min increase and monitor Reactor Parameters. - Select the desired rod - Ensure ROD WITHDRAWAL PERMISSIVE is lit - Hold EMERGENCY ROD IN NOTCH OVERRIDE to OVERRIDE and NOTCH OUT to effect continuous withdrawal - Release both switches and ensure rod settles to desired position - Monitor CRD position reactor power during withdrawal - Check coupling integrity at 48 - Stop rod withdrawal one notch before desired position - Notch rods with movement of 3 notches or less - Place ROD SELECT POWER to OFF at end of rod movement
		4) Swap Feedwater Level Control to 3-ELEM (if not previously done)

Op-Test No.: _____ Scenario No.: 1 Event No.: 2

Event Description: Increase power from 28% to 40%. Note: The next event (CRD filter clogged) will be initiated when the Reactivity Manipulation has been satisfied or at the Lead Examiner's discretion

Notes:1. The Reactor Engineer has specified power increase be accomplished using continuous withdrawal with Control Rod Sequence A-1 not to exceed 1% per minute.
2.The RO will be given a copy of 0GP-12 with sections 5.2.1 through 5.2.17 completed (signed off).

Time	Position	Applicant's Actions or Behavior
		5) Perform APRM Normal Trip Setpoint (if not previously done) - CONFIRM FCBB is less than 1.0 - DEPRESS the NORMAL/SETUP pushbutton on FCTR card for each APRM - CONFIRM NORMAL/SETUP LED is yellow
		6) CONFIRM Turbine Stop Valve/Control Valve Fast Closure Reactor Scram is enabled - Confirm C71 Relays are deenergized - Confirm TURB CV FAST CLOS/SV TRIP BYPASS (A-05, 6-7) is clear
		6) Confirm core thermal limits are per TS; Power/Flow is outside Restricted Region
	BOP	1 Obtain a copy of OP-26.
		2. Monitor Turbine Power Increase per OP-26, Figure 3. - Power change from 28% to 40 % can be done as a step increase - Routinely monitor Turbine Parameters (Oil, vibration, vacuum)
		3. Start a second Condensate Booster Pump
		4. Notify Radwaste to put additional CDDs and CFDs in service

Op-Test No.: _____ Scenario No.: 1 Event No.: 2

Event Description: Increase power from 28% to 40%. Note: The next event (CRD filter clogged) will be initiated when the Reactivity Manipulation has been satisfied or at the Lead Examiner's discretion

Notes: 1. The Reactor Engineer has specified power increase be accomplished using continuous withdrawal with Control Rod Sequence A-1 not to exceed 1% per minute.
2. The RO will be given a copy of 0GP-12 with sections 5.2.1 through 5.2.17 completed (signed off).

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 3	Page 8 of 20

Event Description: "A" CRD Pump suction filter plugged. Execute malfunction MDR018F per Lead Examiner direction.

Time	Position	Applicant's Actions or Behavior
	RO	1. Observe alarm "CRD Pump Inlet Filter DP High" alarm and monitor CRD parameters.
		2. Review Alarm Response Procedure A-05 5-1.
		3. Stop any power increase
		4. If more than one control rod is drifting THEN INSERT a manual scram
	SRO	1. Direct RO to complete ARP actions
		2. Dispatch an AO to locally monitor filter DP and pump suction pressures.
		CUE: *AO will report that filter dp > 10 psid*
		3. Obtain a copy of 0AOP-02.0 and direct RO actions as appropriate (pump failure)
		4. Consult TS 3.1.5 - Recognize that If/when one accumulator alarm comes in have 20 minutes to restore CRD pump (TS 3.1.5.B.1) - Acknowledge/Specify 20 minute "clock" to get CRD pump ON
		CUE: Simulator provide second accumulator alarm in approximately 3 minutes
	RO	1. Observe running CRD pump trips and "CRD Pump 2A Lo Suct Press" alarm (Note: Other CRD alarms will come in as well)
		2. Monitor core thermal parameters to keep within TS
		3. Direct the AO to shift CRD Pump suction filters
		4.When advised (simulator operator) that the CRD suction filters have been shifted, Re-Start the 2A CRD Pump

Op-Test No.: ____	Scenario No.: 1	Event No.: 3	

Event Description: "A" CRD Pump suction filter plugged. Execute malfunction MDR018F per Lead Examiner direction.

Time	Position	Applicant's Actions or Behavior
		CUE: *The AO will report back in 10 minutes that filter swap has been completed*
		5. Observe CRD parameters return to "normal"; specifically that charging water >940 psig within 20 minutes
		6. Request the AO ensure the CRD suction filter is less than 3 PSID.
		CUE: *The AO will report suction filter DP is 1 psid*
	SRO	1. Note shifting of CRD filters in log and initiate MR to replace filters in clogged unit.
		2. Ensure TS are satisfied in new configuration/ exit TS 3.1.5

Op-Test No.: _____ Scenario No.: 1 Event No.: 3	Page 10 of 20

Event Description: "A" CRD Pump suction filter plugged. Execute malfunction MDR018F per Lead Examiner direction.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: __1__ Event No.: __4__
Event Description: #2 APRM Fails Low

Insert malfunction MNI037F at direction from Lead Examiner

Time	Position	Applicant's Actions or Behavior
	RO	1 Observe "APRM Downscale" alarm and notify SRO
		2. Compare #2 APRM with #3 and #4 APRM channels (A-06 2-7)
		3. Refer to TS 3.3.1.1 and TRMS 3.3 for APRM operability requirements
		3. Confirm other APRM channels are functional (except #1)
		4. Trip the affected APRM channel (Steps)
	SRO	1. Consult TS 3.3.1.1 and TRMS 3.3.
		2. Ensure no other half scrams are present
		3. Direct RO (or BOP) to place the appropriate APRM Channel to TRIP
	RO	Ensure half scram- RPS Group Lights for Channel "A" out
	SRO	1. Ensure TS are satisfied and log information
		2. Initiate MR to get APRM #2 repaired

Op-Test No.: _____ Scenario No.: 1 Event No.: 5	Page 11 of 20
Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC014F, MRC007F and MRC009F when directed by Lead Examiner.	

Time	Position	Applicant's Actions or Behavior
	RO	1.Observe "Recirc Pmp A Motor Vib High" alarm and notify SRO. -Monitor Recirc Pump A motor bearing temperatures on recorder B32-R601 -Attempt to reset the alarm to determine if it was spurious
		CUE: *The recorder shows increasing motor bearing temperatures and the vibration alarm cannot be reset. MRC007F should be initiated after RO and SRO confer on vibration, but before any decision to Trip the pump is made*
		2.Observe "Outer Seal Leakage Flow Detection Hi" alarm and notify SRO. *Note: This should be a clear indication that the seal has failed and that a discharge into containment is occurring.*
		3. Observe changing seal DPs and notify SRO that both seals appear to have failed or are failing
		3. Request AO verify correct seal lineup and monitor parameters in reactor building
		4. Monitor seal cavity temperatures on recorder B32-R601 at Panel P614. *Note: Pump should be tripped and isolated prior to seal temperatures exceeding 200 F*
	SRO	1. Direct RO to complete actions in ARPs (A-06 3-3 and A-06 5-3)
		2. Direct BOP to monitor containment parameters
	BOP	1. Observe Drywell pressure and temperature and sump levels all increasing, refer to 0AOP-14.0 and advise SRO.
		2. Monitor drywell equipment drain sump pumps for frequency of operation and run time.
		3. Ensure all drywell coolers are operating and RBCCW lined up.

Op-Test No.: _____ Scenario No.: __1__ Event No.: __5__	Page 10 of 20

Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC014F, MRC007F and MRC009F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	SRO	1. Direct RO to reduce power in anticipitation of tripping "A" Recirc Pump. *Notes: Pump should be tripped and isolated if seal temperatures reach or exceed 200 F (A-06 5-3) SRO may direct to Transfer per OP-02.*
		2. Call dispatcher and advise of power reduction/possible scram
	RO* *SRO may direct the BOP to Trip and Isolate the pump if the RO is driving rods	1. Reduce power as necessary per 0GP-12 Note: Should insert rods per 0GP-10, but may just transfer from two loop to one loop as specified in the following steps.
		2. Place control switch for Seal Staging Vlv B32-V14 to MAN/OPEN
		3. Shutdown recirc Pump 2A by placing RECIRC MG SET 2A control switch to STOP Note:Shutdown of "A" Recirc Pump should be before 200F seal temperatures are reached. This should be done without a Turbine Trip from High RPV water level
		5. Isolate "A" Recirc Pump **(Critical Task)** - Close suction valve prior to closing discharge valve - Close discharge valve before closing discharge bypass - Isolate seal purge (from CRD) prior to closing bypass valve
		6.Determine core flow using point WTCF
		7.Raise speed on Recirc pump B if permitted by feedwater interlocks.
		8. Ensure total core flow is<45mlb/hr
		9. Ensure Precautions and Limitations of 0GP-12 are satisfied following pump trip (E.G., Power/Flow)
		10 CHANGE or REMOVE rods on SRI bus if approaching 30%

Op-Test No.: _____	Scenario No.: 1	Event No.: 5	Page 11 of 20

Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC014F, MRC007F and MRC009F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	BOP/RO	1. If/when Drywell Pressure exceeds 1.7 psig advise SRO.
		2. Start all available DW coolers- **DO NOT vent DW** (leak present)
	SRO	1. Notify Operations Management and Reactor Engineer of power reduction.
		2. Ensure TS for single loop operation are satisfied
		3. Log events
		4. Enter 0EOP-02-PCCP if/when drywell pressure exceeds 1.7 psig

Op-Test No.: _____ Scenario No.: 1 Event No.: 5

Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC014F, MRC007F and MRC009F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 6	Page 13 of 20
Event Description: AOG Guard Bed Fire. Execute malfunction MCN017F when directed by Lead Examiner.	

Time	Position	Applicant's Actions or Behavior
	BOP	1 Acknowledge receipt of "2-AOG-D1 Guard Bed Temperature High" alarm and advise SRO
		2. Request AO to observe AOG parameters at local panel H2E *Note: AO will report back that "guard bed temperatures have significantly increased but reheater temperatures are normal".*
		2. Ensure AOG HVAC equipment is normal
		3. Bypass and Isolate guard bed -Open 2-AOG-V013 and 2-AOG-V014 THEN -Close 2-AOG-V009, 2-AOG-V010, 2-AOG-V011 and 2-AOG-V012
		4. Monitor guard bed temperatures. - If guard bed continues to rise, immediately purge with N2 per OP-33
		CUE: *Guard bed temperatures continue to rise even after isolating*
	SRO	1.Based on hearing that the guard bed temperatures continue to rise after isolation, he should obtain a copy of OP-33 and direct an AO to "purge the guard bed with N2 in accordance with Section 8.5" - Close AOG-V011 and V-012 - Disconnect switch ON - Throttle open Nitrogen Purge supply Valve AOG-NP-V079 - Establish 50 SCFM N2 flow
		2. Notify OPS Management and HP of the problem
		CUE: *Once the order is given (to the AO) to purge the guard bed, MCN017F will be removed. This event will be considered complete when temperatures of the guard bed begin to decrease.*

Op-Test No.: _____ Scenario No.: 1 Event No.: 6		Page 14 of 20
Event Description: AOG Guard Bed Fire. Execute malfunction MCN017F when directed by Lead Examiner.		

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 7

Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	SRO	1. Direct actions of EOP-01-RSP.
		2. Obtain copy of AOP-36.1 and direct actions. *Note: Although the loss of one EDG to start is considered a separate event in the D-1 it is combined into this event with the BOP to take additional actions for manually restarting the EDG.*
		3. If/when suppression pool temperature exceeds 95 F, enter EOP-02-PCCP.
	RO	1. Complete actions of EOP-01-RSP "Reactor Scram". -Mode switch to Shutdown -Verify power<5% -Operate RCIC in level band +170" to +200" -Open SRV to obtain 950 psig -Insert nuclear instrumentation
		2. Recognize that not all rods are full in and advise the SRO
		2. Monitor and control RPV level and pressure using RCIC and SRVs
		3. Place RHR in suppression pool cooling without keepfill Note: Starting of RHR SW Pumps will be dependent on power availability
		4.Start RPS MG Sets
		5. Restart CRD pump(s) as necessary
	BOP	1. Recognize/announce that one EDG did not start
		2. Attempt to start the failed EDG *Note: This sequence is continued below*
		3. Ensure DC Oil Pumps start
		4. Ensure NSW Pumps running, Start CSW Pumps to support RCC
		5. Ensure battery chargers operating

Op-Test No.: _____ Scenario No.: 1 Event No.: 7		Page 16 of 20
Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.		

Time	Position	Applicant's Actions or Behavior
		6. Start Control Room and battery Room HVAC
		7. Restore Drywell Cooling.
		8. IF motor driven Fire Pump is ON, then Start Diesel Driven Fire Pump and shutdown motor driven (power conservation)
		9. Trip Main Turbine (part of RSP) and monitor lube oil
	SRO	1. When advised of one rod full out he should recognize that the reactor will remain shutdown under all conditions and NOT go to "Level/Power Control"
		2. When advised that one EDG did not start he should direct the BOP to try a manual start per AOP-36.1 *Note: It is likely that the BOP would do this on his own per 0AOP-36.1*
	BOP	1. Manually start the failed EDG. Note: The BOP will be "permitted" to manually start the EDG but it will immediately Trip on differential current when he attempts to close the output breaker
		2. Attempt to close breaker (insert malfunction MDG024F)
		3. Recognize that the Differential Fault is serious and probably means the EDG is lost for the duration of the event.
		4. Direct an AO to investigate the fault.
		Note: Once the BOP recognizes the EDG is "gone" and actions of 0AOP-36.1 have been completed, step into the next event (1000 GPM leak)

Op-Test No.: ______ Scenario No.: 1 Event No.: 7		Page 17 of 20
Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.		

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____	Scenario No.: 1	Event No.: 8	

Event Description: 1000 GPM leak inside the drywell. Malfunction MNB009F will be initiated as directed by the lead examiner

Time	Position	Applicant's Actions or Behavior
	BOP	1. Observe containment parameters and identify that a leak inside containment is in progress. *Note: The leak will start small and gradually increase to 1000 GPM. Initially RCIC and CRD will be able to maintain level, but level decrease will be apparent when leakage exceeds approximately 600 GPM*
	SRO	1. Direct actions per EOP-01-RVCP. Execute RC/L and RC/P concurrently
		2. Once it is apparent that Reactor Water cannot be maintained above TAF the SRO will direct the RO to initiate a cooldown to the point that either Core Spray or LPCI will makeup to the Reactor.
		3. When drywell pressure or suppression pool temperature exceeds entry conditions, he will enter 0EOP-02-PCCP
		.
	RO	1. Maximize RPV injection with available high pressure sources (RCIC and CRD)
		2. Place Torus Sprays in service prior to exceeding 11.5 psig. Note: The RO should observe power constraints with only one EDG available and use 0AOP-36.1 when starting RHR pump(s)
		3. Alternate SRVs to maintain/reduce RPV pressure (cooldown < 100 F/hr)
		4. Inhibit ADS
		5. Recognize/advise SRO that available high pressure sources will not be adequate to maintain RPV level above TAF
		6. Ensure Core Spray is lined up for injection

		7. Depressurize RPV using one or more SRVs until Core Spray is injecting Notes: 1. Shutoff head approximately 350 psig 2. The SRO may determine that water level cannot be maintained above TAF and proceed to Emergency Depressurization
		The scenario will be terminated once Core Spray flow has been established to the RPV.

Event 8 LOCA exceeding high pressure makeup

Time	Position	Applicants actions or behavior

Event 8 LOCA exceeding high pressure makeup		
Time	Position	Applicants actions or behavior

Scenario 1 - Initial Conditions

REFERENCES

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	-----NOTE----- Stuck control rod may be bypassed in the rod worth minimizer (RWM) or RWM may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ---	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	AND	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	AND	
		(continued)
A. (continued)	A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND	
	A.4 Perform SR 3.1.1.1.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- Inoperable control rod may be bypassed in the RWM or RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	AND		(continued)
C. (continued)	C.2	Disarm the associated CRD.	4 hours
D. ---------NOTE--------- Not applicable when THERMAL POWER > 8.75% RTP. --------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with BPWS.	4 hours
	OR		
	D.2	Restore control rod to OPERABLE status.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1 --------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------- Manually scram the reactor.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ 940 psig.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 950 psig.	B.1	Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND		
	B.2.1	--------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------- Declare the associated control rod scram time "slow."	1 hour
	OR		
	B.2.2	Declare the associated control rod inoperable.	1 hour

(continued)

C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 950 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	AND		
	C.2	Declare the associated control rod inoperable.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 950 psig.	B.1	Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND		
	B.2.1	--------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------------------	
		Declare the associated control rod scram time "slow."	1 hour
	OR		
	B.2.2	Declare the associated control rod inoperable.	1 hour

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 950 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	AND		
	C.2	Declare the associated control rod inoperable.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

Scenario 1, Initial Conditions

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ Stuck control rod may be bypassed in the rod worth minimizer (RWM) or RWM may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ----------------------------	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	AND	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	AND	
		(continued)
A. (continued)	A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND	
	A.4 Perform SR 3.1.1.1.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- Inoperable control rod may be bypassed in the RWM or RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	<u>AND</u>		(continued)
C. (continued)	C.2	Disarm the associated CRD.	4 hours
D. ---------NOTE--------- Not applicable when THERMAL POWER > 8.75% RTP. ---------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with BPWS.	4 hours
	<u>OR</u>		
	D.2	Restore control rod to OPERABLE status.	4 hours

Scenario #1, Event #1

EVENT 1 MSIV TESTING

Instructor Activities

- ☐ Establish communications with BOP operator; report you are on station to observe relay actuation during PT performance.
- ☐ If asked MSIV DC & AC coil lights (P622/P623) are lit and RPS Group 1-4 (P609/P611) are lit
- ☐ Monitor MSIVs on panel mimic or cameras. When B21-F022A is closed, report relay C72-K3A contacts 1-2 & 3-4 are open, 5-6 & 7-8 are closed (P609) and that C72-K3B contacts 1-2 & 3-4 are open, 5-6 & 7-8 are closed (P611)
- ☐ When B21-F028A is attempted to be stroked, report no relay actuation.
- ☐ NOTE: The SCO has no way of knowing exactly why MSIV fails to stroke (could simply be a faulty test circuit)

Plant Response

- ☐ B21-F028A fails to stroke when attempted

Operator Activities

SCO

- ☐ Direct Reducing Reactor Power to <85% using Recirculation flow
- ☐ Direct PT-40.2.8 be performed
- ☐ Direct suspension of PT
- ☐ Direct I&C to investigate failure of B21-F028A
- ☐ Initiate tracking LCO on MSIV 28A and request engineering evaluation as to why valve did not stroke (OI-01.08). (No TS requirement for valve to slow close, the potential operability issue is will the valve respond to an isolation)

RO

- ☐ Reduce Reactor Power to <85% using Recirculation flow
- ☐ Monitor the plant during performance of PT-40.2.8

BOP

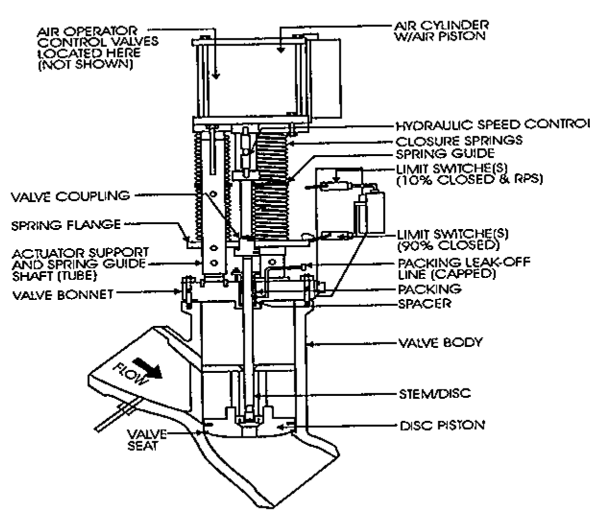
- ☐ Verify prerequisites for test are met
- ☐ Confirm no scram alarms are sealed in
- ☐ Establish communications with individuals in back panels
- ☐ Determine FFTR not in effect
- ☐ Transfer Feedwater control to 1-elem per OP-32
- ☐ Confirm scram group light are illuminated
- ☐ Depress B21-F022A test pushbutton, confirm valve strokes to dual position indication and relays operate
- ☐ Release B21-F022A pushbutton and confirm B21-F022A fully opens
- ☐ Depress B21-F028A test pushbutton
- ☐ Recognize and report failure of B21-F028A to stroke

Scenario #1, Event #1

The bases associated with the MSIV stroke time limitations serve to both preserve the integrity of the Reactor Coolant System and to minimize radioactive releases. The minimum stroke time of 3 seconds allows the MSIV closure scram signal to reduce reactor power and pressure before the motion of the MSIV has caused pressure to start to rise. Since the MSIV Closure Scram signal is initiated by valve position less than 90% open and the valve does not significantly effect steam line flow or pressure until less than ~25% open, it can be seen that, ideally, power and pressure are both decreasing before the MSIV closures can effect reactor power. The maximum stroke time of 5 seconds is twofold in that it serves to minimize the inventory loss from the Reactor Vessel and minimize the radioactive releases to the environment.

Minimizing the inventory loss will reduce the likelihood of core uncovery, thus prevent fuel damage. Minimizing radioactive releases prevents exceeding the accident release limitations stated in 10-CFR-100. Analysis has shown that excessive releases or core uncovery are not expected to occur for closing times up to 10.5 seconds.

Each MSIV operator contains two AC solenoids and one DC solenoid. One of the AC solenoids is used for valve stroke testing at power and is called the Slow Closure Test Solenoid. The other two solenoids (one AC and one DC) determine the position of the MSIV by porting or venting the pneumatic source to or from the operator. Both of these solenoids must be deenergized for the MSIV to be closed. The AC solenoids are powered from the Reactor Protection System and the DC solenoids are powered from the Station Battery System. This arrangement prevents inadvertent MSIV closure unless redundant signals are received yet, is fail safe in that a loss of power will result in closure. MSIV position is controlled via CLOSE-AUTO/OPEN control switches located on the P601 panel. The automatic function is associated with the AUTO/OPEN position and closure is caused by the Primary Containment Isolation System (PCIS) or a Low Vacuum condition in the Main Condenser.



MSIV isolation signal status is given by a group of white lights located on the P601 panel. These lights are arranged above the MSIV control switches as follows:

TABLE 25-3, MSIV ISOLATION SIGNAL STATUS

Light	INBD DC	INBD AC	OUTBD DC	OUTBD AC
Solenoid Power	125 VDC "A"	RPS "A"	125 VDC "B"	RPS "B"
PCIS Logic	B	A	A	B

A half isolation signal sensed by the "A" PCIS logic will result in extinguishing the two lamps in the center. The two outboard lamps will remain lit and no valve motion will occur. The two extinguished lamps represent the inboard AC and the outboard DC solenoids. The white lamps for the inboard valves are repeated on the P622 panel while those for the outboard valves are repeated on the P623. MSIV solenoid power status is provided by red LED indicators located on the back of P622 for the Inboard MSIVs and the back of P623 for the Outboard MSIVs. Each location contains eight RED LED lights, four for DC power and four for AC.

Isolation capability to the inboard and outboard AC MSIV Pilot Solenoids circuitry is provided by a key locked isolation switch located on Panel P622 for the inboard valves and on Panel P623 for the outboard valves. In the event of a fire the key locked isolation switches provide the capability for total isolation of the AC power supplies to the pilot Solenoids. The isolation contacts will remove the ground path from the circuit in case of a "hot" short in the field cable which will prevent spurious valve operation.

2.5.1 Pneumatic Operator

Operation of the MSIVs (Figure 25-6A thru 25-6C) is pneumatic to open and pneumatic with spring assist to close. Each MSIV is supplied with two pneumatic sources via a dual header and check valve arrangement. This arrangement allows for the loss of either source without effecting the other. The pneumatic sources for the outboard MSIVs are Reactor Building Non-Interruptible Air (RNA) System Division I and Division II. The pneumatic sources for the inboard MSIVs are Pneumatic Nitrogen System (PNS) Division I and Division II. Unlike the SRVs, a loss of PNS does not result in lining up the BU N2 System to the pneumatic operators.

An air accumulator located between the MSIV air operator and the check valves provides backup operating air. The capacity of the accumulator is sufficient for the air operator to exercise the valve through one-half of a cycle (open-to-closed or closed-to-open) should the supply air to the operator be interrupted.

The MSIV air operator control unit is attached to the air cylinder and contains the pneumatic, AC, and DC solenoid valves and control valves for opening, closing, and slow speed exercising of the main valve. The control power for each MSIV is 120 volts AC, and 125 volts DC. During normal operation (MSIV open) either or both the AC and DC coils of the solenoid valve(s) are energized and instrument air is directed to the underside of the air operator piston. Thus, the closing force exerted by the springs is overcome by the air operator and the valve is maintained in the open position.

When the solenoid operated control valve(s) are deenergized as in a two-channel trip or whenever the manual control switch is taken to the closed position, the air supply pressure is switched to pressurize the opposite side of the air operator piston and assists the spring to close the valve.

Air supply pressure acting on the operator piston or spring force is capable of independently closing the valve with the reactor vessel at full pressure. Thus if one fails, the other will successfully close the valve (provided area below the operator piston is vented off to atmosphere). Two vent valves provide a redundant means for bleeding off the under piston air in the event that a valve fails to operate on a valve closure signal.

A separate test pushbutton is provided for a manual test of slow closure of each MSIV from the Control Room. Slow Closure of a MSIV when testing should require 45 to 60 seconds (Figure 25-6C). Slow closure utilizes springs only and vents the area under the operator piston through an adjustable aperture.

Fast closure time is set between 3 and 5 seconds. These time limits are set in order to: (1) prevent uncovering of the core through loss of inventory, (2) reduce the amount of activity released to the environs in the event of a gross steam leak, and (3) minimize the pressure transients on the reactor vessel and fuel. Stroke time is obtained through adjustment of two timing control knobs located on the pneumatic operator. The upper speed control knob is to adjust opening speed and the lower control knob is to adjust closing speed. The air supply piping to the operator is sized such that no depressurization of the accumulator will take place during valve operation.

2.5.2 Automatic Closure of MSIV's

The main steam isolation valves are designed to close on any of the following primary containment isolation signals:

- Reactor Water Low Level 3
- Main steam line high flow
- Reactor Building steam line tunnel high temperature
- Turbine Building main steam tunnel high temperature
- Low Condenser Vacuum - The low condenser vacuum trip bypassed when the turbine stop valves are less than 90% open and bypass switches (A71B-S34A-D) are in the bypass position and the reactor mode switch is not in the run position.
- Main Steam Line Low Pressure - The low turbine inlet pressure trip is bypassed whenever the reactor mode switch is not in the run position.

(See Table 25-5, Instrument and Control Setpoints.)

It is possible to completely isolate two steam lines by manually closing up to four MSIVs from the Unit RTGB without initiating a full reactor scram, provided the four MSIVs are only in two steam lines. Any attempt to isolate three main steam lines will cause a full scram through both RPS trip channels.

The automatic isolation signal is a one-out-of-two-taken-twice logic. A trip must occur in one or both trip channels of Trip System "A" and in one or both trip channels of Trip System "B", in order to initiate valve closure (Figure 25-7, 25-7A, and 25-7B).

3.5 Reactor Protection System

In the Figure 25-7 and 25-7A, we can see that the position of each MSIV is sensed by contact position from limit switches mounted on the valve. When the valve is >90% open the contacts (trip system A and B) will be closed indicating the valve is FULL OPEN. Likewise, if the valve is <90% open, the contacts are open and the appropriate relays are de-energized. This will be indicated by the MAIN STEAM ISOL VLV NOT FULL OPEN (A-05, 4-6) (any two of eight valves) annunciator. Shutting the second valve in the same steam line has no further effect on the relays.

Assuming that either valve in the A steam line has been shut, if a valve in either the B or C steam line is also shut then a Half-Scram will result in either RPS A or B respectively. Notice that having MSL A and D valves shut does not result in a Half-Scram. This is also true if only MSL B and C have valves shut. In either of these two cases, the first response of the RPS system will be a Half-Scram. The conclusion to be drawn from this is that any combination of valves may be shut in any two steam lines and the worst case is still only a Half-Scram. No matter the selected combination, any time a valve is closed in the third steam line a Full-Scram will result. Restated, the shutting of the three MSIVs may only result in a Half-Scram while the shutting off of any three steam lines will result in a Full-Scram.

3.6 PCIS Actuation/Bypasses

Unless bypassed automatically or manually, any one of the isolation signals listed in Section 2.5.1 will result in full closure of the Group I Isolation Valves. In addition, this signal will block opening of these valves until reset by operator action. It should be noted that the isolation circuitry does not necessarily discriminate between sensed parameters. If an A or C channel signal exists from a Group I signal, a B or D channel signal from any other parameter will result in a Group I Isolation.

The Unit 2 Main Steam Line High Flow (30%) Isolation is bypassed automatically when the Reactor Mode Switch is in the RUN position. This isolation is required at lower powers (out of RUN) to protect against **pressure regulator failures** which result in excessive Reactor Vessel cooldown caused by the opening of Turbine Bypass Valves. In this instance, the steam flow spike (thermal transit) resulting from the failure could be significant because of the 105% bypass valve capacity. On Unit 1, with only 25% bypass capacity, the failure is limited to a spike of 25% flow and the 30% isolation is not needed.

4.1.2 Surveillance Testing

During normal plant operation, the routine tests performed on the system include monthly Slow Closure Testing of the MSIVs, Full Closure Time Testing and Periodic Isolation Logic Instrumentation and Circuitry Testing.

MSIV Slow Closure Testing (PT-40.2.8) is performed to exercise the valve along its stroke and to verify that the Reactor Protection System recognizes that the valve is out of position. This test is performed at reduced Reactor Power (<80%) to avoid the high steam line flow isolation as sensed in the other three lines when the tested valve is shut.

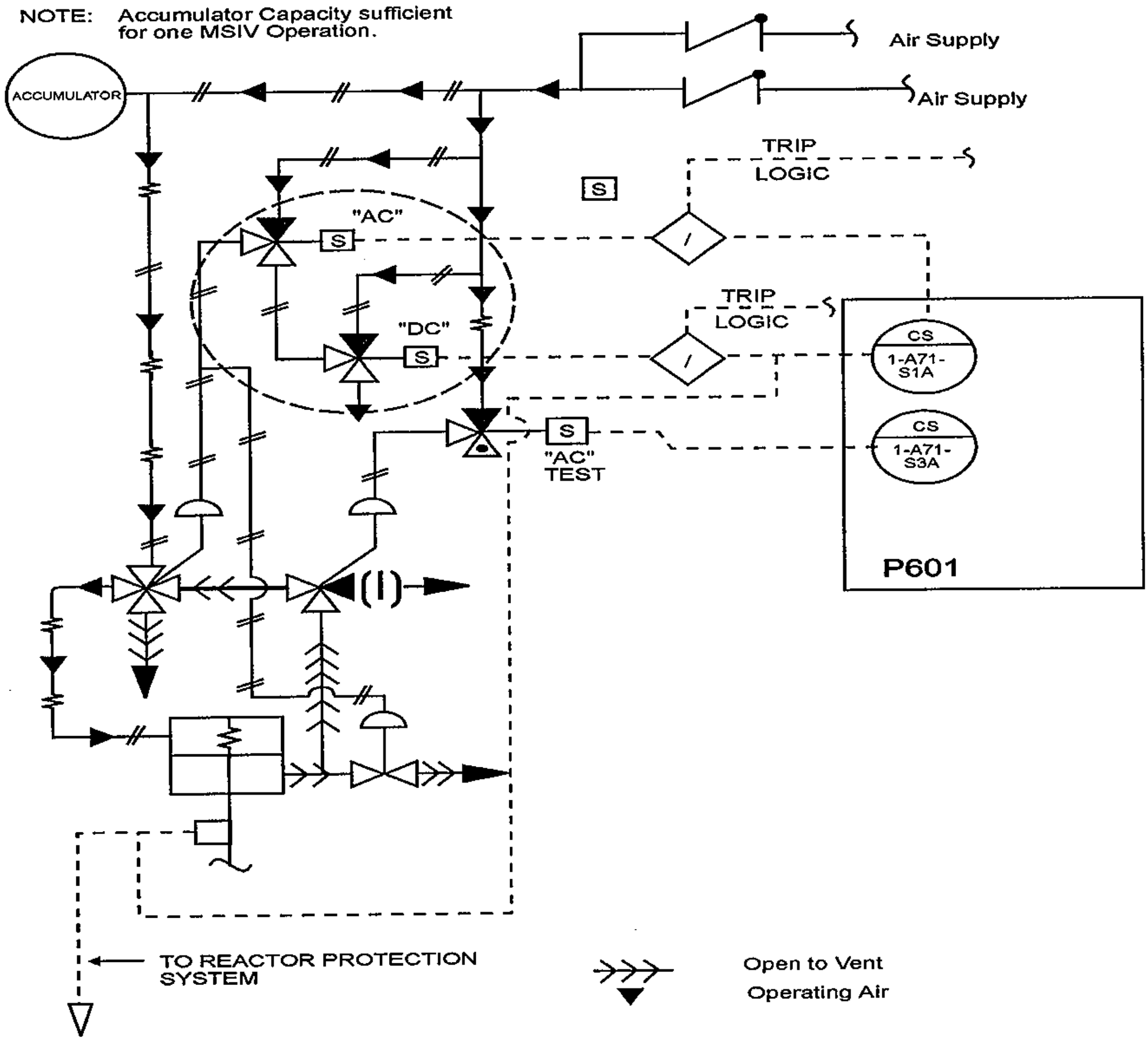
The Full Closure Timing Test (PT-25.1) is the operability test of the Main Steam Line Isolation Valves. This test times the normal closure of the MSIV from switch selection to full close light indication. As discussed before, this stroking must be completed in a 3 to 5 second interval.

0PT-31.1, Non-interruptible Instrument Air System Valve Operability Test verifies proper operation of Division I and II check valves to the Inboard/Outboard MSIV and the SRV accumulators. This test is performed during periods of reactor shutdown when the drywell is accessible. During performance of this test for the MSIV's, Division II(I) non-interruptible instrument air header is isolated. The associated MSIV accumulator drain valve is opened and depressurized. Continued air flow past the affected MSIV accumulator drain valve is verification that the associated Division I(II) non-interruptible instrument air check valve goes to the open position. It should be noted that MSIV accumulator depressurization should take approximately one minute. After verification of check valve operation the affected air accumulator drain valve is closed and capped.

Other less routine testing of the system includes measuring the containment isolation valve seat leakages, verifying the operability of other functions such as SRV lifting and ASSD functions, and post maintenance testing which restores reliability in the operability of the system/components. Most of this testing is associated with post refueling or post maintenance activities.

FIGURE 25-6A

Main Steam Line Isolation Valve Schematic Control Diagram
MSIV Close Mode AC/DC Solenoids Deenergized

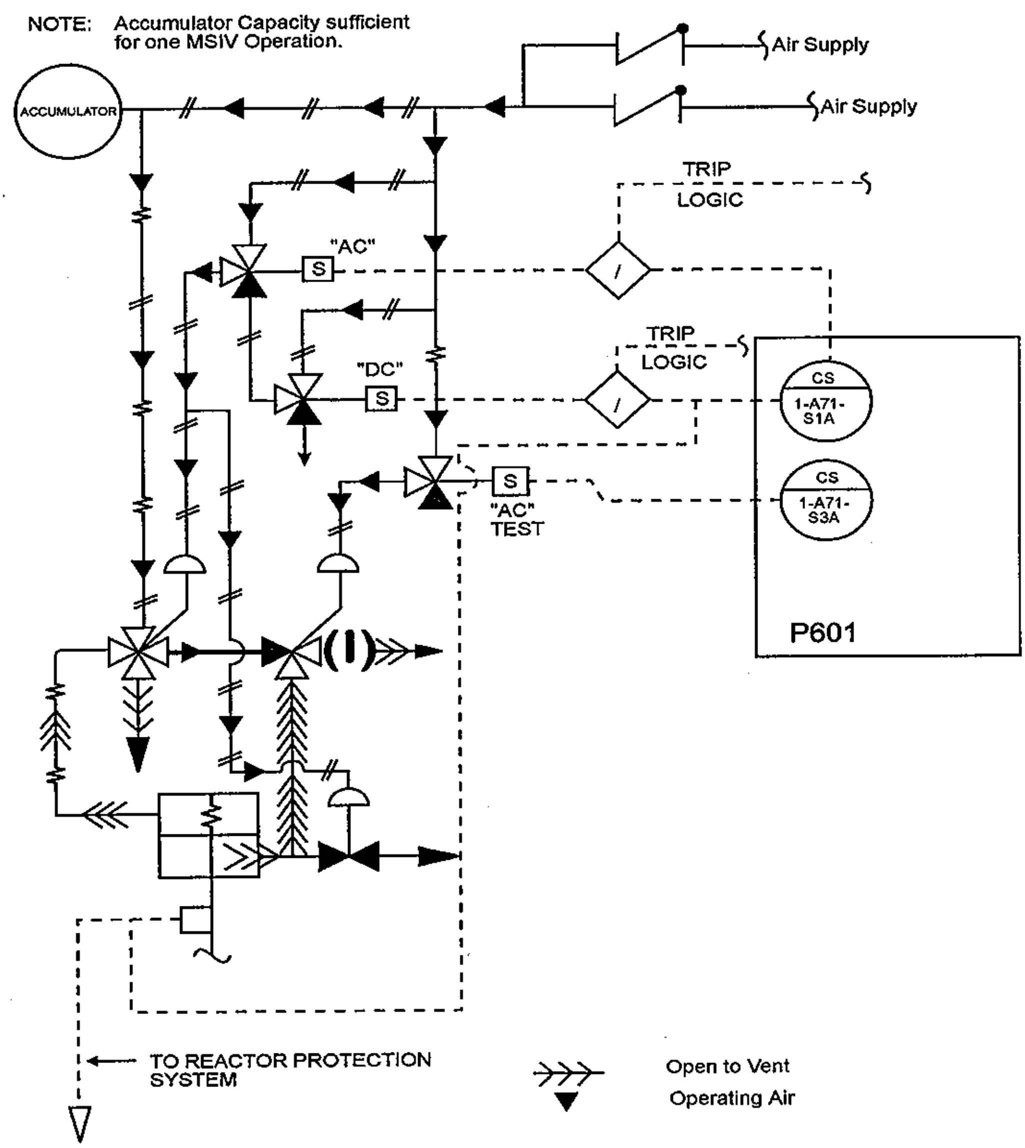


MAIN STEAM LINE ISOLATION VALVE SCHEMATIC CONTROL DIAGRAM
MSIV CLOSE MODE AC/DC SOLENOIDS DEENERGIZED

NOTE:	"CS" in Schematic above is for Valve B21-F022A; for others, see table below.

VALVES REFERENCE	CS TRIP REF. LOGIC	CS REF. OTHER	VALVES REFERENCE	CS TRIP REF. LOGIC	CS REF. OTHER
F022-A	I-A71-S1A	I-A71-S3A	F028-A	I-A71-S2A	I-A71-S4A
F022-B	I-A71-S1B	I-A71-S3B	F028-B	I-A71-S2B	I-A71-S4B
F022-C	I-A71-S1C	I-A71-S3D	F028-C	I-A71-S2C	I-A71-S4D
F022-D	I-A71-S1D	I-A71-S3C	F028-D	I-A71-S2D	I-A71-S4C

FIGURE 25-6C
Main Steam Line Isolation Valve Schematic Control Diagram
MSIV Slow Closure Test Mode Test Solenoid Energized

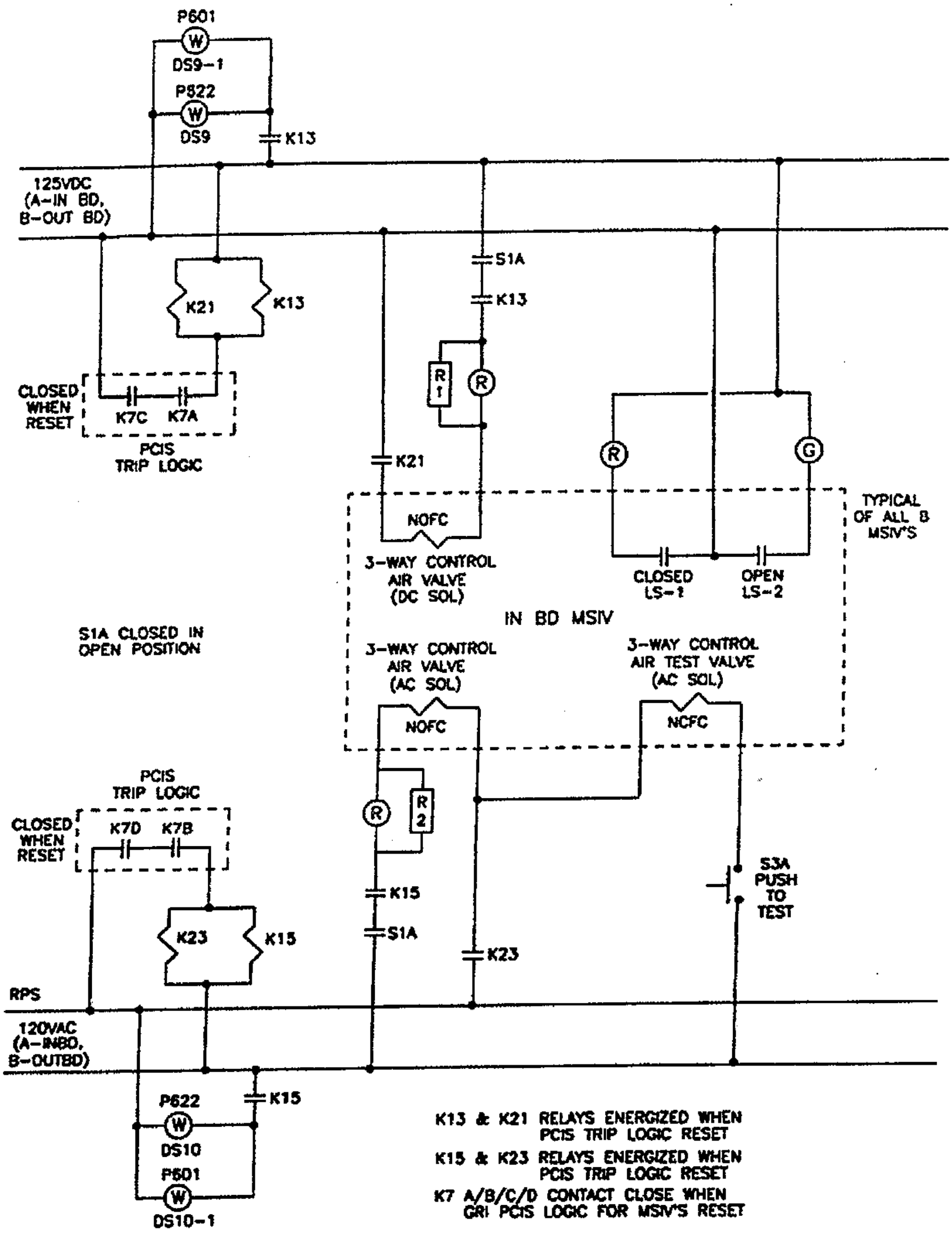


MAIN STEAM LINE ISOLATION VALVE SCHEMATIC CONTROL DIAGRAM
MSIV CLOSE MODE AC/DC SOLENOIDS DEENERGIZED

NOTE: "CS" in Schematic above is for Valve B21-F022A; for others, see table below.

VALVES REFERENCE	CS TRIP REF. LOGIC	CS REF. OTHER	VALVES REFERENCE	CS TRIP REF. LOGIC	CS REF. OTHER
F022-A	I-A71-S1A	I-A71-S3A	F028-A	I-A71-S2A	I-A71-S4A
F022-B	I-A71-S1B	I-A71-S3B	F028-B	I-A71-S2B	I-A71-S4B
F022-C	I-A71-S1C	I-A71-S3D	F028-C	I-A71-S2C	I-A71-S4D
F022-D	I-A71-S1D	I-A71-S3C	F028-D	I-A71-S2D	I-A71-S4C

FIGURE 25-7A
MSIV Control Circuit



Scenario 1, event 2

3.0 PRECAUTIONS AND LIMITATIONS

3.17 Operation at power levels between 23% RTP (Unit 1) or 25% RTP (Unit 2) and 90% RTP without a backup Main Turbine Pressure Regulator may be an unanalyzed condition. Operation at higher power levels is bounded by other transient analyses. Operation at low power levels has a large inherent margin that ensures MCPR is **NOT** exceeded. **WHEN** reactor thermal power is greater than or equal to 23% RTP (Unit 1) or 25% RTP (Unit 2) **AND** less than or equal to 90% RTP **AND** a main turbine pressure regulator is inoperable, **THEN** the inoperable pressure regulator must be restored to operable status within 4 hours. **IF** the pressure regulator can **NOT** be restored operable within 4 hours, **THEN** a power change to less than 23% RTP (Unit 1) or 25% RTP (Unit 2) **OR** greater than 90% RTP within the following four hours must be accomplished to avoid operation in an unanalyzed condition. The Nuclear Fuels (NFM&SA) Section should be notified immediately to analyze and recommend operation without a backup pressure regulator. It is important to note that **IF** reactor power is greater than 90% RTP **WHEN** a pressure regulator is found inoperable, **THEN** reactor power should **NOT** be reduced below 90% RTP. Operation in the permissible power ranges with an inoperable pressure regulator is a degraded condition which must have a time frame established for corrective actions to restore the pressure regulator to operable status. (GE SIL 614)

3.18 Unit 1 Only: Power operation of Unit 1 is limited as follows:

1. Maximum Core Power (due to cross around relief valve capacity)

    a. Less than or equal to 2825 CMWt (2nd Stage Reheat in Service).

    b. Less than or equal to 2752 CMWt (2nd Stage Reheat **NOT** in Service).

2. Main Generator Gross Output (Main Transformer Rating limit)

    a. 955 MWe (Bus 1C **AND** Bus 1D fed from UAT)

    b. 945 MWe (Bus 1C **OR** Bus 1D **NOT** fed from UAT)

4.0 PREREQUISITES

4.1 Reactor is in Mode 1 with Reactor Recirculation pumps above minimum speed.

4.2 The Load Dispatcher concurs with loading plans.

Scenario #1, event 2

5.0 PROCEDURAL STEPS

Initials

5.2 Power Increases

Unit___ Date/Time Started__________/__________

5.2.1 All applicable prerequisites as listed in Section 4.0 are met. _____

NOTE: The following indications should be observed to verify proper response to increased speed demand from a recirculation pump speed controller:

1. Recirculation pump speed increases.
2. Recirculation loop flow increases.
3. Reactor power increases.

NOTE: Turbine load should be increased in accordance with 1(2)OP-26, Figure 3.

NOTE: Procedural steps directing power increases may be performed concurrently with other steps of this procedure.

NOTE: **IF** thermal power is increased more than 15% in one hour, **THEN** reactor coolant shall be sampled in accordance with TR 7.3.7.2 (ODCM Table 7.3.7-1, footnote c).

NOTE: Process Computer Point B018 total core flow and H12-P603 recorder *1/2B21-PDR/FR-R613* will read lower than Process Computer Point *WTCF* as the stability region is approached. Computer Point *WTCF* is the primary indication of total core flow and should be used for stability region compliance.

5.0 PROCEDURAL STEPS

Initials

CAUTION

Reactor recirculation pumps should be operated in accordance with the Flow Control Operation Map. Care should be taken to avoid the regions of possible core thermal hydraulic instability, as specified in the COLR.

Unit 1 Only: The OPRM system monitors the LPRMs for indication of thermal-hydraulic instability when greater than or equal to 25% thermal power **AND** less than or equal to 60% recirculation flow. This system provides alarms and automatic trips as applicable. **IF** the OPRM system is inoperable **AND** operation is within Region A, **THEN** an immediate manual scram is required. **IF** the OPRM system is inoperable **AND** indications of thermal-hydraulic instability are present with operation within Region B, 5% Buffer Region, or the OPRM Enabled Region of the applicable Flow Control Operation Map, **THEN** an immediate manual scram is required.

Unit 2 Only: **IF** the Exclusion Region is entered, an automatic reactor scram will occur. **IF** operations are required in the Monitored Region or the Restricted Region, additional controls are required to be in place per Technical Specifications (Fraction of Core Boiling Boundary 3.2.3 and/or Period Based Detection System 3.3.1.3).

5.2.2 **PERFORM** Attachment 2, each 10% power change increment. ____

R25 5.2.3 **PERFORM** power increases, as directed by the Unit SCO, by withdrawing control rods in accordance with 1(2)OP-07 in the sequence designated by 0GP-10, Attachment 1 and increasing recirculation flow in accordance with Reactor Engineer's recommendation. ____

5.2.4 **IF** Digital Feedwater Level Control System is in 1-ELEM control, **THEN** swap to 3-ELEM control in accordance with 1(2)OP-32. ____

5.2.5 **IF** operating using *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177*, to stabilize feedwater flow, **THEN CLOSE** *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177.* ____

1. **WHEN** *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177*, is closed, **THEN CLOSE** *FW-FV-177 ISOL VLV, FW-V10.* ____

Initials

5.2.6 **PERFORM** 0PT-13.1, Reactor Recirculation Jet Pump Operability, prior to exceeding 25% reactor power. ____

5.2.7 Unit 1 Only: **WHEN** reactor power is between 23% and 28%, **THEN CONFIRM** APRM GAFs are less than or equal to 1.00. ____

5.2.8 **IF** reactor power was decreased to less than 23% (Unit 1) or 25% (Unit 2), **THEN PERFORM** 1(2)PT-01.11, Core Performance Parameter Check, within 12 hours after reaching or exceeding 23% RTP (Unit 1) or 25% RTP (Unit 2). ____

NOTE: Heater drains recirculation should be conducted such that the system will be ready for forward pumping of the heater drains when turbine load reaches 200 MWe.

5.2.9 **IF** secured, **THEN PLACE** heater drains in the recirculation mode in accordance with 1(2)OP-35. ____

5.2.10 Unit 2 Only: **IF** recommended by the Reactor Engineer, **AND** the additional controls of Technical Specifications 3.2.3 and 3.3.1.3 for operating in the Restricted Region have been implemented, **THEN PERFORM** APRM Normal Trip Setpoint setup as follows:

1. **CONFIRM** FCBB is less than or equal to 1.0 using Core Monitor edit program. ____
2. **DEPRESS** the *NORMAL/SETUP* push-button on the FCTR card for each APRM. ____
3. **CONFIRM** NORMAL/SETUP LED is yellow. ____

5.2.11 **IF** SJAE Condensate Recirculation Valve, *CO-FV-49*, is open, **THEN THROTTLE CLOSED** as necessary to maintain Condensate Pump discharge pressure between 150 psig and 190 psig. ____

5.2.12 **IF** Condensate Booster Pump suction pressure approaches 114 psig during power increase **AND** it is desired to maintain HWC supply on liquid, **THEN CONTROL** HWC on liquid supply in accordance with 1(2)OP-59. ____

5.0 PROCEDURAL STEPS

Initials

5.2.13 **NOTIFY** radwaste to perform the following:

1. **PLACE** CDDs, CFDs, and Master Flow Controllers in service as required. _____

2. **PLACE** Hotwell level control in feed and bleed in accordance with 1(2)OP-32, as desired. _____

5.2.14 **ADJUST** Low Load Valve Panel Loaders at IR-TB-13 and IR-TB-14, as main turbine load increases, to increase second stage tube bundle pressure at 15 minute intervals in accordance with 1(2)OP-36, Figure 1. _____

5.2.15 **WHEN** turbine load increases to between 200 MWe and 360 MWe, **THEN COMMENCE** forward pumping of the heater drains in accordance with 1(2)OP-35. _____

5.2.16 **WHEN** turbine load reaches approximately 240 MWe, **THEN ENSURE** *HP TURB 7TH STAGE EXHAUST DRAIN VLVS MVD-MOV-CA-4/3/1/2* are closed. _____

5.2.17 Unit 2 Only: **WHEN** turbine load reaches approximately 240 MWe, **THEN PERFORM** the following:

1. **OBTAIN** Select Rod Insert (SRI) control rod list from the Reactor Engineer. _____

NOTE: The following step must be performed by a licensed person and independently verified by a Senior Reactor Operator.

2. **PLACE** the SRI control rods on the SRI bus recommended by the Reactor Engineer in accordance with 2OP-07. _____/_____ IV SRO

3. **RESET** the APRM setdown. _____

5.0 PROCEDURAL STEPS

Initials

NOTE: The Turbine Stop Valve/Control Valve Fast Closure Reactor Scram MUST be enabled **PRIOR** to exceeding 26% RTP (Unit 1) or 30% RTP (Unit 2). This may be accomplished by annunciator and relay confirmation of automatic enabling **OR** by manually enabling this function by removing fuses.

5.2.18 **PRIOR** to 26% RTP (760 MWT on Unit 1) or 30% RTP (767 MWT on Unit 2), **CONFIRM** Turbine Stop Valve/Control Valve Fast Closure Reactor SCRAM is enabled by performing the following for the applicable Unit:

1. Unit 1 Only: **CONFIRM** *TURB CV FAST CLOS/SV TRIP BYPASS* (A-05, 6-7) is clear. _____

2. Unit 2 Only: **CONFIRM** *TURB CV FAST CLOS/SV/RPT TRIP BYPASS* (A-05, 6-7) is clear. _____

NOTE: The *K9A-D* relays are deenergized when they are at the stop screws.

3. **CONFIRM** relay *C71A(72A)-K9A* on Panel H12-P609 is deenergized. _____

4. **CONFIRM** relay *C71A(72A)-K9C* on Panel H12-P609 is deenergized. _____

5. **CONFIRM** relay *C71A(72A)-K9B* on Panel H12-P611 is deenergized. _____

6. **CONFIRM** relay *C71A(72A)-K9D* on Panel H12-P611 is deenergized. _____

5.0 PROCEDURAL STEPS

5.2.32 Unit 2 Only: **ENSURE** APRM setdowns are reset. _____

R25

> **NOTE:** Control rod withdrawal to the Full Out position in a sequence other than that called for in 0GP-10 shall be documented on Attachment 1.

5.2.33 Unit 1 Only: **INCREASE** reactor power as directed by the Unit SCO, in accordance with the Reactor Engineer's recommendation, to the most limiting of the values stated in Step 3.18. _____

5.2.34 Unit 2 Only: **INCREASE** reactor power to 100% as directed by the Unit SCO, in accordance with the Reactor Engineer's recommendation. _____

5.2.35 **WHEN** unit is at 100% maximum achievable reactor power, **THEN ENSURE** reactor pressure is at rated pressure of 1030 psig utilizing narrow range indications (preferably Computer Point B015 if available). _____

5.2.36 **CONFIRM** core thermal limits are within the prescribed limits of Technical Specifications. _____

Date/Time Completed ____________________

Performed By (Print)	Initials
____________________	_____
____________________	_____
____________________	_____
____________________	_____

Reviewed By: ____________________
Unit SCO

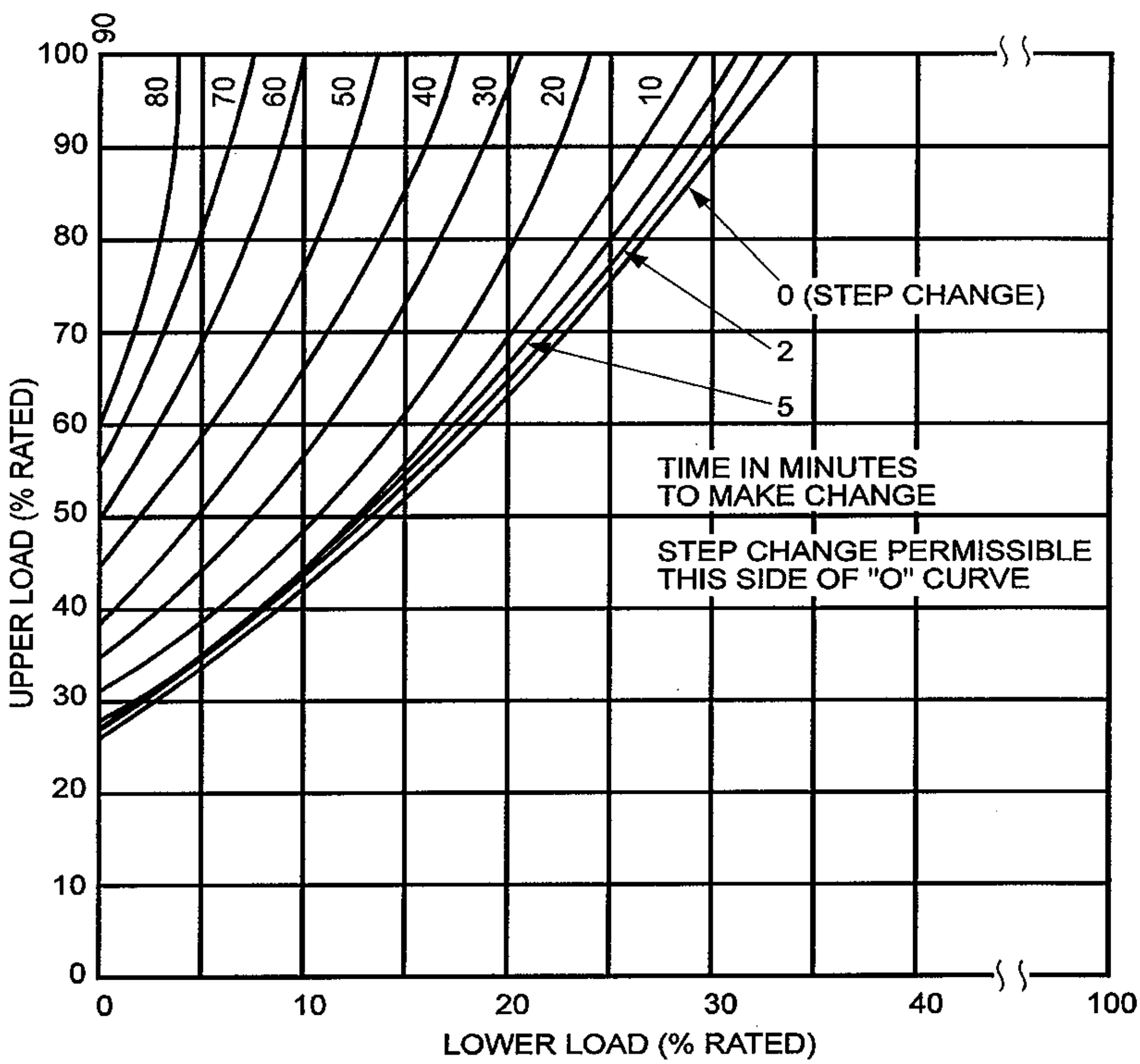
COMMENTS:

____

____

____

FIGURE 3
Page 1 of 1
Time to Make Load Change
Saturated Nuclear Units



EXAMPLE:
POWER CHANGE FROM 10% TO 50% SHOULD BE ACCOMPLISHED AT A RATE SUCH THAT IT WILL REQUIRE AT LEAST 20 MINUTES TO REACH 50% LOAD.

Scenario 1, Event 2

6.0 SYSTEM OPERATION

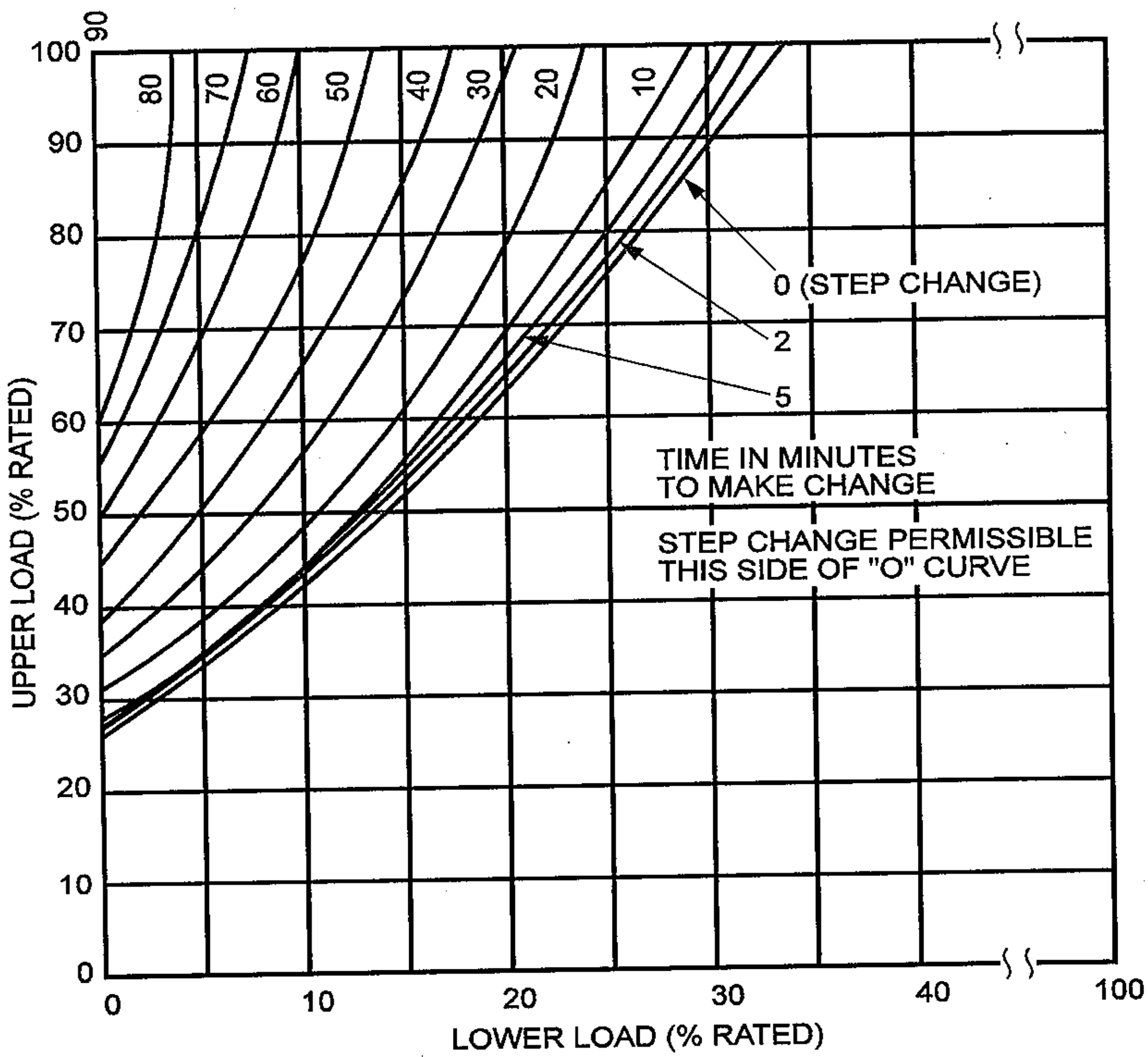
Information Use

During normal full loading of the main turbine, the following parameters should be routinely monitored:

6.1	Operating oil pressure TO-PI-621 Turbine front standard	220 to 250 psig
6.2	Bearing header pressure TO-PI-623 Turbine front standard	23 to 29 psig
6.3	Pump suction pressure TO-PI-622 Turbine front standard	20 to 30 psig
6.4	TBCCW to main turb oil cooler TCC-TIC-615 Panel XU-2	110° to 120°F
6.5	Journal bearing metal temperatures TSI-TR-642 Panel XU-4	170 to 190°F
6.6	Thrust bearing metal temperatures TSI-TR-642 Panel XU-4	130 to 150°F
6.7	Turbine vibration TSI-XR-640 Panel XU-4	Less than 5 mils
6.8	Condenser Vacuum OG-PR-23 Panel XU-2	Less than or equal to 30% load, > 26" Hg Vac Greater than 30% load, > 25" Hg Vac

Scenario 1, event 2

FIGURE 3
Page 1 of 1
Time to Make Load Change
Saturated Nuclear Units



EXAMPLE:
POWER CHANGE FROM 10% TO 50% SHOULD BE ACCOMPLISHED AT A RATE SUCH THAT IT WILL REQUIRE AT LEAST 20 MINUTES TO REACH 50% LOAD.

Scenario 1, Event 2

2.0 REFERENCES

2.16 GE Service and Information Letters: Serials 52; 139, Supplement 2; and 407

2.17 EER 88-0259

2.18 0OI-53, Rod Worth Minimizer (NUMAC-RWM) Operating Instruction

2.19 Control Rod Drive, GEI-924184B

R20 2.20 INPO SER 17-92, Inadequate Control of Testing Results In An Unintended Reactor Power Transient.

R21 2.21 INPO SOER 84-2, Control Rod Mispositioning

2.22 0-FP-50012 Sheet 3, Units 1 & 2 Reactor Manual Control System Elementary Diagram

2.23 2-FP-50012 Sheet 13, Reactor Manual Control Elem Diag. Unit No. 2

2.24 0-FP-50012 Sheet 14, Elem. Diag. Reactor Manual Control Sys

2.25 2-FP-50472 Sheet 2, CRD Sel. Relay IR H12-P616 Conn Diag. Unit No. 2

2.26 0-FP-50472 Sheet 3, CRD Sel. Relay IR Conn Diag. H12-P616 Units 1 & 2.

2.27 0PT-14.2.1, Single Rod Scram Insertion Times Test

3.0 PRECAUTIONS AND LIMITATIONS

3.1 During a hot startup following a reactor Scram from power, extremely high rod notch worths can be encountered due to peak xenon with no moderator voids.

3.2 The reactor should **NOT** be operated with a stable period of less than 100 seconds.

3.3 **IF** single notch withdrawals result in reactor periods approaching 20 seconds, the control rod(s) should be inserted to achieve a stable period of greater than or equal to 100 seconds and the rod withdrawal sequence discontinued until a thorough assessment has been performed by the Reactor Engineer and approved by the Unit SCO.

3.0 PRECAUTIONS AND LIMITATIONS

3.4 **IF** a reactor period of less than or equal to 12 seconds is reached, the reactor shall be shut down until a substantial shutdown margin is achieved. **IF** this is done, at least ten control rods shall be fully inserted past the step in GP at which the short period was experienced. The reactor startup shall be discontinued until a thorough assessment as to the cause/recommendation to prevent recurrence has been made by the Reactor Engineer and approved by the Unit SCO.

3.5 Reactor Coolant System pressure and temperature shall meet the Technical Specification limits prior to withdrawing control rods for an approach to criticality and during critical operations thereafter (see Technical Specification 3.4.9).

3.6 The Reactor Engineer should be present to monitor power flux shaping during non-routine power ascensions.

3.7 Heatup of the Reactor Coolant System with reactor heat should be coordinated with the BOP Operator to prevent the Reactor Coolant System from being heated at a faster rate than the BOP can be placed in service.

3.8 During reactor shutdown, plant cooldown should be coordinated with control rod drive insertion to prevent an inadvertent criticality.

3.9 Coupling integrity of a control rod shall be checked anytime a control rod is fully withdrawn by verifying that the rod does **NOT** reach the overtravel position (see Technical Specification SR 3.1.3.5).

3.10 All rod select push buttons should be deselected whenever rod movement has stabilized to minimize select switch damage from overheating.

3.11 Any time a control rod has been determined immovable or untrippable, determination of the shutdown margin shall be made within 72 hours (see Technical Specification 3.1.3.A).

3.0 PRECAUTIONS AND LIMITATIONS

3.12 Continuous control rod withdrawal should **NOT** be utilized when approaching criticality.

3.13 The RWM shall be operable when reactor power is less than or equal to 10% rated power. **IF** the RWM is bypassed or a control rod is bypassed in the RWM, the Reactor Engineer should be notified prior to further rod movements and a second licensed operator or other qualified member of the technical staff shall verify that the rod sequence is correctly followed in accordance with the applicable GP (see Technical Specification 3.3.2.1.C, 3.3.2.1.D).

R21 3.14 Any deviation from the original withdrawal sequence should be recommended by the Reactor Engineer, authorized by the Unit SCO, and documented on the proper rod sequence checkoff sheet.

3.15 **IF** an uncoupled control rod is recoupled and coupling integrity verified, the control rod should be restricted to the "Notch Out" mode of withdraw operation. It should be removed for filter screen inspection and replacement at the next scheduled outage.

3.16 The following Technical Specification requirements shall be observed for the Reactor Manual Control System:

3.16.1 Section 2.0, Safety Limits (SLs)

3.16.2 Section 3.1.1, Shutdown Margin (SDM)

3.16.3 Section 3.1.2, Reactivity Anomalies

3.16.4 Section 3.1.3, Control Rod Operability

3.16.5 Section 3.1.6, Rod Pattern Control

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

3.16.7 Section 3.2.2, Minimum Critical Power Ratio (MCPR)

3.16.8 Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation

3.16.9 Section 3.3.2.1, Control Rod Block Instrumentation

3.0 PRECAUTIONS AND LIMITATIONS

3.16.10 Section 3.4.9, Reactor Pressure and Temperature (P/T) Limits

3.16.11 Section 3.6.1.1, Primary Containment

3.16.12 Section 3.9.1, Refueling Equipment Interlocks

3.16.13 Section 3.9.3, Control Rod Position

3.16.14 Section 3.10.3, Single Control Rod Withdrawal - Hot Shutdown

3.16.15 Section 3.10.4, Single Control Rod Withdrawal - Cold Shutdown

3.16.16 Section 3.10.6, Multiple Control Rod Withdrawal - Refueling

3.16.17 Section 3.10.8, Shutdown Margin (SDM) Test - Refueling

3.17 The single rod scram test panel was **NOT** designed for multiple single rod scrams for rapid power reduction. **IF** used in this manner, the risk of incurring unauthorized rod patterns is greatly increased, as are potential challenges to the scram discharge volume high water level automatic scram.

R20 3.18 Plant parameters should be closely monitored when control rod movement is performed with the Reactor at or near rated power since such movement can cause thermal limits or rated thermal power to be exceeded.

3.19 Use of hydraulic modules "JMC" test jacks on Panel H12-P610 will disable the seal in logic for the rod drift circuitry for the associated set of rods, causing rod drift alarms, and rod drift indication on matrix display to automatically reset.

3.20 To minimize the possibility of inadvertent control rod misposition when inserting or withdrawing a control rod to an intermediate position (notch positions '02' through '46'), and the control rod is to be moved more than one notch, the following practices **SHOULD** be adhered to:

NOTE: This guidance is waived during emergency conditions.

1. When moving a control rod four notches or more, the control rod **SHOULD** be stopped one notch prior to reaching the intended position and then single notched into the final intended position. This guidance does **NOT** supersede any other requirement to single notch control rods.

3.0 PRECAUTIONS AND LIMITATIONS

2. When moving a control rod three notches or less, the control rod **SHOULD** be single notched for the entire move.

3.21 **WHEN** moving control rods, wait a minimum of 3 seconds after settle function to select another rod for movement to preclude inadvertent rod movement.

4.0 PREREQUISITES

4.1 Reactor Protection System is in operation in accordance with 2OP-03.

4.2 Control Rod Drive Hydraulic System is in operation in accordance with 2OP-08.

4.3 Neutron Monitoring System is in operation in accordance with 2OP-09.

4.4 Radiation Monitoring System is in operation in accordance with 2OP-11.

4.5 120 Volt AC UPS, Emergency and Conventional Electrical Systems are in operation in accordance with 2OP-52.

4.6 Reactor Manual Control System Electrical Lineup is complete in accordance with Attachment 1.

4.7 Reactor Manual Control System Panel Lineup is complete in accordance with Attachment 2.

5.0 STARTUP

I
Information
Use

5.1 Continuous Control Rod Withdrawal

5.1.1 Initial Conditions

1. All applicable prerequisites as listed in Section 4.0 are met.

2. Reactor power is greater than 25%.

OR

3. Continuous control rod withdrawal is desired **AND** Reactor Engineer approval has been obtained to continuously withdraw the control rod.

5.1.2 Procedural Steps

1. **ENSURE** *ROD SELECT POWER* control switch is in *ON*.

2. **SELECT** the desired control rod by depressing its *CONTROL ROD SELECT* push button.

3. **ENSURE** the backlighted *CONTROL ROD SELECT* push button is brightly illuminated **AND** the white indicating light on the full core display is also illuminated.

4. **ENSURE** the *ROD WITHDRAWAL PERMISSIVE* indication has illuminated.

5.1.2 Procedural Steps

R21

5. **CONTINUOUSLY WITHDRAW** the control rod to the position designated on GP pull sheets by holding *EMERGENCY ROD IN NOTCH OVERRIDE, C12A-CS-Z9-S1*, to *OVERRIDE*, while simultaneously holding *ROD MOVEMENT, C12A-CS-Z8-S1,* to *NOTCH OUT.*

6. **MONITOR** control rod position **AND** nuclear instrumentation while withdrawing the control rod.

NOTE: Normal operation of the RMCS should be monitored in accordance with Section 6.0.

7. **IF** the control rod fails to withdraw, **THEN GO TO** Sections 8.1, 8.2, 8.6, or 8.7 to free the control rod **AND RETURN TO** Step 5.1.2.8.

8. **PERFORM** the following for control rods to be withdrawn to an intermediate position:

R21

   a. **BEFORE** the control rod reaches the position designated on GP pull sheets, **RELEASE** *ROD MOVEMENT, C12A-CS-Z8-S1,* and *EMERGENCY ROD IN NOTCH OVERRIDE, C12A-CS-Z9-S1,* control switches.

   b. **ENSURE** the control rod settles into the desired position.

   c. **ENSURE** the rod settle light extinguishes.

5.1.2 Procedural Steps

> **NOTE:** **IF** the rod is uncoupled, **THEN** the four rod display indication will go out for the uncoupled rod **AND** the *ROD OVER TRAVEL* (A-05 4-2) annunciator will illuminate.

9. **PERFORM** the following for control rods to be fully withdrawn:

   a. **WHEN** the control rod reaches position 48, **THEN** **MAINTAIN** the continuous withdraw signal for approximately 3 to 5 seconds **OR** apply a separate notch withdraw signal.

   b. **ENSURE** the control rod does **NOT** retract beyond position 48. (ref. SR 3.1.3.5)

   c. **RELEASE** *ROD MOVEMENT, C12A-CS-Z8-S1,* and *EMERGENCY ROD IN NOTCH OVERRIDE, C12A-CS-Z9-S1,* if used.

   d. **ENSURE** the control rod settles at position 48 **AND** the rod settle light extinguishes.

   e. **ENSURE** the control rod reed switch position indicators agree with the *FULL OUT* indication on the full core display.

R21 10. **WITHDRAW** the remaining control rods, as necessary, utilizing the GP pull sheets and repeating Section 5.1 or 5.2.

11. **IF** movement of control rods is no longer required, **THEN** **DESELECT** the rods by placing *ROD SELECT POWER* to *OFF*.

5.2.2 Procedural Steps

6. **MONITOR** control rod position **AND** nuclear instrumentation while withdrawing the control rod.

NOTE: Normal operation of the RMCS should be monitored in accordance with Section 6.0.

7. **IF** the control rod fails to notch out, **THEN GO TO** Section 8.1, 8.2, 8.5, 8.6, or 8.7, to free the control rod **AND RETURN TO** Step 5.2.2.8.

8. **PERFORM** the following for control rods to be withdrawn to an intermediate position:

R21 a. **WHEN** the control rod reaches the position designated on GP pull sheets, **THEN ENSURE** the control rod settles into the desired position.

b. **ENSURE** the rod settle light extinguishes.

NOTE: **IF** the rod is uncoupled, **THEN** the four rod display indication will go out for the uncoupled rod **AND** the *ROD OVER TRAVEL* (A-05 4-2) annunciator will illuminate.

9. **PERFORM** the following for control rods to be fully withdrawn:

a. **WITHDRAW** the control rod to position 48 using either single notch or continuous withdraw.

b. **MAINTAIN** the continuous withdraw signal for approximately 3 to 5 seconds **OR** apply a separate notch withdraw signal.

c. **ENSURE** the control rod does **NOT** retract beyond position 48. (ref. SR 3.1.3.5)

Scenario 1, event 3

CRD PUMP 2A LO SUCT PRESS

AUTO ACTIONS

1. If low suction pressure condition exceeds approximately 3 seconds, CRD Pump 2A will trip.

CAUSE

1. Plugged suction filter.
2. Low condensate storage tank level.
3. Improper CRD pump suction valve lineup.
4. Circuit malfunction.

OBSERVATIONS

1. CRD Pump 2A is off.
2. CRDHS flow rate decreasing to zero (C12-FI-R606).
3. Cooling water flow rate decreasing to zero (C12-FI-R605).
4. Charging water pressure decreasing to zero (C12-FI-R601).
5. Cooling water differential pressure decreasing to zero (C12-PDI-R603).
6. Drive water differential pressure decreasing to zero (C12-PDI-R602).
7. CRD Pump 2A suction pressure less than 18 inches Hg absolute (C12-PI-R017A as read locally).
8. CRD pump suction filter differential pressure greater than 10 psid (CO-PDIS-1490 as read locally).
9. CST level less than 11 feet (CO-LI-1160A).
10. CRD PUMP 2B LO SUCT PRESS (A-05 5-1) alarm.
11. CRD PUMP INLET FILTER ΔP HIGH (A-05 6-1) alarm.
12. CRD HYD TEMP HIGH (A-05 1-2) alarm.
13. CRD ACCUM LO PRESS/HI LEVEL (A-07 6-1) alarm.

ACTIONS

CAUTION

Reactor power and generator load should be maintained constant during the time period that the CRDHS is not operating (no CRD pumps running).

1. Refer to 0AOP-02.0.
2. If CRD pump suction filter differential pressure is greater than 10 psid, shift CRD pump suction filters and start up the CRDHS per OP-08.
3. If the CST level is low, perform the following steps:
   a. If available, shift the CRD suction from the CST to the Condensate and Feedwater System and startup the CRDHS per OP-08.

ACTIONS (Continued)

b. If the CRD suction was not shifted to the Condensate and Feedwater System, fill the CST per 0OP-31.2 and start up the CRDHS per OP-08.

4. If CST level and CRD pump suction filter differential pressure are normal, verify that the CRDHS valve lineup is correct and startup the CRDHS per OP-08.

DEVICE/SETPOINTS

Pressure Switch C12-PSL-N001A   18 inches Hg absolute

POSSIBLE PLANT EFFECTS

1. If the reactor is in operation and the CRDHS cannot be returned to operation, the CRD temperatures will increase.
2. If the reactor is in operation and the CRDHS cannot be returned to operation, the CRD accumulator pressures will decrease, which may require the reactor to be shutdown.
3. Loss of the CRDHS may result in a technical specification LCO.
4. If reactor pressure is less than 800 psig, improper reactor Scram from control rods.

REFERENCES

1. LL-9364 - 73
2. Technical Specification 3.1.3
3. OP-08, Control Rod Drive Hydraulic System
4. 0OP-31.2, Condensate and Demineralized Water Storage and Transfer System
5. 0AOP-02.0, Control Rod Malfunction/Misposition
6. APP A-05 5-1, CRD PUMP 2B LO SUCT PRESS
7. APP A-05 6-1, CRD PUMP INLET FILTER ΔP HIGH
8. APP A-05 1-2, CRD HYD TEMP HIGH
9. APP A-07 6-1, CRD ACCUM LO PRESS/HI LEVEL

1.0 SYMPTOMS

1.1 Abnormal flux pattern.

R1 1.2 Control Rod found out of intended position.

1.3 Thermal limits approaching or exceeding Technical Specification limit.

1.4 *CRD PUMP 1A(2A) LO SUCT PRESS* (A-05 3-1) annunciator in alarm.

1.5 *CRD PUMP 1B(2B) LO SUCT PRESS* (A-05 5-1) annunciator in alarm.

1.6 *CRD PUMP INLET FILTER ΔP HIGH* (A-05 6-1) annunciator in alarm.

1.7 *CRD ACCUM LO PRESS/HI LEVEL* (A-07 6-1) annunciator in alarm.

1.8 *ROD DRIFT* (A-05 3-2) annunciator in alarm.

1.9 Low control rod drive water pressure (normal pressure is reactor pressure + 260 psig to 275 psig) *C11(C12)-PDI-R602*.

1.10 Low cooling water pressure (normal pressure is reactor pressure + 10 psig to 26 psig) *C11(C12)-PDI-R603*.

1.11 Low charging water header pressure (955 psig @ accumulator) *C11(C12)-PI-R601*.

1.12 Failure of CRD to operate on normal withdraw or insert signal.

1.13 Loss of one or more control rod position indicators in the four-rod group display.

1.14 Loss of one or more control rod indicators on the full core display.

1.15 Double image on control rod position indicators.

2.0 AUTOMATIC ACTIONS

2.1 Possible rod block or select block from a failed reed switch or a loss of power.

2.2 CRD pumps trip after a 3 second delay on low suction pressure.

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

3.1.1 **STOP** any power changes in progress.

3.1.2 **IF** more than one control rod is drifting, **THEN INSERT** a manual scram.

3.2 Supplementary Actions

3.2.1 **MONITOR** core thermal parameters to keep within Tech Spec limits.

R1 3.2.2 **CONTACT** the Reactor Engineer for further control rod movement instructions.

3.2.3 **MONITOR** off-gas radiation **AND NOTIFY** E&RC to take coolant samples if fuel element failure is suspected.

3.2.4 **IF** a control rod is drifting, **THEN GO TO** 1(2)APP-A-05 Window 3-2 for response.

3.2.5 **IF** unable to move control rods, **THEN PERFORM** the following:

1. **IF** the operating CRD Pump has failed, **THEN RESTART** the CRD Hydraulic System following loss of a CRD Pump in accordance with1(2)OP-08.

   a. **IF** reactor pressure is below 800 psig (e.g., during startup or shutdown evolutions), **AND** CRD pressure **CANNOT** be restored with either CRD Pump, **THEN INSERT** a manual reactor SCRAM.

2. **REFER** to Technical Specification 3.1.5 for any control rod scram accumulator required actions.

3. **TRY** to move control rods in accordance with 1(2)OP-07.

4. **CHECK** the following circuits:

-	Unit 1:	120-volt UPS:	Panel V7A, CKT 12 (49' Control Bldg)
		120-volt Inst Power:	Panel 1AB, CKT 2 (Cable Spread)

3.2 Supplementary Actions

- Unit 2: 120-volt UPS: Panel V8A, CKT 12 (49' Control Bldg)

  120-volt Inst Power: Panel 2AB, CKT 2 (Cable Spread)

5. **RESET** any electrical circuit that is tripped.

6. **MONITOR** the following CRD System parameters for possible system leakage or flow control valve failures:

   a. CRD Drive Water Pressure , *C11(C12)-PDI-R602.*

   b. CRD Cooling Water Pressure, *C11(C12)-PDI-R603.*

   c. CRD Drive Temperature, *C11(C12)-TR-R018.*

   d. CRD Charging Water Header Pressure, *C11(C12)-PI-R601*

7. **CHECK** CRD Pump suction and drive water filters for high differential pressure.

8. **CHECK** for air in the drive water header by venting the system in accordance with 1(2)OP-08.

9. **IF** single control rod failure is indicated, **THEN CHECK** the following:

   a. CRD directional control valve filters for plugging (W/R).

   b. Individual HCU power and control logic.

3.2.6 **IF** a loss of control rod position has occurred, **THEN PERFORM** the following:

1. **OBTAIN** control rod positions from the plant process computer or ERFIS.

2. **CHECK** the following electrical circuits:

   - Unit 1: 120-volt UPS Panel V9A, CKT 1 (49' Control Bldg)

   - Unit 2: 120-volt UPS Panel V10A, CKT 1 (49' Control Bldg)

Scenario 1, Event 4

APRM DOWNSCALE

AUTO ACTIONS

1. Rod withdrawal block (bypassed when reactor mode switch is not in RUN).
2. Reactor half-Scram (if companion IRM channel is upscale or inop and the reactor mode switch is in RUN).
3. Computer printout.

CAUSE

1. APRM channel(s) indicate downscale.
2. Circuit malfunction.

OBSERVATIONS

1. APRM recorder(s) on RTGB Panel P603 indicates less than or equal to 4.7 on the 0 to 125 scale.
2. APRM channel(s) downscale (DNSC) white indicating light on RTGB Panel P603 illuminated.
3. ROD OUT BLOCK (A-05 2-2) alarm.
4. The rod withdrawal permissive indicating light will be off.
5. The following alarms may be activated:
   a. NEUT MON SYS TRIP (A-05 4-7) alarm.
   b. REACTOR AUTO SCRAM SYS A (A-05 1-7) or REACTOR AUTO SCRAM SYS B (A-05 2-7) alarm.
   c. LPRM DOWNSCALE (A-06 1-7) alarm.
6. LPRM downscale white indicating lights, for LPRMs associated with affected APRM channel(s), are illuminated.
7. On Control Panel H12-P608, observe the following:
   a. Affected APRM indicates less than or equal to 4.7 on the 0 to 125 scale.
   b. LPRM downscale amber indicating lights for LPRMs associated with affected APRM channel(s) are illuminated.

ACTIONS

1. Compare affected APRM channel indication with the other APRM channels.
2. If the affected APRM channel indication differs from other channels or is erratic, perform the following:

NOTE: The APRM downscale and companion IRM upscale/inop scram channels are:

Channel	Instruments
A1	APRM A <u>AND</u> IRM A, <u>OR</u> APRM E <u>AND</u> IRM E
A2	APRM C <u>AND</u> IRM C, <u>OR</u> APRM E <u>AND</u> IRM G
B1	APRM B <u>AND</u> IRM B, <u>OR</u> APRM F <u>AND</u> IRM F
B2	APRM D <u>AND</u> IRM D, <u>OR</u> APRM F <u>AND</u> IRM H

   a. Refer to Technical Specification 3.3.1.1 and TRMS 3.3 for the APRM operability requirements.
   b. Notify the Unit SCO.
   c. Bypass the affected APRM channel.

ACTIONS (Continued

3. If the APRM downscale condition is caused by LPRM input failure, perform the following:
   a. Bypass the failed LPRMs at Control Panel H12-P608.
   b. Verify that the remaining LPRM inputs to the affected APRM channel meet the minimum requirements of Technical Specifications 3.3.1.1 and TRMS 3.3.
   c. Return the affected APRM channel to service by placing the APRM bypass switch to NEUTRAL.
4. If necessary, reset the half-Scram signal.

DEVICE/SETPOINTS

APRM channel A-F, downscale trip unit	4.7 on the 0 to 125 scale

POSSIBLE PLANT EFFECTS

1. Reactor Scram if one APRM in each RPS trip system is downscale and their companion IRMs are upscale or inop (Run mode only).
2. If an APRM channel is bypassed or inoperable, a technical specification LCO may result.
3. APRM inoperable.

REFERENCES

1. LL-9364 - 94
2. FP-5851 - 47
3. Technical Specification 3.3.1.1 and TRMS 3.3
4. APP A-05 2-2, ROD OUT BLOCK
5. APP A-05 4-7, NEUT MON SYS TRIP
6. APP A-05 1-7, REACTOR AUTO SCRAM SYS A
7. APP A-05 2-7, REACTOR AUTO SCRAM SYS B
8. APP A-06 1-7, LPRM DOWNSCALE

Scenario 1, Event 5

RECIRC PMP A MOTOR VIB HIGH

AUTO ACTIONS

NONE

CAUSE

1. Pump motor imbalance.
2. Pump motor bearing wear.
3. Circuit malfunction.

OBSERVATIONS

1. Recirculation Pump A motor bearing temperatures increasing (Recorder B32-R601 on Control Panel H12-P614).

ACTIONS

1. Monitor Recirculation Pump A motor bearing temperatures on Recorder B32-R601.
2. Attempt to reset the alarm to determine if it was spurious.
3. If motor bearing temperatures are increasing and the vibration alarm cannot be reset, notify Shift Supervisor.

DEVICE/SETPOINTS

Vibration Detector B32-C001A-XSH2833 3 mils

POSSIBLE PLANT EFFECTS

1. Loss of Recirculation Pump A.

REFERENCES

1. LL-9364 - 88

Scenario 1, Event 5

OUTER SEAL LEAKAGE FLOW DETECTION HI

AUTO ACTIONS

NONE

CAUSE

1. Failure of Seal No. 2 (upper seal).
2. Failure of both seals.
3. Closure of seal staging valve.
4. Circuit malfunction.

OBSERVATIONS

1. If Seal No. 2 fails:
    a. Seal No. 2 cavity pressure decreases.
    b. PUMP A SEAL STAGING FLOW HIGH/LOW (A-06 6-3) alarm.
2. If both seals fail:
    a. Both Seal Cavity No. 1 and Seal Cavity No. 2 pressures decrease.
    b. PUMP A SEAL STAGING FLOW HIGH/LOW (A-06 6-3) alarm.
    c. Drywell pressure and temperature may increase.
3. If seal staging valve is closed, Seal No. 2 cavity pressure approaches Seal No. 1 cavity pressure.

ACTIONS

1. Verify that Seal Injection Valve, B32-V22, is open.
2. Verify that Seal Staging Valve, B32-V14, is open.
3. Observe the Recirc Pump A outer seal leakage flow indicator to determine leakage rate (20' El. Reactor Building adjacent to H21-P009).
4. Monitor seal cavity pressures for indication of seal failure.
5. Monitor seal temperatures on Recorder B32-R601 on Control Panel H12-P614 for increasing temperatures.
6. If seal temperatures reach or exceed 200°F, trip Recirculation Pump A and isolate the loop.
7. Monitor the drywell pressure and temperature to determine if increasing.
8. Monitor the drywell equipment drain sump pumps for frequency of operation and pump run time.
9. If drywell pressure and temperature are increasing and/or the drywell equipment drain sump pump operations indicate excessive leak rates, refer to 0AOP-14.0 and notify the Unit SCO.

DEVICE/SETPOINTS

Flow Switch B32-FIS-N5514A	0.5 gpm

POSSIBLE PLANT EFFECTS

1. Loss of Recirculation Pump A.
2. Increased drywell pressure, temperature and activity.
3. Reactor Scram.

REFERENCES

1. LL-9364 - 89
2. AOP-14.0, Abnormal Primary Containment Conditions
3. APP A-06 6-3, PUMP A SEAL STAGING FLOW HIGH/LOW

Scenario 1, Event 5

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless denoted in the **Comments**.

Step 1 - Obtain a current revision of OP-02, Section 8.1.

Current Revision of OP-02, Section 8.1 obtained and verified via NRCS, if applicable.

SAT/UNSAT* _____

PROMPT: If asked as Unit SCO, you concur that speeds be mismatched between the 2A and 2B Recirc Pumps by the maximum allowable mismatch prior to shutting down Recirc Pump 2A

Step 2 – Slowly raise Pump B speed while slowly lowering Pump A speed until a 20% mismatch in speed is achieved.

Recirc Pump B speed 20% above Recirc Pump A speed.

SAT/UNSAT* _____

Step 3 – Place the control switch for Seal Staging Vlv B32-V14 to MAN/OPEN.

B32-V14 switch in MAN/OPEN.

**** CRITICAL STEP ** SAT/UNSAT*** _____

Step 4 – Shutdown Recirc Pump 2A by placing RECIRC MG SET 2A control switch to STOP.
Recirc Pump 2A control switch in STOP.

**** CRITICAL STEP ** SAT/UNSAT*** _____

Step 5 – Close Disch Bypass Vlv B32-F032A.
B32-F032A is full closed.

**** CRITICAL STEP ** SAT/UNSAT*** _____

Step 6 – Close Pump A Disch Vlv B32-F031A
B32-F031A is full closed.

SAT/UNSAT* _____

Step 7 – Determine core flow using point WTCF.
Core flow determined using point WTCF.

SAT/UNSAT* _____

PROMPT: As Unit SCO, direct examinee to raise core flow to at least 30.8 mlb/hr.

Step 8 – Raise speed on Recirc Pump B to raise core flow above 30.8 mlb/hr.

Core flow indicates >30.8 mlb/hr on WTCF.

SAT/UNSAT* _____

Step 9 - Ensure total core flow is <45 mlb/hr.

Total core flow >30.8 but less than 45 mlb/hr.

SAT/UNSAT* _____

Step 10 – Within 5 minutes, open Pump A Disch Vlv B32-F031A and Disch Bypass Valve B32-F032A.

B32-F032A placed to OPEN

**** CRITICAL STEP ** SAT/UNSAT*** _____

NOTE: Recirc discharge and discharge bypass valves should be reopened within 5 minutes. The examinee may not have a chance to complete steps 7-9 prior to the loss of Recirc Pump A and may not have a chance to open the discharge valve.

NOTE: Pump Discharge Valve is a throttle open valve.

NOTE: Recirc Pump 2B field breaker trips 5 seconds after B32-F032A switch is placed to open.

NOTE: No alarms are received when Recirc 2B field breaker trips. The Recirc Pump A only out of service alarm clears. Recirc Pump 2B speed indication goes to 100%.

Scenario 1, event 6

2-AOG-D1 GUARD BED TEMPERATURE HIGH

AUTO ACTIONS

NONE APPLICABLE

CAUSES

1. AOG Building HVAC System malfunction.
2. Reheater outlet temperature high.
3. Fire in guard bed.
4. Guard bed temperature element failure.
5. Circuit failure.

OBSERVATIONS

1. High guard bed temperature indicated by 2-AOG-TI-015 (Local Control Panel H2E) or UR-152 (XU-80).
2. High reheater outlet temperature alarm at Local Control Panel H2E.
3. Low moisture indicated on 2-AOG-MI-013 (Local Control Panel H2E) or UR-155 (XU-80, RED).

ACTIONS

1. Check operation of AOG Building HVAC equipment in accordance with OP-37.8, Section 6.0.
2. Check operation of Reheater, 2-AOG-HTR.
3. Bypass and isolate guard bed by opening 2-AOG-V013 and 2-AOG-V014, then close 2-AOG-V009, 2-AOG-V010, 2-AOG-011, and 2-AOG-V012.
4. If guard bed temperature continues to rise, a guard bed fire is possible. Immediately purge the guard bed with N2 in accordance with OP-33, Section 8.5. Notify the Control Room.
5. If cause of alarm is 2-AOG-TE-015 failure, place 2-AOG-TE-016 in service in accordance with OP-33. If both sensing elements fail, monitor process gas temperature by downstream charcoal adsorber bed temperature indicators.
6. If an equipment or circuit malfunction exists, ensure a WR is submitted.

DEVICE/SETPOINTS

2-AOG-TSH-015	95°F

POSSIBLE PLANT EFFECTS

1. Purging the guard bed requires bypass and isolation of the AOG System.
2. Bypassing the AOG System may cause high stack gas radiation and may result in ODCM Required Compensatory Measures.

R
Reference
Use

8.2 Bypassing the Charcoal Guard Bed

8.2.1 Initial Conditions

1. All applicable prerequisites as listed in Section 4.0 are met.

8.2.2 Procedural Steps

> **CAUTION**
>
> Operating the AOG Charcoal Adsorber System with the charcoal guard bed bypassed may carry over moisture and radioactive long lived particulate daughter products into the charcoal adsorber tanks.

1. **OPEN** the following:

    a. *GUARD BED BYPASS VALVE, AOG-V014*

    b. *GUARD BED BYPASS VALVE, AOG-V013*

2. **CLOSE** the following:

    a. *GUARD BED OUTLET ISOLATION VALVE, AOG-V011*

    b. *GUARD BED OUTLET ISOLATION VALVE, AOG-V012*

    c. *GUARD BED INLET ISOLATION VALVE, AOG-V009*

    d. *GUARD BED INLET ISOLATION VALVE, AOG-V010*

C
Continuous Use

8.4 Drying Charcoal Guard Bed with Nitrogen

8.4.1 Initial Conditions

1. *CHARCOAL GUARD BED DEW POINT, AOG-UR-155,* on Panel XU-80, and *GUARD BED OUTLET DEW POINT, AOG-MI-018,* is greater than 40°F.

2. AOG Charcoal Adsorber System is shut down in accordance with Section 7.0.

8.4.2 Procedural Steps

CAUTION

Operating the AOG Charcoal Adsorber System with the charcoal guard bed bypassed may carry over moisture and radioactive long lived particulate daughter products into the charcoal adsorber tanks.

1. **CLOSE** the following to isolate the charcoal guard bed:

   a. *GUARD BED INLET ISOLATION VALVE, AOG-V009*

   b. *GUARD BED INLET ISOLATION VALVE, AOG-V010*

   c. *GUARD BED OUTLET ISOLATION VALVE, AOG-V011*

   d. *GUARD BED OUTLET ISOLATION VALVE, AOG-V012*

2. **ENSURE** *AOG SYS VLV CONT SEL SW, AOG-CS-3161,* is in *LOCAL* on Panel XU-80.

8.4.2 **Procedural Steps**

3. **OPEN** the following:

   a. *OFFGAS SYS OUTLET PRIMARY ISO VALVE, AOG-XCV-143*

   b. *OFFGAS SYS OUTLET SECONDARY ISO VALVE, AOG-XCV-141*

4. **ENSURE** *AOG PURGE GAS HEATER OUTLET VALVE, AOG-NP-V99,* is open.

5. **PLACE** *AOG PURGE N2 GAS HTR CONTROL SWITCH 2-AOG-CS-3638*, located on Panel H0M in AOG Building 21' N-S hallway next to *AOG-FI-150* rotameter, in *ON.*

6. **OPEN** *NITROGEN PURGE GAS ISOL FOR 1&2 AOG-CV-2981,* using control switch located behind Panel H1E.

7. **OPEN** the following:

   a. *CHARCOAL ADSORBERS BYPASS VALVE, AOG-V039*

   b. *CHARCOAL ADSORBERS BYPASS VALVE, AOG-V061*

8. **IF** the Condenser Air Removal System is shut down, **THEN CLOSE** *AOG SYSTEM BYPASS VALVE, AOG-HCV-102.*

9. **PLACE** *DISCONNECT SWITCH, 2-AOG-EHT-1-DISC-SW*, located on *AOG N2 PURGE GAS HEATER CONTROL PANEL* on AOG roof, in *ON.*

10. **THROTTLE OPEN** *NITROGEN PURGE SUPPLY VALVE TO CHARCOAL GUARD BED D1, AOG-NP-V080,* to maximize nitrogen flow, **NOT** to exceed 50 scfm, as indicated by *OFFGAS SYS OUTLET FLOW, AOG-FI-035.*

8.5.2 Procedural Steps

6. **CLOSE** the following:

   a. *GUARD BED OUTLET ISOLATION VALVE, AOG-V011*

   b. *GUARD BED OUTLET ISOLATION VALVE, AOG-V012*

7. **IF** the Condenser Air Removal System is shut down, **THEN CLOSE** *AOG SYSTEM BYPASS VALVE, AOG-HCV-102.*

8. **PLACE** *DISCONNECT SWITCH, 2-AOG-EHT-1-DISC-SW*, located on *AOG N2 PURGE GAS HEATER CONTROL PANEL* on AOG roof, in *ON.*

9. **THROTTLE OPEN** *NITROGEN PURGE SUPPLY VALVE TO CHARCOAL ADSORBERS, AOG-NP-V079,* until nitrogen purge flow is 50 scfm as indicated by *OFFGAS SYS OUTLET FLOW, AOG-FI-035* **OR** *NITROGEN ROTAMETER, 1/2 AOG-FI-150.*

NOTE:	Charcoal adsorbers are being purged with nitrogen purge gas.

10. **WHEN** nitrogen purging is complete, **THEN CLOSE** *NITROGEN PURGE SUPPLY VALVE TO CHARCOAL ADSORBERS, AOG-NP-V079.*

11. **PLACE** *AOG PURGE N2 GAS HTR CONTROL SWITCH 2-AOG-CS-3638*, located on Panel H0M in AOG Building 21' N-S hallway next to *AOG-FI-150* rotameter, in *OFF.*

12. **CLOSE** *NITROGEN PURGE GAS ISOL FOR 1&2 AOG-CV-2981.*

Scenario 1, Event 7

2.1 Loss of Off-Site Power

2.1.7 The following DC oil pumps start on low header pressure:

1. RFPTs
2. Reactor Recirc. M-G Sets
3. Main Turbine
4. Hydrogen Seal Oil

2.2 Loss of E Bus

2.2.1 Partial Groups 2, 3, 6, 8, and 10 isolation. Loss of Bus E2(E4) will result in a Div I and Div II Group 3 isolation.

2.2.2 One Diesel Generator starts and energizes its respective 4160V E Bus.

2.2.3 Reactor Building HVAC isolates.

2.2.4 Standby Gas Treatment initiates.

2.3 Loss of BOP Bus

2.3.1 One or two Diesel Generators start and energize their respective 4160V E Bus(es).

2.3.2 Hydrogen Water Chemistry isolates on loss of Bus 1(2)D.

2.3.3 The following DC oil pumps start on low header pressure:

1. RFPTs
2. Reactor Recirc. M-G Sets
3. Main Turbine
4. Hydrogen Seal Oil

R1 3.0 OPERATOR ACTIONS

3.1 Immediate Actions

None

3.2 Supplementary Actions

3.2.1 Initial Actions Determination

NOTE: Attachments 1 (Unit 1) and 2 (Unit 2) contain a listing of instrumentation, and the associated power supply, that will be available for use in this procedure.

___ 1. **IF ANY** Diesel Generator has **NOT** started and loaded, **THEN START** the Diesel Generator **AND TIE** it to its respective 4160V E Bus.

___ 2. **DETERMINE AND PERFORM** the appropriate Supplementary Actions Section from Table 1 below:

Table 1

DEENERGIZED BUS	SUPPLEMENTARY ACTIONS SECTION
Loss of Off-Site Power	**GO TO** Section 3.2.2 (Page 6)
BOP Bus 1C	**GO TO** Section 3.2.3 (Page 16)
BOP Bus 2C	**GO TO** Section 3.2.4 (Page 19)
BOP Bus 1D	**GO TO** Section 3.2.5 (Page 22)
BOP Bus 2D	**GO TO** Section 3.2.6 (Page 23)
E Bus E1 or E5	**GO TO** Section 3.2.7 (Page 24)
E Bus E2 or E6	**GO TO** Section 3.2.8 (Page 27)
E Bus E3 or E7	**GO TO** Section 3.2.9 (Page 29)
E Bus E4 or E8	**GO TO** Section 3.2.10 (Page 32)

___ 3. **IF** EOP actions require cross-tieing 4160V or 480V E Buses, **THEN GO TO** Section 3.2.11 (Page 35).

3.2.2 Loss of Off-Site Power

NOTE: Attachment 4 (Page 71) contains a listing of loads supplied from E Buses E1 through E4.

CAUTION

If only one diesel is in service, power restrictions may prevent restarting all systems required by this section of the procedure. The Unit SCO must use discretion in determining what loads to restart depending on existing plant conditions. Maximum Diesel Generator loading is 3850 KW.

___ 1. **IF** while executing this procedure, **ALL** AC power is lost to **EITHER** unit, **THEN BOTH** units **GO TO** 0AOP-36.2.

___ 2. **IF** the SAT was lost due to loss of power on the CP&L System, **THEN PLACE** the *AUTO RECLOSE* switches in *MANUAL*, **AND TRIP** all transmission line PCBs.

___ 3. **IF** the SAT was lost due to a fault and is unavailable, **AND** the switchyard is energized, **THEN ESTABLISH** a Unit Auxiliary Transformer Backfeed in accordance with 1(2)OP-50, **AND PERFORM CONCURRENTLY** with this procedure.

NOTE: Remote CST level indicators will fail downscale if off-site power is lost to both Units or Common A and B Busses do not cross-tie on loss of off-site power to one unit.

___ 4. **IF** CST level indication is not available in the Control Room or Radwaste, **THEN MONITOR** CST level locally while HPCI or RCIC is running with suction from the CST, **AND REPORT** the level every hour.

5. **IF** only one Diesel Generator per unit is operating, **AND** the Motor Driven Fire Pump is operating to maintain ring header pressure, **THEN PERFORM** the following:

   ___ a. At the Diesel Driven Fire Pump local control panel, **PLACE** the Diesel Mode Switch in the *MAN A* or *MAN B* position.

   ___ b. **DEPRESS AND HOLD** the *MANUAL START* push button until the Diesel Driven Fire Pump starts.

3.2.2 Loss of Off-Site Power

___ c. At the Motor Driven Fire Pump local control panel, **TRIP** the Motor-Driven Fire Pump by depressing the *MANUAL RELEASE* push button.

NOTE: Each Nuclear Service Water Pump should automatically start once power is restored to its respective 4160V E Bus.

___ 6. For each 4160V E Bus that is energized, **ENSURE** that its associated Nuclear and Conventional Service Water Pumps are operating as appropriate.

NOTE: Steps 7 through 19 can be performed concurrently or in any sequence that supports existing plant conditions and manpower availability. **NOTE:** Battery Charger operation can be verified by proper voltage indication at the RTGB.

7. For each 480V E Bus that is energized, **PERFORM** the following steps on the affected unit:

___ a. **ENSURE** the appropriate 125V DC Battery Chargers return to service by checking the voltage between 130 and 140 volts.

___ b. **ENSURE** the appropriate 24V DC Battery Chargers return to service by checking the voltage between 24 and 28 volts.

___ 8. **MONITOR** 125VDC Batteries and remove loads from the batteries prior to battery terminal voltage reaching 105 volts.

___ 9. **IF** any 125 VDC Battery terminal voltage reaches 105 volts, **THEN REMOVE** the battery from service.

___ 10. **MONITOR** 24 VDC Batteries and remove loads from the batteries prior to battery terminal voltage reaching 21 volts.

___ 11. **IF** any 24 VDC Battery terminal voltage reaches 21 volts, **THEN REMOVE** the battery from service.

3.2.2 Loss of Off-Site Power

12. **START** the Control Building Ventilation System on the affected unit as follows:

___ a. **VERIFY** the Control Building Instrument Air Compressors are functioning properly.

> **NOTE:** Restarting the following Control Building Ventilation System components requires approximately 100 KW.

b. **ENSURE** at least one of the following Control Room A/C and Supply Fan units is operating:

___ - *CTL ROOM A/C & SUPPLY FAN, 1D-CU-CB* and *1D-SF-CB.* (Sub E5/MCC 1CA)

___ - *CTL ROOM A/C & SUPPLY FAN, 2D-CU-CB* and *2D-SF-CB.* (Sub E7/MCC 2CA)

___ - *CTL ROOM A/C SPARE FAN, 2E-SF-CB.* (Sub E8/MCC 2CB)

c. **IF** available, **THEN START** the following Battery Room Ventilation Fans as required:

___ - *BATTERY ROOM 1A VENT FANS, 1C-SF-CB* and *1C-EF-CB*. (Sub E5/MCC 1CA)

___ - *BATTERY ROOM 1B VENT FANS, 1B-SF-CB* and *1B-EF-CB*. (Sub E6/MCC 1CB)

___ - *BATTERY ROOM 2A VENT FANS, 2C-SF-CB* and *2C-EF-CB*. (Sub E7/MCC 2CA)

___ - *BATTERY ROOM 2B VENT FANS, 2B-SF-CB* and *2B-EF-CB*. (Sub E8/MCC 2CB)

3.2.2 Loss of Off-Site Power

13. **RESTORE** Drywell Cooling in accordance with the following steps:

> **NOTE:** Deenergized RBCCW pumps will start when power is restored by the Diesel Generators if their control switches are in *AUTO* and a low pressure is sensed, or if their control switches are in the *ON* positions.
>
> **NOTE:** Operating each RBCCW pump will require approximately 48 KW.

___ a. **IF** all three RBCCW pumps are running, **THEN STOP** one RBCCW pump, **AND PLACE** its control switch in *AUTO*.

___ b. **IF** only one RBCCW pump is running, **THEN**, if available, **START** a second RBCCW pump.

> **NOTE:** Each Drywell Cooler will start automatically when its MCC is energized while a SCRAM signal is present.

___ c. **ENSURE** that all available Drywell Coolers on the affected unit are operating.

> **NOTE:** The Nuclear Service Water header supply valves to RBCCW, *RBCCW HXS SW INLET VLV, SW-V103*, and *RBCCW HXS SW INLET VLV, SW-V106* will be closed due to a LOOP signal.
>
> **NOTE:** The following steps transfer RBCCW heat exchanger cooling water from the Nuclear Service Water header to the Conventional Service Water header.

___ d. **VERIFY** that the Conventional Service Water header is available.

___ e. **CLOSE** the *NUC HDR TO RBCCW HXS SPLY VLV, SW-V193.*

___ f. **OPEN** the *CONV HDR TO RBCCW HXS SPLY VLV, SW-V146.*

3.2.2 Loss of Off-Site Power

___ 14. **IF** the opposite unit has power to its Instrument and Service Air Compressors, **THEN OBTAIN** the opposite Unit SCO's permission **AND CROSS-TIE** air by opening *UNIT 1 CROSSTIE VALVE, 2-SA-V7*.

___ 15. **IF** opening the service air receiver cross-tie valve adversely affects the opposite unit, **THEN CLOSE** *UNIT 1 CROSSTIE VALVE, 2-SA-V7*.

16. **IF** Suppression Pool Cooling has been directed by the EOP **AND** only one Diesel Generator is available on the unit, **THEN INITIATE** Suppression Pool Cooling as follows:

> **NOTE:** Placing a loop of RHR in Suppression Pool Cooling requires approximately:
>
> - 1100 KW without an RHRSW Booster Pump.
> - 1700 KW with an RHRSW Booster Pump.

___ a. **CLOSE EITHER** *SW TO TBCCW HXS OTBD ISOL, SW-V3* (MCC 1(2)XB), **OR** *SW TO TBCCW HXS INBD ISOL, SW-V4* (MCC 1(2)XA).

> **NOTE:** RHR Service Water Booster Pumps may be started depending on power availability.

> **CAUTION**
>
> Restarting fluid systems that have lost power may cause severe water hammer and possible damage to piping and supports. Consideration should be given to filling and venting these systems prior to returning them to service.

___ b. **START** RHR Service Water in accordance with 1(2)OP-43.

___ c. **START** Suppression Pool Cooling without keepfill in accordance with 1(2)OP-17.

3.2.2 Loss of Off-Site Power

17. **IF** Sub E5 is deenergized, **THEN SHIFT** the Manual Bus Transfer (MBT) devices for the following panels to the alternate power source: (Sub E6/DP 1E6)

___ a. Panel 1AB (Control Building 23')

___ b. Panel 31AB (Control Building 49')

___ c. Panel 1AB-RX (Reactor Building 20')

18. **IF** Sub E7 is deenergized, **THEN SHIFT** the Manual Bus Transfer (MBT) devices for the following panels to the alternate power source: (Sub E8/DP 2E8)

___ a. Panel 2AB (Control Building 23')

___ b. Panel 32AB (Control Building 49')

___ c. Panel 2AB-RX (Reactor Building 20')

NOTE:	Restarting RPS will require approximately 45 KW.

___ 19. **START** RPS MG Sets A and B in accordance with 1(2)OP-03.

NOTE:	Restarting CRD requires approximately 200 KW.

___ 20. **START** the CRD System in accordance with 1(2)OP-08.

3.2.2 Loss of Off-Site Power

> **NOTE:** Restarting Reactor Building ventilation requires approximately 365 KW.
>
> **NOTE:** Both RPS buses must be energized to allow restoration of Reactor Building ventilation.
>
> **NOTE:** In order to open the Reactor Building isolation dampers, and to restart Reactor Building HVAC, it may be necessary to cross-tie E Buses.

21. **START** Reactor Building HVAC as follows:

___ a. **IF** *PROCESS OG VENT PIPE RAD HI-HI* (UA-03 5-4) annunciator is in alarm, **AND** is **NOT** the result of a valid high radiation signal, **THEN PLACE** *CAC PURGE VENT ISOL OVRD, CAC-CS-5519*, in *OVERRIDE.*

b. **RESET** the following Reactor Building Ventilation Radiation Monitors on Panel H12-P606:

___ - *PROCESS REACTOR BLDG VENTILATION RADIATION MONITOR "A", D12-RM-K609A*

___ - *PROCESS REACTOR BLDG VENTILATION RADIATION MONITOR "B", D12-RM-K609B*

c. **DEPRESS** the following Isolation Reset Groups push buttons:

___ - *ISOLATION RESET GROUPS 1, 2, 3, 6, 8, A71-S32*

___ - *ISOLATION RESET GROUPS 1, 2, 3, 6, 8, A71-S33*

___ d. **ENSURE** Instrument Air header pressure is greater than 95 psig.

3.2.2 Loss of Off-Site Power

> **CAUTION**
>
> Attempting to open the Reactor Building Ventilation Isolation Valves when the latches are engaged, will damage the latch/damper assemblies.

e. **DISPATCH** an Auxiliary Operator to **ENSURE** the latching mechanisms are disengaged for the following valves:

___ - *RB VENT INBD SUPPLY ISOL VALVE, A-BFIV-RB.*

___ - *RB VENT OTBD SUPPLY ISOL VALVE, B-BFIV-RB.*

___ - *RB VENT INBD EXHAUST ISOL VALVE, C-BFIV-RB.*

___ - *RB VENT OTBD EXHAUST ISOL VALVE, D-BFIV-RB.*

___ f. **OPEN** *RB VENT INBD ISOL VALVES, A-BFIV-RB* and *C-BFIV-RB.*

___ g. **OPEN** *RB VENT OTBD ISOL VALVES, B-BFIV-RB* and *D-BFIV-RB.*

> **NOTE:** Operating one (1) set of Reactor Building Ventilation Fans requires approximately 120 KW of load.

___ h. **START** three (3) sets of Reactor Building Ventilation Fans in accordance with 1(2)OP-37.1 to maintain Reactor Building static pressure negative.

___ i. **IF** HPCI or RCIC is not operating, **THEN SHUT DOWN** the running SBGT trains in accordance with 1(2)OP-10.

> **NOTE:** Restarting RWCU requires approximately 40 KW.
>
> **NOTE:** Both RPS buses must be energized to allow the restart of RWCU.

___ 22. **START** RWCU in accordance with 1(2)OP-14.

3.2.2 Loss of Off-Site Power

NOTE: Starting a Fuel Pool Cooling pump will require approximately 50 KW.

___ 23. **START** Fuel Pool Cooling in accordance with 1(2)OP-13.

___ 24. **DIRECT** an Auxiliary Operator to close the Main Turbine Lube Oil Reservoir *SIGHT OVERFLOW ISOLATION VALVE, LO-V23*, to prevent a conditioner oil spill.

NOTE: Turbine Building Sump and Radwaste Building Sump pumps will **NOT** be available until Off-site Power is restored.

___ 25. **MONITOR** sump levels locally, **AND IMPLEMENT** appropriate actions to control level in the sumps.

NOTE: Transfer Switch (LJ4) is located in the Unit 2 Control Building Cable Spread Area El. 23' on the west wall behind the ERFIS Power Distribution Panels next to the Lighting & Communications UPS Inverter.

___ 26. **IF** the chimney obstruction (stack) lights were deenergized from the Loss of Off-site Power, **THEN DIRECT** an Auxiliary Operator to place *TRANSFER SWITCH FOR FILTER HOUSE & STACK LIGHTING* (LJ4) in the *AUX FEED* position (Sub E8/DP E12) to reenergize the chimney obstruction (stack) lights.

___ 27. **IF** the meteorological (microwave) tower lights were deenergized from the Loss of Off-site Power, **THEN DIRECT** an Auxiliary Operator to the Switchyard Relay House to verify that the Microwave Tower Power Transfer Switch (1K1) shifted to the ALT position (Sub 1SY/MCC SYA).

Facility: Brunswick Scenario No.: 1 Op-Test No.: ______

Title:Medium Break LOCA inside containment with Loss of Offsite Power and failure of one EDG

Examiners: ____________________ Operators: ____________________

____________________ ____________________

Initial Conditions: The crew assumes the shift with the plant at 79% power, BOL. HPCI is OOC, the ~~#1~~ APRM failed low and is bypassed. One CRD is inoperable; stuck at position 48. Severe weather has been reported in the area. The previous shift completed stroking of all MSIVs except "B" and "C" inboards

?

T.S. ?
SID ?
calc. d
would n
give as
C.T. if
noted he

Turnover: Night orders include direction to slow stroke the "B" and "C" inboard MSIVs (full closed) and then to increase power to 100%. The Reactor Engineer recommends using Recirculation Flow for the power increase.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N(BOP)	Slow stroke "B" and "C" Inboard MSIV's (full closed)
2	N/A	R(RO) (SRO)	Increase power to 100%.
3	MRD018F	C (RO)	"A" CRD Pump suction filter plugged.
4	MNI037F	I (RO)	#2 APRM fails low.
5	MRC007F MRC009F	C (RO) (SRO)	"A" Recirc Pump Seal Leak.
6	MCN017F	I (BOP) (SRO)	~~AOG Guard Bed Fire~~
7	MEE032	M(ALL)	Loss of Offsite Power
8	MDG002F	C(BOP)	One EDG fails to start
9	MNB009F	M (ALL)	1000 GPM leak inside drywell

do not allow one valve to close full (>600gpm leak)

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

Op-Test No.: _____ Scenario No.: 1 Event No.: 1

Event Description: Perform SlowFull Stroke Test for the "B" and "C" Inboard MSIVs. The surveillance is complete for the remaining MSIV's. -Verify sig. margin to flow biased scram. - should be ok for slow closure @ 79% procedure says <8

Time	Position	Applicant's Actions or Behavior
	SRO	Direct the BOP to perform the Quarterly MSIV functional surveillance for the "B" and "C" Inboard MSIV's.
		.
		obtains & reviews procedure XXX - XXX -XXX
	BOP	Review Precautions and Limitations. Instruct RO to monitor Rx Pressure and Main Steam Line Flow during valve stroking.
	BOP	Confirm that applicable relays are energized. -put designation for relays, if checked in Control room.
		not observed in simulator
		Confirm that MSIV lights/indications are illuminated.
	BOP	Direst SSS to check relay status on Panel XXX & Panel XXX — relay n
		Place/Confirm applicable MSIV control switch in OPEN.SLOW TEST
		.
		Take MSIV Vlv Test Switch to TEST and HOLD. Should show
		Valves one @ a time. ie. show 'B' MSIV with
		all steps then same steps for 'c'. only do one
	BOP	With RO monitoring Reactor pressure and MSL flow and SSS monitoring relay status at PXXX and PXXX, the BOP will release the Test Switch immediately upon any one of the following:
	RO	1) MSL flow change observed } both should change
	RO	2) Reactor pressure change observed. } MSL flow on other line
		will ↑ & pressure will ↑
		When MSIV is completely closed OR precaution XXX is met, THEN release test switch

Event Description: Perform SlowFull Stroke Test for the "B" and "C" Inboard MSIVs. The surveillance is complete for the remaining MSIV's.

Time	Position	Applicant's Actions or Behavior
	BOP	• Verify appropriate relays are energized. / relay no. ✓
	RO	• If required, reset the Half Scram. (will this be required for 'B' or 'C' not required
	SRO	• Review completed surveillance Acceptance Criteria.
		NOTE: During the scenario, the applicant will only be required to perform ~~2~~ valves. only do one valve F022A
		'B' & 'C'
		As stated before these steps should be repeated
		2x, once for each valve. However, You
		May be able to see what you want to see
		with only 1 valve. ie. Maint. stroke
		test only one.

Op-Test No.: _____ Scenario No.: 1 Event No.: 2 Page 4 of 19		
Event Description: Increase power to 100%. Note: The next event (CRD filter clogged) will be initiated when the Reactivity Manipulation has been satisfied or at the Lead Examiner's discretion Verify significant margin to instability region of power/flow MAP.		

Time	Position	Applicant's Actions or Behavior
	SRO	1. Ensure power increase is acceptable to Load Dispatcher
1.	need ramp rate.	2.Direct the RO to increase power in accordance with 0GP-12. He will specify using the Reactor Recirculation Flow.
2.	If >15% power chem. sample.	3. He will direct the CRO to monitor Turbine Operation in accordance with OP-26, Figure 3.
	RO	1) Review Precautions and Limitations and Prerequisites in 0GP-12.
		2) Ensure steps 5.2.1 through 5.2.34 have been completed ? *
		3) Increase Recirculation Pump speed(s) and monitor Reactor Parameters. / Match speeds or flow / Power, Pressure, level.
		4) When at 100% power ensure reactor pressure is 1030 psig.
		5) Confirm core thermal limits are per TS
	BOP	1 Check Precautions and Limitations in OP-26. steps - ones that allow
		2. Monitor Turbine Power Increase per OP-26. you to obtain insight into
		operator performance (actions
	RO	After power increase has been completed, complete filling out page 28 of 0GP-12).
		~~Note: at discretion of ok Examiner may put next fault in before~~
	SRO	Review and sign 0GP-12, page 28
	*.	I would not expect the operator to verify all items unless
		this is a startup. Will hand procedure
		with completed steps signed off.

Op-Test No.: _____	Scenario No.: 1	Event No.: 2	

Event Description: Increase power to 100%. Note: The next event (CRD filter clogged) will be initiated when the Reactivity Manipulation has been satisfied or at the Lead Examiner's discretion

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 3	

Event Description: "A" CRD Pump suction filter plugged.
Execute malfunction MDR018F per Lead Examiner direction.

Time	Position	Applicant's Actions or Behavior
	RO	1. Observe alarm "CRD Pump Inlet Filter DP High" alarm and monitor CRD parameters.
		2. Review Alarm Response Procedure A-05 5-1.
		1a. stops power increase
	SRO	1. Direct RO to complete ARP actions spell out actions
		2. Dispatch an AO to locally monitor filter DP and pump suction pressures.
	RO	1. Observe running CRD pump trips and "CRD Pump 2A Lo Suct Press" alarm (Note: Other CRD alarms will come in as well)
		2. Refer to $\boxed{\text{0AOP-02-0}}$ - actions. / observable actions.
	T.S. LCO?	3. Direct the AO to shift CRD Pump suction filters (5min to shift?)
		4.When advised (simulator operator) that the CRD suction filters have been shifted, Re-Start the 2A CRD Pump. (why not start the other pump?
		5. Observe CRD parameters return to "normal".
		6. Request the AO ensure the CRD suction filter is less than 3 PSID.
	SRO	Note shifting of CRD filters in log and initiate MR to replace filters in clogged unit.

B pum[p]
OOC

Two or more occur ala[rms]

20 minutes to restore charging water

Op-Test No.: ____ Scenario No.: 1 Event No.: 3	

Event Description: "A" CRD Pump suction filter plugged.
Execute malfunction MDR018F per Lead Examiner direction.

Time	Position	Applicant's Actions or Behavior
	RO 1.	Pump trip - diagnose high suction pressure
	RO 2.	Dispatch operator to switch suction filters or start 2nd pump. (?) will occur quick.
	RO 3.	Accumulator alarms @ ~3 min
	RO 4.	high CRD temp alarms
	SRO 5.	T.S. on accumulator alarms / No CRD pump on. 15 min LCO (>3?) → should be >3 or now 2 since 1 rod stuck @ 48? - T.S. require SDM calc. to show ok.
	RO. 6.	Suction filter swap / restart pump
		If you want a T.S. call should ~~have~~ prevent CRD from starting for several min. to verify T.S. LCO entered. (short duration LCO)
		• allow them to get back before scram via T.S.
		should take several minutes to ~~get~~ for A.O to shift suction filters

Op-Test No.: ______ Scenario No.: 1 Event No.: 4	Page 8 of 19
Event Description: ~~#2~~ B APRM Fails Low	
Insert malfunction MNI037F at direction from Lead Examiner	

Time	Position	Applicant's Actions or Behavior
	RO	1 Observe "APRM Downscale" alarm and notify SRO
		2. ~~Complete actions in~~ obtains & enters ARP A-06 2-7
	3a. verify rod block.	3. Observe other APRM channels are functional (except #1)
		4. Bypass the ~~affected~~ 'B' APRM channel ~~by placing joystic to # 2 position~~
	SRO	1. Consult TS 3.3.1.1.
		2. Ensure no other half scrams are present
		3. Direct RO (or BOP) to place the ~~appropriate~~ 'B' APRM Channel to TRIP ~~/ should be manual scram. If APRM not functioning~~
		~~how do you know that you will get scram - must verify lights.~~
	RO	Observe half scram, verify rod block clears.
	SRO	1. Ensure TS are satisfied and log information
		2. Initiate MR to get APRM ~~#2~~ 1 repaired
		Is this APRM on the same channel? If so then can not bypass. If on other chanel then why place in trip? (TS 2/chanel?)
		If 'A' is bypassed and want a 1/2 scram then 'C' or 'E' must fail downscale. If on same chanel how will you get rid of rod block out since you can not bypass? MODE switch out of operate? ~~If on same chanel~~

Op-Test No.: _____ Scenario No.: __1__ Event No.: __5__

Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC007F and MRC009F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	RO 2	~~1~~. Observe changing seal DPs and notify SRO that both seals appear to have failed or are failing.
	1	~~2~~. Observe "Outer Seal Leakage Flow Detection Hi" alarm and notify SRO. *Note: This should be a clear indication that the seal has failed and that a discharge into containment is occurring*
	SRO	1. Direct RO to complete actions in ARP (A-06 5.3)
		2. Direct BOP to monitor containment parameters
	BOP	Observe Drywell pressure and temperature and sump levels all increasing and advise SRO.
	SRO	1. Direct RO to reduce power in anticipitation of tripping "A" Recirc Pump. *Notes: Pump should be tripped and isolated if seal temperatures reach or exceed 200 F (A-06 5-3) SRO may direct to Transfer per OP-02. If so refer to LOT-SIM-JP-002 Steps 1-10*
		2. Call dispatcher and advise of power reduction/possible scram
XXX	RO*	1. Reduce power as necessary per 0GP-12 Note: Should insert rods per 0GP-10, but may just increase Recirc Pump "B" to max before reducing flow on "A". (T.S. issue?) should have flows matched
	*SRO may direct the BOP to Trip and Isolate the pump if the RO is driving rods	2. Trip "A" Recirc Pump before 200F seal temperatures are reached. This should be done without a Turbine Trip from High RPV water level **(Critical Task)** why critical?
		3. Isolate "A" Recirc Pump **(Critical Task)** ← Yes - stop leak
		4. Ensure Precautions and Limitations of 0GP-12 are satisfied

will enter region 1

This may result in an auto scram if in Region I from OPRM. starting from reduced flow.

Op-Test No.: _____	Scenario No.: 1	Event No.: 5	Page 10 of 19

Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC007F and MRC009F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	BOP/RO	1. If Containment parameters exceed EOP entry conditions advise SRO.
		- start available DW cooling / do not vent.
	SRO	1. Notify Operations Management and Reactor Engineer of power reduction.
		2. Ensure TS are satisfied at new plant configuration — T.S. on single loop. APRM setc
		3. Log events
		~~4. Check if any E-Plan reporting is required Note: No reporting will be required at this point~~

Op-Test No.: ____ Scenario No.: 1 Event No.: 5		Page 11 of 19
Event Description: "A" Recirc Pump Seal Leak. Enter malfunctions MRC007F and MRC009F when directed by Lead Examiner.		

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 6		Page 12 of 19
Event Description: AOG Guard Bed Fire. Execute malfunction MCN017F when directed by Lead Examiner.		

Time	Position	Applicant's Actions or Behavior
	BOP	1 Acknowledge receipt of "2-AOG-D1 Guard Bed Temperature High" alarm and advise SRO
		2. Complete observation steps of UA-49 2-7
		3. Bypass and Isolate guard bed (Action 3 of UA-49 2-7) list
		4. Monitor guard bed temperatures.
		5. Monitor Off Gas and Stack Radiation Monitors
	SRO	1.Based on hearing that the guard bed temperatures continue to rise after isolation, he should direct an AO to "purge the guard bed with N2 in accordance with OP-33, Section 8.5" list steps
		2. Notify OPS Management and HP of the problem
		Note: Once the order is given (to the AO) to purge the guard bed, MCN017F will be removed. This event will be considered complete when temperatures of the guard bed begin to decrease.

Op-Test No.: _____ Scenario No.: 1 Event No.: 6

Event Description: AOG Guard Bed Fire. Execute malfunction MCN017F when directed by Lead Examiner.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 7		
Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.		

Time	Position	Applicant's Actions or Behavior
	SRO	1. Direct actions of EOP-01-RSP. Need AOP-36.1 to see actions
		2. Direct actions of AOP-36.1. *Note: Although the loss of one EDG to start is considered a separate event in the D-1 it is combined into this event with the BOP to take additional actions for manually restarting the EDG.*
		3. If/when suppression pool temperature exceeds 95 F, enter EOP-02-PCCP.
	RO	1. Complete actions of EOP-01-RSP. Note: The RO should recognize that not all rods are full in and should advise the SRO
		2. Monitor and control RPV level and pressure using RCIC and SRVs
		3. Place RHR in suppression pool cooling
		4. Start CRD per OP-08
	BOP	1. Recognize/announce that one EDG did not start
		2. Complete Actions of AOP-36.1
		3. Ensure NSW Pumps running, Start CSW Pumps to support RCC
		4. Ensure battery chargers operating
		5. Start Control Room and battery Room HVAC
		6. Restore Drywell Cooling.
	SRO	1. When advised of one rod full out he should recognize that the reactor will remain shutdown under all conditions and NOT go to "Level/Power Control" ~~(Critical Step)~~ ?
		2. When advised that one EDG did not start he should direct the BOP to try a manual start

Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.

Time	Position	Applicant's Actions or Behavior
	BOP	1. Manually start the failed EDG. Note: The BOP will be "permitted" to manually start the EDG but it will immediately Trip on differential current when he attempts to close the output breaker
		2. Attempt to close breaker (insert malfunction MDG024F)
		3. Recognize that the Differential Fault is serious and probably means the EDG is lost
		4. Direct an AO to investigate the fault.
		Note: Once the BOP recognizes the EDG is "gone" step into the next event (1000 GPM leak)
	1.	~~Motor driven or steam driven FW? I would~~
		~~expect MSIVs to stay open & FW (steam) to~~
		~~supply loads until~~ No Condensate.

Event Description: Loss of Offsite Power (with failure of #1 EDG to start). Malfunctions MEE032 and MDG002F shall be inserted at the direction of the Lead Examiner.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____	Scenario No.: 1	Event No.: 8	Page 17 of 19

Event Description: 1000 GPM leak inside the drywell. Malfunction MNB009F will be initiated as directed by the lead examiner

Time	Position	Applicant's Actions or Behavior
	BOP	1. Observe containment parameters and identify that a leak inside containment is in progress. *Note: The leak will start small and gradually increase to 1000 GPM. Initially RCIC and CRD will be able to maintain level, but level decrease will be apparent when leakage exceeds approximately 600 GPM* / ~~800~~ gpm with CRD ?
	SRO	1. Direct actions per EOP-01-RVCP. Execute RC/L and RC/P concurrently
		2. Once it is apparent that Reactor Water cannot be maintained above TAF the SRO will direct the RO to initiate a cooldown to the point that Core Spray will makeup to the Reactor. He will also direct that injection from LPCI be prevented
		3. When drywell pressure or suppression pool temperature exceeds entry conditions, he will enter 0EOP-02-PCCP
	RO	1. Maximize RPV injection with available high pressure sources (RCIC and CRD) 600 ~~+ 100 + 100~~
		2. Place Torus Sprays in service. / when (torus #) HS
		3. Alternate SRVs to maintain/reduce RPV pressure (cooldown < 100 F/hr) — give order
		4. Recognize/advise SRO that available high pressure sources will not be adequate to maintain RPV level above TAF ? ~~may be able~~ ok
		5. Prevent RPV injection from LPCI. — why — must use for Adequate Core Cooling then bec of
		6. Ensure Core Spray is lined up for injection
	C.T.	7. Depressurize RPV until Core Spray is injecting
		The scenario will be terminated once Core Spray flow has been established to the RPV.

1700 gpm

113

Need additional work for Scenario # 2.

1. Turnover Notes for Scenario
2. Reactivity Plan for Scenario
3. What ever else is necessary.

Facility: Brunswick Scenario No.: 2 Op-Test No.: ________

Title:Steam leak outside containment with ATWS

Examiners: ____________________ Operators: ____________________

____________________ ____________________

____________________ ____________________

Initial Conditions: The crew assumes the shift with the plant at 100% power, Mid Life. The "A" Condensate Booster Pump is OOC.. HPCI is "available" but OOC due to I&C maintenence being conducted for the past two hours.

Turnover: Night orders include direction to shift RWCU pumps.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N(RO)	Shift RWCU Pumps
2	RD005M	C(RO)	Rod 18-19 Drift Out
3	MRW010F	C(BOPand RO*) *Assume RO stays at RX Pnl	Inadvertent HPCI initiation.
4	N/A	R (RO)	Reduce power in preparation for removing "A" RFP from service
5	MCF010F and MCF036F	C (BOP)	"B" Condensate Booster Pump Sheared Shaft and "B" RFP Trip (Low Suction Pressure).
6	MCF027F (increasing)	C(BOP) C(RO*) *RCIC trips upon initiation	Manual shutdown/Trip of "A" RFP, Steam Leak from RCIC Manual Scram (RO)
7	MRP009F	M(ALL)	ATWS

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

Op-Test No.: _____ Scenario No.: 2 Event No.: 1

Event Description: Shift RWCU pumps.

Time	Position	Applicant's Actions or Behavior
	SRO	Direct the RO to shift RWCU pumps per 2OP-14, Section 8.11.
		.
	RO	1. Obtains copy of 2OP-14 and Reviews Precautions and Limitations.
		2. Dispatches an AO to Ensure at least one F/D A(B) Flow Controller is in MAN (2OP-14, Step 8.11.2.1).
		3. Request the AO to decrease each manually controlled F/D approximately 10 GPM. (2OP-14, Step 8.11.2.2) -Observe this occurs on FI-R605A/B
		4. Start the non operating RWCU Pump
		5. Stop the running RWCU pump
		6. Request the AO to return F/Ds to desired flow rate (70 to 170 GPM on each F/D) and control mode
		7. Monitor normal operation per Section 6.0 of 2OP-14 -Observe system flow 140-214 GPM on FI-R605A/B - System inlet pressure equal to reactor pressure - F/D inlet temperature <130 F

Op-Test No.: ____	Scenario No.: 2	Event No.: 1	Page 4 of 18

Event Description: Shift RWCU pumps.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____	Scenario No.: 2	Event No.: 2	

Event Description: Control Rod 18-19 drifts out.
Note: Once rod is driven to 00, malfunction RD005M will be removed and rod will remain at 00. If asked to disarm 18-19 hydraulically, acknowledge the request.

Time	Position	Applicant's Actions or Behavior
	SRO	1. Direct the RO to complete actions of APP
		2.Direct entry into AOP-03.0
		3. Direct rod 18-19 be inserted to position 00
		4. Refer to TS 3.1.3 for Control Rod Operability
	RO	1. Determine rod drift on 18-19 drifting out.
		2. Attempt arrest rod using RMCS.
		3. Enter and announce AOP-03.0
		4. Insert rod 18-19 to position 00

Op-Test No.: ____ Scenario No.: 2	Event No.: 2	Page 6 of 18

Event Description: Control Rod 18-19 drifts out.
Note: Once rod is driven to 00, malfunction RD005M will be removed and rod will remain at 00. If asked to disarm 18-19 hydraulically, acknowledge the request.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 2 Event No.: 3

Event Description: Inadvertant HPCI initiation.

Note: Malfunction MRW010F will be inserted at direction of lead examiner

Time	Position	Applicant's Actions or Behavior
	SRO	1. Direct entry into AOP-03.0 2. Direct termination of HPCI operation.
		3. Direct reduction of reactor power as necessary to prevent Scram
		4. Contact I&C to determine cause of inadvertent start Note: SRO will be told by I&C that the instrument they were working on shorted out and that HPCI will not work manually or automatically.
		5. Refer to OI-01.07 and determine reportability requirements. Note: No reporting will be actually done as part of scenario
	RO	1. Diagnose and report inadvertant HPCI initiation
		2. Enter and announce AOP-03.0
		3. Reduce reactor power as required to prevent a reactor scram and/or to maintain reactor power< 100%
	BOP	1. Verify by two independent methods that the initiation is not valid and terminate HPCI operation. Note: Once HPCI is shutdown, initiate malfunction MES016F to incapacitate HPCI for further use.

Op-Test No.: _____ Scenario No.: 2 Event No.: 3	Page 8 of 18
Event Description: Inadvertant HPCI initiation.	
Note: Malfunction MRW010F will be inserted at direction of lead examiner	

Time	Position	Applicant's Actions or Behavior

Event Description: Oil leak on "A" RFP.
Execute malfunction MCF027F per Lead Examiner direction.

Note: It is assumed that the first indication in the simulator of an oil leak will be "RFP A Turb Oil Tank Level HI-LO" alarm. If no action is taken, bearing oil drain temperatures are expected to increase followed by RFP Trip on Low Lube Oil Pressure.

Time	Position	Applicant's Actions or Behavior
	BOP	1. Observe alarm "RFP A Turb Oil Tank Level HI-LO" alarm and advise SRO.
		2. Review Alarm Response Procedure UA-04 4-2.
	SRO	1. Direct BOP to complete ARP actions
		2. Dispatch an AO to locally monitor RFP "A" lube oil reservior level.
		3. Develop MR to get oil leak repaired.
		4. Consider actions if RFP must be shutdown/discuss with OPS MGMT
	AO (Simulator Operator)	1. Advise SRO that there is a small oil leak and that reservior level is approximately 3" below normal)
	BOP/SRO	1. Direct the AO to transfer oil from the Lube Oil and Conditioning system to the "A" RFP reservior per OP-49.
		2. Contact Chemistry to initiate oil spill control measures (optional).
		3. If/when bearing oil drain temperature increases to 225 F, Trip RFP "A" and refer to AOP-23.0

Op-Test No.: _____ Scenario No.: 2 Event No.: 4

Event Description: Oil leak on "A" RFP.
Execute malfunction MCF027F per Lead Examiner direction.

Note: It is assumed that the first indication in the simulator of an oil leak will be "RFP A Turb Oil Tank Level HI-LO" alarm. If no action is taken, bearing oil drain temperatures are expected to increase followed by RFP Trip on Low Lube Oil Pressure.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 2 Event No.: 4	

Event Description: Oil leak on "A" RFP.
Execute malfunction MCF027F per Lead Examiner direction.

Note: It is assumed that the first indication in the simulator of an oil leak will be "RFP A Turb Oil Tank Level HI-LO" alarm. If no action is taken, bearing oil drain temperatures are expected to increase followed by RFP Trip on Low Lube Oil Pressure.

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 2 Event No.: 5

Event Description: Reduce Power in preparation for removing "A" RFP from service

Increase malfunction MCF027F at direction from Lead Examiner

Note: The AO will report that the oil leak on the "A" RFP is getting worse and that he doesn't believe the reservior level can be maintained for more than about 20 minutes. Oil has entered the floor drain.

Time	Position	Applicant's Actions or Behavior
	SRO	1 Obtain feedback from AO on extent of oil leak (increasing)
		2. Direct the RO to start reducing power to 60% per 0GP-12
		3. Direct the BOP to start actions to shutdown "A" RFP and to review AOP-23 "Condensate/Feedwater System Failure"
		4. Notify Dispatcher and OPS Management
		5. Notify Chemistry of oil spill into floor drains (optional)
	RO	1. Obtain copy of 0GP-12 and review Precautions and Limitations
		2. Reduce power to 60% using OGP-12. *Note: Following reduction to 90%, the next event will be started.* - Make small changes (2 to 4%) to keep recirc loop flows within 10% - Observe recirc pump speed, loop flow, and reactor power all decrease - Ensure not in unstable region of power/flow map
	BOP	Start shifting control to "B" RFP in anticipation of potential "A" RFP trip using 2OP-32. BOP will complete as much of this as is possible with the existing power level. - Adjust level setpoint on MSTR RFPT SP/RX LVL CTL to 190" - Place FW-FV-V46 to OPEN - DEPRESS A/M pushbutton and check A/M indicator in "M" (manual) - Return level setpoint to 187" - Lower RFP A MSC to low speed stop and trip turbine - Ensure "A RFPT on turning gear

Event Description: "B" Condensate Booster Pump Sheared Shaft, RFP "B" trip (low suction pressure) RCIC trip on initial start plus RCIC Steam Leak after re-start. Enter malfunction MCF010F when directed by Lead Examiner. Malfunction MCF 036F is initiated approximately 15 seconds later to trip the "B" RFP.Malfunction MES027F will be initiated on first start and then removed for second start. Malfunction MES025F will be initiated when RO re-starts RCIC. Malfunction MRP009F will be inserted in case a manual scram (or automatic scram) is initiated. Once the "B" RFP is tripped it will not be available for the remainder of the scenario.

Notes:
1. Since "A" Condensate Booster Pump is OOC it is expected that the RO will need to rapidly reduce reactor power (per AOP-23) to avoid a SCRAM when the "B Condensate Booster Pump and "B" RFP fail. The "A" RFP may automatically trip on low lube oil pressure at some point in the event.
2. The RCIC steam leak will be initiated after the restart (following reset from trip). This leak is intended to be small enough to NOT cause immediate automatic RCIC isolation nor to exceed 4 MR in exhaust radiation.
3. If/when a SCRAM occurs, this event will be performed coincident with Event #7, ATWS

Time	Position	Applicant's Actions or Behavior
	BOP	1. Recognize and announce failure of "B" Condensate Booster Pump and (15 seconds later) the "B" RFP.
		2. Adjust feed pump "A" flow as necessary to restore normal RPV level.
		3. Perform actions in AOP-23 - IF level increases to RFP trip setpoint, then TRIP "A" RFPT - Maintain hotwell level +7" and -7" - CLOSE any open feed system recirc valves
	SRO	1. Direct RO to complete actions in AOP-23 (rapidly reduce recirc flow to 45%) Notes: 1. Auto runback may occur before operator action 2. Considering the oil leak on the "A" RFP, the SRO may decide to manually SCRAM the reactor and to manually Trip the "A" RFP at this point.
		2. Direct BOP to monitor condensate/feedwater parameters
		3. Call dispatcher and advise of power reduction/possible scram
		4. Conduct shift brief to discuss ramifications of losing the "A" RFP

Op-Test No.: _____	Scenario No.: 2	Event No.: 6	

Event Description: "B" Condensate Booster Pump Sheared Shaft, RFP "B" trip (low suction pressure) RCIC trip on initial start plus RCIC Steam Leak after re-start. Enter malfunction MCF010F when directed by Lead Examiner. Malfunction MCF 036F is initiated approximately 15 seconds later to trip the "B" RFP.Malfunction MES027F will be initiated on first start and then removed for second start. Malfunction MES025F will be initiated when RO re-starts RCIC. Malfunction MRP009F will be inserted in case a manual scram (or automatic scram) is initiated. Once the "B" RFP is tripped it will not be available for the remainder of the scenario.

Notes:
1. Since "A" Condensate Booster Pump is OOC it is expected that the RO will need to rapidly reduce reactor power (per AOP-23) to avoid a SCRAM when the "B Condensate Booster Pump and "B" RFP fail. The "A" RFP may automatically trip on low lube oil pressure at some point in the event.
2. The RCIC steam leak will be initiated after the restart (following reset from trip). This leak is intended to be small enough to NOT cause immediate automatic RCIC isolation nor to exceed 4 MR in exhaust radiation.
3. If/when a SCRAM occurs, this event will be performed coincident with Event #7, ATWS

Time	Position	Applicant's Actions or Behavior
	RO	1. Rapidly reduce recirc flow to SPEED DEMAND of 45% to avoid reactor scram. Note: Power will need to be reduced to approximately 60%.
		2. Monitor power/flow and ensure PBDS is operable if in Monitored Region (Refer to 2AOP-04.0)
	*SRO may direct the BOP to Trip the "A" RFP	3. If/when a Reactor SCRAM occurs (manual or automatic), complete actions in EOP-01-RSP. Must recognize that a reactor power is >5% and that a significant number of rods (114) have not inserted **(Critical Task)** *Note: Event 6 will be conducted concurrently with Event 5 if/when the SCRAM occurs*
	BOP	1.Continue to monitor "A" RFP parameters and respond to subsequent alarms as additional lube oil system failures occur. NOTE: The RFP should be manually tripped before total lube oil failure occurs. Per APP UA-04 4-2 the pump must be tripped before bearing drain temperatures exceed 225 F.
		2. Trip the "A" RFP when directed by SRO
		3. Trip the Main Turbine as part of Reactor Scram Procedure

Op-Test No.: _____ Scenario No.: 2 Event No.: 6

Event Description: "B" Condensate Booster Pump Sheared Shaft, RFP "B" trip (low suction pressure) RCIC trip on initial start plus RCIC Steam Leak after re-start. Enter malfunction MCF010F when directed by Lead Examiner. Malfunction MCF 036F is initiated approximately 15 seconds later to trip the "B" RFP.Malfunction MES027F will be initiated on first start and then removed for second start. Malfunction MES025F will be initiated when RO re-starts RCIC. Malfunction MRP009F will be inserted in case a manual scram (or automatic scram) is initiated. Once the "B" RFP is tripped it will not be available for the remainder of the scenario.

Notes:
1. Since "A" Condensate Booster Pump is OOC it is expected that the RO will need to rapidly reduce reactor power (per AOP-23) to avoid a SCRAM when the "B Condensate Booster Pump and "B" RFP fail. The "A" RFP may automatically trip on low lube oil pressure at some point in the event.
2. The RCIC steam leak will be initiated after the restart (following reset from trip). This leak is intended to be small enough to NOT cause immediate automatic RCIC isolation nor to exceed 4 MR in exhaust radiation.
3. If/when a SCRAM occurs, this event will be performed coincident with Event #7, ATWS

Time	Position	Applicant's Actions or Behavior
	RO	1.Once "A" RFP has tripped, manually start RCIC (using the hard card) to Maintain RPV water level as directed by SRO. Note: Malfunction MES027F (turbine trip) on initial start.
		2. Respond to RCIC trip - recognize and announce trip - run trip throttle valve closed and hold closed to reset overspeed trip -restart RCIC (MES027F removed) Note:MES025F will be inserted (Steam Leak in RHR Room from RCIC) as soon as RCIC is re-started.
		3. Respond to RCIC Steam leak per APP A-02 5-7 - Enter 0EOP-03-SCCP when directed by SRO - Confirm which leak area temperature is high (Monitors on P614) - Before area exceeds Max Safe Temperature Manually Scram Note: EOPs will take precedence over AOP, and RCIC should not be isolated to stop the leak (since it is required for RPV makeup)
	SRO	Enter EOP-03-SCCP "Secondary Containment Control" and direct actions Execute SC/T, SC/R and SC/L concurrently

Event Description: "B" Condensate Booster Pump Sheared Shaft, RFP "B" trip (low suction pressure) RCIC trip on initial start plus RCIC Steam Leak after re-start. Enter malfunction MCF010F when directed by Lead Examiner. Malfunction MCF 036F is initiated approximately 15 seconds later to trip the "B" RFP.Malfunction MES027F will be initiated on first start and then removed for second start. Malfunction MES025F will be initiated when RO re-starts RCIC. Malfunction MRP009F will be inserted in case a manual scram (or automatic scram) is initiated. Once the "B" RFP is tripped it will not be available for the remainder of the scenario.

Notes:
1. Since "A" Condensate Booster Pump is OOC it is expected that the RO will need to rapidly reduce reactor power (per AOP-23) to avoid a SCRAM when the "B Condensate Booster Pump and "B" RFP fail. The "A" RFP may automatically trip on low lube oil pressure at some point in the event.
2. The RCIC steam leak will be initiated after the restart (following reset from trip). This leak is intended to be small enough to NOT cause immediate automatic RCIC isolation nor to exceed 4 MR in exhaust radiation.
3. If/when a SCRAM occurs, this event will be performed coincident with Event #7, ATWS

Time	Position	Applicant's Actions or Behavior
	BOP	Perform actions in EOP-03-SCCP - Monitor and Control Reactor Building Temperature, Level and Radiation - Align Service Water to Vital header and start RHR room coolers - Start Reactor Building HVAC fans - Advise SRO if/when any area exceeds Max Safe Temperature

Op-Test No.: ____ Scenario No.: 2 Event No.: 6	Page 14 of 18

Event Description: "B" Condensate Booster Pump Sheared Shaft, RFP "B" trip (low suction pressure) RCIC trip on initial start plus RCIC Steam Leak after re-start. Enter malfunction MCF010F when directed by Lead Examiner. Malfunction MCF 036F is initiated approximately 15 seconds later to trip the "B" RFP.Malfunction MES027F will be initiated on first start and then removed for second start. Malfunction MES025F will be initiated when RO re-starts RCIC. Malfunction MRP009F will be inserted in case a manual scram (or automatic scram) is initiated. Once the "B" RFP is tripped it will not be available for the remainder of the scenario.

Notes:
1. Since "A" Condensate Booster Pump is OOC it is expected that the RO will need to rapidly reduce reactor power (per AOP-23) to avoid a SCRAM when the "B Condensate Booster Pump and "B" RFP fail. The "A" RFP may automatically trip on low lube oil pressure at some point in the event.
2. The RCIC steam leak will be initiated after the restart (following reset from trip). This leak is intended to be small enough to NOT cause immediate automatic RCIC isolation nor to exceed 4 MR in exhaust radiation.
3. If/when a SCRAM occurs, this event will be performed coincident with Event #7, ATWS

Time	Position	Applicant's Actions or Behavior

Op-Test No.: _____ Scenario No.: 1 Event No.: 7

Event Description: ATWS Initiate malfunction MRP009F at request of lead examiner or upon Scram.

Note: The intent of the 114 rods failure is to result in power level >5% but less than 20% after scram

Time	Position	Applicant's Actions or Behavior
	RO	If/when a Reactor SCRAM occurs, complete actions in EOP-01-RSP. Must recognize that power level is above 5% and that a significant number of rods (114) did not insert and announce to SRO **(Critical Task)** -Insert manual scram, place mode switch in Shutdown - Verify no SRVs are cycling - Start RCIC - Enter EOP-01-RVCP at direction from SRO - Insert IRMs to bring on scale
	SRO	1. Once the SCRAM occurs must recognize that the reactor is above 5% and enter EOP-01-LPC. **(Critical Task)** Note: It is possible the SRO will determine the reactor can not be shutdown before suppression pool temperature exceeds 110 F and elect to initiate SLC at the onset (RC/Q-10).
		2. Direct RO to trip Recirc Pumps (RC/Q-7)
		3. Direct the RO and BOP to intentionally lower reactor water level while leaving RCIC, SLC and CRD on. (RC/L-17)**(Critical Task)** Note: Per Table 1 of LPC if RCIC isolates on steam leak, the isolation will be bypassed and RCIC restarted.
		3. Direct BOP to stabilize reactor pressure at 1050 psig using Bypass Valves (RC/P-20)
		4. IF/when more than one area in Secondary Containment exceeds Max Safe Temperature, direct RO to Emergency Depressurize (SCCP-26)
	BOP	1. Maintain RPV pressure using Bypass Valves as directed by SRO

Event Description: ATWS Initiate malfunction MRP009F at request of lead examiner or upon Scram.

Note: The intent of the 114 rods failure is to result in power level >5% but less than 20% after scram

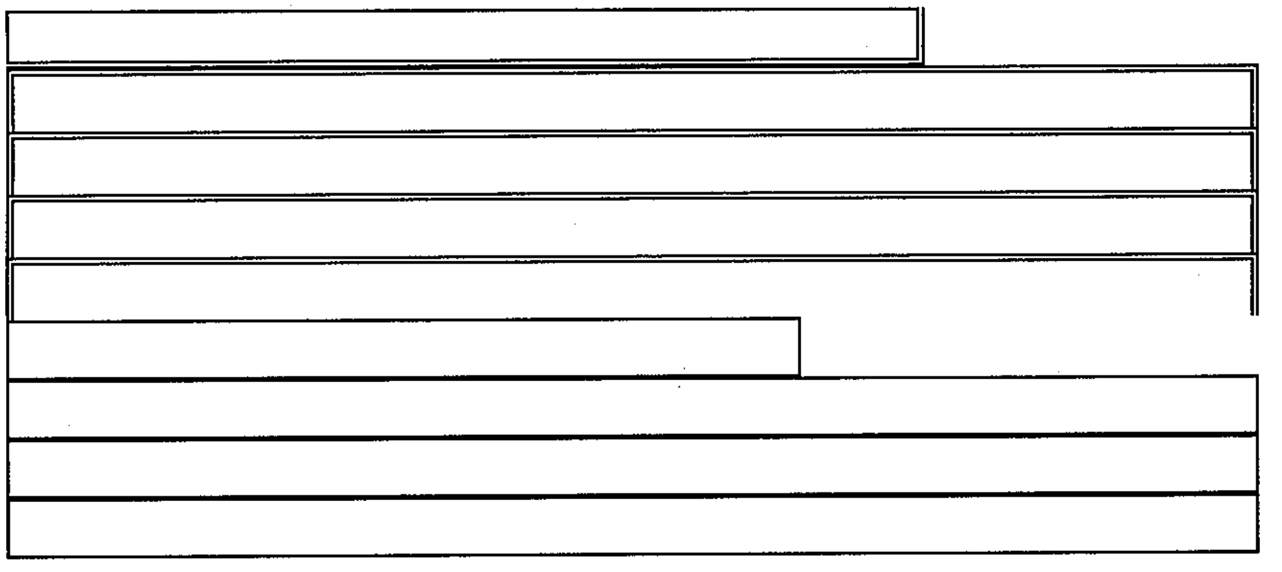
Time	Position	Applicant's Actions or Behavior
		2. Inhibit condensate/feedwater flow as necessary to lower RPV water level to +90 inches as directed by SRO. Note: It is expected that the "A" RFP will have been tripped (manually or automatically) by this time and that RCIC and CRD will be the only high pressure sources of reactor makeup.
		3. Monitor turbine lube oil system following turbine trip
	RO*	1. Initiate EOP-01-LPC RC/L, RC/P as directed by SRO.
	*Some Alternate Rod insertion may be done by BOP as directed by SRO	2. Begin Alternate Control Rod Insertion **(Critical Task)** Note: The ATWS is a hydraulic lock and rod insertion will be "allowed" in order to reduce power to <5%. Any attempt at manually inserting rods from the control room will be successful and all rods will be able to be inserted to 00.
		3. Initiate SLC at direction from SRO
		The scenario is terminated when all rods are at 00, when Emergency Depressurization is initiated or at the discretion of the lead examiner.

Op-Test No.: _____ Scenario No.: 1 Event No.: 7	Page 17 of 18

Event Description: ATWS Initiate malfunction MRP009F at request of lead examiner or upon Scram.

Note: The intent of the 114 rods failure is to result in power level >5% but less than 20% after scram

Time	Position	Applicant's Actions or Behavior



Scenario 2, event 1

6.0 SYSTEM OPERATION

Information Use

> **CAUTION**
>
> **IF** Process Computer Points B074 or B075, or NUMAC LDM *B21-XY-5949B* is **NOT** available, **THEN** the PPC heat balance may be up to 3 CMWT lower than actual reactor power because computer points only get input from RWCU filter demin flow. Raising reactor power to rated will exceed 2558 CMWT if either of the following exist:
>
> - RWCU is in service with *G31-F044* open.
> - RWCU filter demin(s) is in service with RWCU reject flow in progress.

During normal operation of the RWCU System, the operator should routinely observe the following Control Room parameters:

6.1	System flow G31-FI-R605A/B	One demineralizer Two demineralizers	70 - 107 gpm. 140 - 214 gpm.

> **NOTE:** It is desired to obtain a maximum flowrate of 107 gpm through each filter as often as possible to obtain the optimum vessel exchange rate.

6.2	System inlet pressure	Equal to reactor pressure.
6.3	F/D inlet temperature	Less than 130°F.

REFERENCES

Scenario 2, event 2

SIMULATOR SETUP

Initial Conditions

IC	13
Rx Pwr	100%
Core Age	MOC

Event Triggers

Event	Trigger Description
E1	Manually Initiated (Rod 18-19 Drift)
E2	Manually Initiated (Fuel Failure)
E3	Manually Initiated (Recirc Pump B Speed Control Fails)
E4	Manually Initiated (Gross Fuel Failure)
E5	Manually Initiated (Main Steam Line Break Turbine Building)
E6	Manually Initiated (LEP-02 Jumpers)

Malfunctions

Event	System	Tag	Title	Value (Ramp)	Activate Time	Deactivate Time
E2	RP	RP009F	ATWS #2	2	0 SEC	NA
A	NI	NI032F	APRM 4 Fails Low	NA	0 SEC	NA
A	MS	MS042F	Inboard MSIV D Fail To Close	NA	0 SEC	NA
A	MS	MS046F	Outboard MSIV D Fail To Close	NA	0 SEC	NA
E1	RD	RD005M	Control Rod 18-19 Withdraw Drift	NA	0 SEC	NA
E2	NB	NB005F	Fuel Failure	10% 10 MIN	0 SEC	NA
E4	NB	NB010F	Gross Fuel Failure	100% 0 SEC	0 SEC	NA
E5	MS	MS002F	MSL Break In Turbine Bldg	1% 60 SEC	0 SEC	NA
E6	RP	RP005F	Auto Scram Defeat	NA	0 SEC	NA

Scenario 2, event 2

EVENT 1 ROD OUT DRIFT

Instructor Activities
- ☐ When the crew has the watch, initiate trigger E1 to activate rod out drift
- ☐ If contacted as NE, report you will check thermal limits
- ☐ When rod 18-19 has been driven to 00, delete rod drift malfunction
- ☐ If asked to disarm 18-19 hydraulically, acknowledge the request

Plant Response
- ☐ Rod 18-19 drifts full out and will respond to RMCS

Operator Activities

SS
- ☐ Monitor crew activities and maintain overview.

SCO
- ☐ Conduct shift turnover shift briefing.
- ☐ Direct actions of APP for rod drift
- ☐ Direct entry into AOP-03.0
- ☐ Direct rod 18-19 inserted to position 00
- ☐ Refer to Tech Spec 3.1.3 for Control rod Operability

RO/BOP
- ☐ Determine rod drift on 18-19 drifting out
- ☐ Attempt arrest rod using RMCS
- ☐ Enter and announce AOP-03.0
- ☐ Insert rod 18-19 to position 00

STA
- ☐ As directed by the SCO/SS

WCC
- ☐ As directed by the SCO/SS

Scenario 2, event 3

EVENT 1 HPCI INITIATION

Instructor Activities
- ☐ Provide Shift Briefing sheet to the SCO
- ☐ When the crew has the watch, initiate **TRIGGER 1** to initiate the HPCI system
- ☐ If asked as I&C to investigate, acknowledge the request
- ☐ If crew response is not quick enough to prevent a scram, allow crew to perform immediate actions, then freeze discuss expected actions and backtrack to allow crew to perform again.

Plant Response
- ☐ HPCI Initiation causes power to rise. Power stabilizes at $\approx$114% if injection is allowed to continue.

Operator Activities

SS
- ☐ Monitor crew activities and maintain overview.

SCO
- ☐ Direct entry into AOP-03.0
- ☐ Direct termination of HPCI operation
- ☐ Direct reduction of reactor power as necessary to prevent a scram
- ☐ Contact I&C to investigate HPCI failure
- ☐ Refer to Tech Spec 3.5.1. Determine Actions D & E apply since RHR Pump B is under clearance
- ☐ Refer to OI-01.07 and determine reportability requirements

RO/BOP
- ☐ Diagnose and report inadvertent HPCI initiation
- ☐ Enter and announce AOP-03.0
- ☐ Verify by two independent methods the initiation is not valid and terminate HPCI operation
- ☐ Reduce reactor power as required to prevent a reactor scram and/or maintain reactor power$\leq$100%

STA
- ☐ Refer to OI-01.07 and determine reportability requirements.

WCC
- ☐ As directed by the SCO/SS

Scenario 2, event 4

PLANT OPERATING MANUAL

VOLUME IV

GENERAL PLANT OPERATING PROCEDURE

UNIT
0

0GP-12 R19

0GP-12

POWER CHANGES

REVISION 20

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides the prerequisites, precautions, limitations, and instructional guidance for performing reactor power changes by varying Reactor Recirculation System flow or manipulating control rods when reactor power level is above Reactor Recirculation pump minimum speed. This procedure also provides guidance for End-of-Cycle coast down.

This procedure is also used to verify the following Technical Specifications:

1.1 SR 3.1.3.5, The coupling integrity of control rods.

1.2 TR 7.3.7.2 (ODCMS Table 7.3.7-1, footnote c and g). The sample and analysis frequency used to determine the Dose Rate of gaseous effluents.

1.3 SR 3.3.1.1.3. APRM GAFs must be set correctly within 12 hours of reaching or exceeding 23% rated thermal power (Unit 1) or 25% rated thermal power (Unit 2).

2.0 REFERENCES

2.1 Technical Specifications 3.2.1, 3.2.3, 3.3.1.1, 3.3.1.3, 3.4.1, TR 7.3.7.2

2.2 UFSAR

2.3 0OI-01, Conduct of Operations Manual

2.4 0PLP-17, Identification, Development, Review, and Conduct of Infrequently Performed Tests or Evolutions

2.5 0GP-01, Prestartup Checklist

2.6 0GP-04, Increasing Turbine Load to Rated

2.7 0GP-05, Unit Shutdown

2.8 0GP-10, Rod Sequence Checkoff Sheets

2.9 0GP-11, Second Operator Rod Sequence Checkoff Sheets

2.10 0GP-13, Increasing Unit Capacity at End of Core Cycle

2.11 1(2)OP-02, Reactor Recirculation System Operating Procedure

2.12 1(2)OP-07, Reactor Manual Control System Operating Procedure

2.0 REFERENCES

2.13 1(2)OP-26, Turbine System Operating Procedure

2.14 1(2)OP-30, Condenser Air Removal and Off Gas Recombiner System

2.15 1(2)OP-32, Condensate and Feedwater System Operating Procedure

2.16 1(2)OP-34, Extraction Steam System Operating Procedure

2.17 1(2)OP-35, Heater Drains, Vents, and Level Control Operating Procedure

2.18 1(2)OP-36, Moisture Separator Reheater and Moisture Separator Reheater Drains System Operating Procedure

2.19 1(2)OP-59, Hydrogen Water Chemistry System Operating Procedure

2.20 1(2)PT-01.11, Core Performance Parameter Check

2.21 0PT-14.1, Control Rod Operability Check

R22 2.22 NEDO-32339, Licensing Topical Report, Reactor Stability Long-Term Solution: Enhanced Option 1-A, GE Nuclear Energy, April 1998, including Supplements 1 through 4 (ESRs 95-00080, 95-00081, and 96-00499; NRC Generic Letter 94-02, Long-Term Solution and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors.

R23 2.23 SOER 94-01, Non-conservative Decisions and Equipment Performance Problems Result in a Reactor Scram

R24 2.24 LER 1-96-02-01

R25 2.25 INPO SOER 84-2 Control Rod Mispositioning

2.26 GE SIL 614, Backup Pressure Regulator

3.0 PRECAUTIONS AND LIMITATIONS

3.1 This procedure is to be used in accordance with the procedure compliance guidelines of 0GP-01, Section 5.0.

3.2 **IF** it is desired to operate the plant below Reactor Recirculation minimum speed (approximately 22-28% in accordance with the COLR), **THEN** 0GP-04 is to be used for power increases and 0GP-05 is to be used for power decreases.

3.0 PRECAUTIONS AND LIMITATIONS

R23 3.3 Reactor recirculation pumps should be operated in accordance with the Flow Control Operation Map. Care should be taken to avoid the regions of possible core thermal hydraulic instability, as specified in the COLR.

R22 R23 3.4 Unit 2 Only: **IF** the Exclusion Region is entered, an automatic reactor scram will occur. **IF** operations are required in the Monitored Region or the Restricted Region, additional controls are required to be in place per Technical Specifications (Fraction of Core Boiling Boundary 3.2.3 and/or Period Based Detection System 3.3.1.3).

3.4.1 Entry into the Restricted Region (when **NOT** "setup") will result in a Rod Withdrawal Block.

R22

NOTE: Instability may be indicated by:

1. Unit 1 Only: *OPRM PRE-TRIP CONDITION*, A-05 5-8 alarming.

2. Unit 1 Only: *OPRM UPSCALE TRIP*, A-05 6-8 alarming.

3. Unit 2 Only: *PBDS HI DECAY RATIO DIV I* (*DIV II*), A-05 2-8(4-8) alarming and PBDS panel meters indicating higher than normal.

4. Unit 2 Only: *PBDS HI-HI DECAY RATIO DIV I (DIV II)*, A-05 3-8(5-8) alarming and PBDS panel meters indicating 11 confirmation counts.

5. An increase in baseline APRM noise level. SRMs and SRM period meters may be oscillating at the same frequency. Instability is confirmed by selecting various control rods in different quadrants and observing sustained oscillations on the LPRM meters at a peak to peak duration of less than 3 seconds;

OR

6. LPRM or APRM upscale or downscale alarms being received;

OR

7. Sustained reactor power oscillations.

3.5 Unit 2 Only: **IF** operating in the Monitored Region or the Restricted Region, **AND** a operable PBDS channel indicates a HI-HI alarm (verified by panel meter), **INSERT** a manual reactor scram.

3.0 PRECAUTIONS AND LIMITATIONS

3.6 Unit 1 Only: The OPRM system monitors the LPRMs for indication of thermal-hydraulic instability when greater than or equal to 25% thermal power **AND** less than or equal to 60% recirculation flow. This system provides alarms **AND** automatic trips as applicable. **IF** the OPRM system is inoperable **AND** operation is within Region A, **THEN** an immediate manual scram is required. **IF** the OPRM system is inoperable **AND** indications of thermal-hydraulic instability are present with operation within Region B, 5% Buffer Region, or the OPRM Enabled Region of the applicable Flow Control Operation Map, **THEN** an immediate manual scram is required.

3.7 Recirculation Pumps A and B speed changes shall be operated in accordance with 1(2)OP-02.

3.8 **WHEN** increasing reactor power, **THEN** APRM GAFs shall be periodically monitored. **IF** found greater than 1.00, **THEN** power increases should be suspended **AND** the Unit SCO should be informed.

3.9 All rod select push buttons should be deselected whenever rod movement has stabilized to minimize select switch damage from overheating.

3.10 **WHEN** Hydrogen Water Chemistry is in service, **THEN** an open feedwater or condensate minimum flow/recirculation valve downstream of the HWC hydrogen injection point at the condensate booster pump suction will decrease the hydrogen concentration in the feedwater.

3.10.1 This situation decreases hydrogen concentration in the reactor water and the effectiveness of Hydrogen Water Chemistry. Extended operation in this situation should be avoided as much as practical.

3.10.2 During plant downpowers, with HWC in service the Hydrogen/Oxygen Storage Facility will automatically shift from liquid supply (130 psig) to gas supply (180 psig) as CBP suction pressure increases above 114 psig and shifts back to liquid supply as it decreases below this value. **IF** extended operation will be required above the swap over pressure, **THEN** adequate gas supply should be ensured.

3.0 PRECAUTIONS AND LIMITATIONS

3.11 Momentarily depressing the increase or decrease pushbutton on the following controllers will cause the selected parameter to change in increments of 0.1%. Continually depressing the increase or decrease pushbutton on the following controllers will cause the selected parameter to change at an exponential rate:

3.11.1 *SULCV FW-LIC-3269,* Control Station

3.11.2 *RFPT A(B) SP CTL C32-SIC-R601A(B),* Control Stations

3.11.3 *MSTR RFPT SP/RX LVL CTL C32-SIC-R600,* Control Station

3.12 Control rod withdrawal to the Full Out position in a sequence other than that called for in 0GP-10 shall be documented on Attachment 1 (utilize additional copies, as necessary, to document rod movements).

3.13 To ensure control rods are correctly placed during reactor operation, a second licensed operator or other qualified member of the technical staff shall monitor control rod movement and shall document correct selection and placement of control rods on the procedure controlling rod movement: 0GP-04, 0GP-11, etc.

3.14 Performance of 1(2)PT-01.11, Core Performance Parameter Check, is required within 12 hours after reaching or exceeding 23% rated thermal power (Unit 1) or 25% rated thermal power (Unit 2).

3.15 **IF** a Reactor Feed Pump is removed from service during End-Of-Cycle coast down, **THEN** 0PT-37.2.1, Reactor Feed Pump Turbine Tests, is **NOT** required.

3.16 Failure to maintain RWCU at maximum flow and temperature, when operating at low power, reduces feedwater heating which may increase the thermal duty on the feedwater nozzles.

3.0 PRECAUTIONS AND LIMITATIONS

3.17 Operation at power levels between 23% RTP (Unit 1) or 25% RTP (Unit 2) and 90% RTP without a backup Main Turbine Pressure Regulator may be an unanalyzed condition. Operation at higher power levels is bounded by other transient analyses. Operation at low power levels has a large inherent margin that ensures MCPR is **NOT** exceeded. **WHEN** reactor thermal power is greater than or equal to 23% RTP (Unit 1) or 25% RTP (Unit 2) **AND** less than or equal to 90% RTP **AND** a main turbine pressure regulator is inoperable, **THEN** the inoperable pressure regulator must be restored to operable status within 4 hours. **IF** the pressure regulator can **NOT** be restored operable within 4 hours, **THEN** a power change to less than 23% RTP (Unit 1) or 25% RTP (Unit 2) **OR** greater than 90% RTP within the following four hours must be accomplished to avoid operation in an unanalyzed condition. The Nuclear Fuels (NFM&SA) Section should be notified immediately to analyze and recommend operation without a backup pressure regulator. It is important to note that **IF** reactor power is greater than 90% RTP **WHEN** a pressure regulator is found inoperable, **THEN** reactor power should **NOT** be reduced below 90% RTP. Operation in the permissible power ranges with an inoperable pressure regulator is a degraded condition which must have a time frame established for corrective actions to restore the pressure regulator to operable status. (GE SIL 614)

3.18 Unit 1 Only: Power operation of Unit 1 is limited as follows:

1. Maximum Core Power (due to cross around relief valve capacity)

   a. Less than or equal to 2825 CMWt (2nd Stage Reheat in Service).

   b. Less than or equal to 2752 CMWt (2nd Stage Reheat **NOT** in Service).

2. Main Generator Gross Output (Main Transformer Rating limit)

   a. 955 MWe (Bus 1C **AND** Bus 1D fed from UAT)

   b. 945 MWe (Bus 1C **OR** Bus 1D **NOT** fed from UAT)

4.0 PREREQUISITES

4.1 Reactor is in Mode 1 with Reactor Recirculation pumps above minimum speed.

4.2 The Load Dispatcher concurs with loading plans.

5.0 PROCEDURAL STEPS

5.1 Power Reduction

Unit___ Date/Time Started__________/________

Initials

5.1.1 All applicable prerequisites as listed in Section 4.0 are met. _____

NOTE: The following indications should be observed to verify proper response to decreased speed demand from a recirculation pump speed controller:

1. Recirculation pump speed decreases.
2. Recirculation loop flow decreases.
3. Reactor power decreases.

NOTE: Process Computer Point *B018* Total Core Flow and H12-P603 recorder *1/2B21-PDR/FR-R613* will read lower than *WTCF* as the stability region is approached. Computer Point *WTCF* is the primary indication of total core flow and should be used for stability region compliance.

R23 **NOTE:** The Shift Reactor Engineer will leave a completed copy of 0ENP-24.0, Form 2 with appropriate Power/Flow Map specified by COLR, in the Control Room for power reductions when the Reactor Engineer is **NOT** immediately available. These instructions should be designed for a rapid reduction in power and updated as control rod patterns change.

NOTE: HWC will automatically transfer to gas supply when Condensate Booster Pump suction pressure increases above 114 psig.

NOTE: The Reactor Feed Pump suction flows should be maintained approximately the same during the power reduction.

CAUTION

WHEN increasing or decreasing reactor recirculation pump speed with the Recirculation Pump A(B) speed controllers manual control knob, **THEN** small changes of 2% to 4% should be made keeping the pump speeds within 20% when below 75% core flow or within 10% when above 75% core flow (75% core flow equals 58 x 10^6 lb/hr). Normally, recirculation loop flows should be approximately the same.

5.0 PROCEDURAL STEPS

Initials

5.1.2 **IF** final feedwater temperature reduction and pressure set adjustment has been implemented, **THEN ENSURE** the plant configuration supports the power reduction in accordance with 0GP-13. _____

5.1.3 **IF** Condensate Booster Pump suction pressure approaches 114 psig during power reduction **AND** it is desired to maintain HWC supply on liquid, **THEN CONTROL** HWC on liquid supply in accordance with 1(2)OP-59. _____

CAUTION

Reactor recirculation pumps should be operated in accordance with the Flow Control Operation Map. Care should be taken to avoid the regions of possible core thermal hydraulic instability, as specified in the COLR.

Unit 1 Only: The OPRM system monitors the LPRMs for indication of thermal-hydraulic instability when greater than or equal to 25% thermal power **AND** less than or equal to 60% recirculation flow. This system provides alarms and automatic trips as applicable. **IF** the OPRM system is inoperable **AND** operation is within Region A, **THEN** an immediate manual scram is required. **IF** the OPRM system is inoperable **AND** indications of thermal-hydraulic instability are present with operation within Region B, 5% Buffer Region, or the OPRM Enabled R egion of the applicable Flow Control Operation Map, **THEN** an immediate manual scram is required.

Unit 2 Only: **IF** the Exclusion Region is entered, an automatic reactor scram will occur. **IF** operations are required in the Monitored Region or the Restricted Region, additional controls are required to be in place per Technical Specifications (Fraction of Core Boiling Boundary 3.2.3 and/or Period Based Detection System 3.3.1.3).

R25 5.1.4 **PERFORM** reactor power decreases, as directed by the Unit SCO, in accordance with the Reactor Engineer's recommendation by decreasing recirculation flow and inserting control rods in the sequence designated by 0GP-10, Rod Sequence Checkoff Sheets or Attachment 1. _____

5.0 PROCEDURAL STEPS

Initials

NOTE: RSHLV-1 and RSHLV-2 positions are indicated on the 480V MCC-TH and the 480V MCC-TL, respectively.

CAUTION

R24 **IF** Main Turbine journal bearing vibration levels begin trending upward during load reductions, **THEN** load should be either held constant or increased slightly until vibration levels either stabilize or decrease before resuming the load reduction.

5.1.5 **WHEN** generator load decreases below 520 MWe, **THEN CONFIRM** the following reheat steam high load valves go closed.

1. *RSHLV-1* _____

2. *RSHLV-2* _____

NOTE: **IF** steam pressure is decreased in the second stage tube bundles in compliance with 1(2)OP-36, Figure 1, **THEN** the cooldown rate (125°F/hr) will **NOT** be exceeded.

CAUTION

Exceeding a cooldown rate of greater than 125°F/hr on moisture separator reheater tube bundles could cause extensive tube damage.

CAUTION

A differential temperature of greater than 50°F between the outlet of the east and west moisture separator reheaters could cause damage to the low pressure turbine blades.

5.1.6 **ADJUST** Low Load Valve Panel Loaders at IR-TB-13 and IR-TB-14, as main turbine load decreases to less than 520 MWe, to decrease second stage tube bundle pressure at 15 minute intervals in accordance with 1(2)OP-36, Figure 1. _____

5.0 PROCEDURAL STEPS

Initials

NOTE: The Scram Reduction Task Force has recommended one RFPT be idled with one RFPT in service. This will reduce the time required for injections if the on-line RFPT should malfunction.

NOTE: **IF** condenser waterbox is isolated, **THEN** it is preferred to remove the RFPT which exhausts into that condenser.

5.1.7 **WHEN** reactor power is approximately 53% (Unit 1) or 60% (Unit 2), **THEN REMOVE** one Reactor Feed Pump from service **OR IDLE** a Reactor Feed Pump in accordance with 1(2)OP-32. _____

5.1.8 **WHEN** turbine load reaches approximately 440 MWe, **THEN STOP** one of the Heater Drain Pumps. _____

5.1.9 **CONFIRM** the following associated discharge level control valve closes:

1. *HEATER DRAIN PUMP A DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-1.* _____
2. *HEATER DRAIN PUMP B DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-2.* _____
3. *HEATER DRAIN PUMP C DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-3.* _____

5.1.10 **CHECK** the remaining heater drain discharge level control valve stays throttled to maintain deaerator level between 45 and 59 inches. _____

CAUTION

Condenser vacuum should be monitored closely.

5.1.11 **IF** necessary, **THEN THROTTLE OPEN** *DEAERATOR FILL AND DRAIN VALVE, HD-V57*, to control deaerator level between 45 and 59 inches as power is decreased. _____

5.0 PROCEDURAL STEPS

Initials

NOTE: The following steps are performed in accordance with recommendations from GE associated with minimizing release of corrosion product activity. The final Heater Drain Pump in operation will be secured at a turbine load of 360 MWe at the discretion of the Unit SCO, but in all cases by 200 MWe.

5.1.12 **IF** desired, **THEN PERFORM** the following at approximately 360 MWe:

1. **STOP** the remaining Heater Drain Pump. _____

2. **CONFIRM** the following associated discharge level control valve closes:

   a. *HEATER DRAIN PUMP A DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-1* _____

   b. *HEATER DRAIN PUMP B DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-2* _____

   c. *HEATER DRAIN PUMP C DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-3* _____

CAUTION
Heater Drain Tank level should be monitored closely during operation of the Main Turbine with no Heater Drain Pump in service.

3. **THROTTLE** *DEAERATOR FILL & DRAIN VLV, HD-V57*, as necessary to Maintain deaerator level between 48 and 57 inches. _____

5.1.13 **WHEN** reactor power decreases to approximately 30% (Unit 1) or 35% (Unit 2) **OR** condensate booster pump discharge pressure increases to greater than 400 psig, **THEN STOP** one of the condensate booster pumps per 1(2)OP-32. _____

5.0 PROCEDURAL STEPS

Initials

CAUTION

A reactor power reduction below 26% RTP (Unit 1) or 30% RTP (Unit 2) with core flow greater than 50% rated will result in an additional penalty (specified in COLR) to the thermal limit calculations for CPR and APLHGR.

5.1.14 **IF** reactor power is to be reduced below 26% RTP (Unit 1) or 30% RTP (Unit 2), with core flow greater than 50% rated, **THEN CONTACT** the Reactor Engineer. _____

5.1.15 **WHEN** reactor power approaches 25% (Unit 1) or 30% (Unit 2) **OR** condensate pump discharge pressure approaches 190 psig, **THEN THROTTLE OPEN** as necessary the SJAE Condensate Recirculation Valve, *CO-FV-49*, to maintain condensate pump discharge header pressure between 150 psig and 190 psig. _____

5.1.16 **IF** recommended by the Reactor Engineer, **THEN PERFORM** a rod sequence exchange. _____

5.1.17 As directed by the Unit SCO, **INSERT OR WITHDRAW** rods per the Reactor Engineer's recommendation to correct insert and withdrawal errors displayed by RWM. _____

NOTE: The following step must be performed by a licensed person and independently verified by an SRO. All control rods must be removed from the SRI bus prior to reducing rated thermal power below 30%.

5.1.18 Unit 2 Only: Following Reactor Engineer's instructions, **CHANGE OR REMOVE** rods on the SRI bus. _____

5.0 PROCEDURAL STEPS

Initials

CAUTION

The following annunciator should alarm at a decreasing power of approximately: Unit 1 Only @ 26%: *TURB CV FAST CLOS/SV TRIP BYPASS* (A-05, 6-7), Unit 2 Only @ 30%: *TURB CV FAST CLD/SV/RPT TRIP BYPASS* (A-05, 6-7). **IF** the annunciator is **NOT** in alarm as power decreases below approximately 26% (Unit 1) or 30% (Unit 2), **THEN** a turbine trip would result in a reactor SCRAM.

5.1.19 **IF NOT** performed at 360 MWe, **THEN STOP** the remaining heater drain pump at 200 MWe. _____

5.1.20 **CONFIRM** the associated operating discharge level control valve closes:

1. *HEATER DRAIN PUMP A DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-1.* _____

2. *HEATER DRAIN PUMP B DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-2.* _____

3. *HEATER DRAIN PUMP C DISCHARGE DEAERATOR LEVEL CONTROL VALVE, HD-LV-91-3.* _____

5.1.21 **THROTTLE** *DEAERATOR FILL AND DRAIN VALVE, HD-V57*, as necessary to maintain deaerator level between 48 and 57 inches. _____

5.0 PROCEDURAL STEPS <u>Initials</u>

NOTE: The following indications should be observed to verify proper response to decreased speed demand from a recirculation pump speed controller:

1. Recirculation pump speed decreases.
2. Recirculation loop flow decreases.
3. Reactor power decreases.

CAUTION

WHEN increasing or decreasing reactor recirculation pump speed with the Recirculation Pump A (B) speed controllers manual control knob, **THEN** small changes of 2% to 4% should be made keeping the pump speeds within 20% when below 75% core flow (75% core flow equals 58 x 10^6 lb/hr). Normally, recirculation loop flows should be approximately the same.

5.1.22 **REDUCE** recirculation pump speeds to the low speed limit. _____

NOTE: The Digital Feedwater Level Control System will automatically shift to 1-ELEM control if total feedwater flow is less than 2.0 X 10^6 lbm/hr.

5.1.23 **IF** required to stabilize feedwater flow (RFPT operation), **THEN PERFORM** the following:

1. **ENSURE** *FW-FV-177 ISOL VLV, FW-V10,* is open. _____
2. **OPEN** *FEDWATER RECIRC TO CONDENSER VLV, FW-FV-177*, to bypass approximately 1 x 10^6 lbm/hr to the hotwell. _____

5.1.24 **CONFIRM** core thermal limits are within the prescribed limits of Technical Specifications. _____

5.0 PROCEDURAL STEPS

Initials

5.1.25 **IF** reactor power is decreased to less than 25%, **THEN PERFORM** 0PT-13.3, Reactor Recirculation Jet Pump Operability Below 25% Rated Thermal Power. _____

5.1.26 Unit 1 Only: **IF** reactor power is decreased to less than 25%, **THEN CONFIRM** PSS *ACTIVE* light is off at Power System Stabilizer control cabinet, 1-GEN-PSS. _____

Date/Time Completed ____________________

Performed By (Print)	Initials
____________________	_____
____________________	_____
____________________	_____
____________________	_____

Reviewed By: ____________________
Unit SCO

 Initials

5.2 Power Increases Unit___ Date/Time Started__________/________

5.2.1 All applicable prerequisites as listed in Section 4.0 are met. _____

NOTE: The following indications should be observed to verify proper response to increased speed demand from a recirculation pump speed controller:

1. Recirculation pump speed increases.
2. Recirculation loop flow increases.
3. Reactor power increases.

NOTE: Turbine load should be increased in accordance with 1(2)OP-26, Figure 3.

NOTE: Procedural steps directing power increases may be performed concurrently with other steps of this procedure.

NOTE: **IF** thermal power is increased more than 15% in one hour, **THEN** reactor coolant shall be sampled in accordance with TR 7.3.7.2 (ODCM Table 7.3.7-1, footnote c).

NOTE: .Process Computer Point B018 total core flow and H12-P603 recorder *1/2B21-PDR/FR-R613* will read lower than Process Computer Point *WTCF* as the stability region is approached. Computer Point *WTCF* is the primary indication of total core flow and should be used for stability region compliance.

5.0 PROCEDURAL STEPS

Initials

CAUTION
Reactor recirculation pumps should be operated in accordance with the Flow Control Operation Map. Care should be taken to avoid the regions of possible core thermal hydraulic instability, as specified in the COLR. Unit 1 Only: The OPRM system monitors the LPRMs for indication of thermal-hydraulic instability when greater than or equal to 25% thermal power **AND** less than or equal to 60% recirculation flow. This system provides alarms and automatic trips as applicable. **IF** the OPRM system is inoperable **AND** operation is within Region A, **THEN** an immediate manual scram is required. **IF** the OPRM system is inoperable **AND** indications of thermal-hydraulic instability are present with operation within Region B, 5% Buffer Region, or the OPRM Enabled Region of the applicable Flow Control Operation Map, **THEN** an immediate manual scram is required. Unit 2 Only: **IF** the Exclusion Region is entered, an automatic reactor scram will occur. **IF** operations are required in the Monitored Region or the Restricted Region, additional controls are required to be in place per Technical Specifications (Fraction of Core Boiling Boundary 3.2.3 and/or Period Based Detection System 3.3.1.3).

5.2.2 **PERFORM** Attachment 2, each 10% power change increment. _____

R25 5.2.3 **PERFORM** power increases, as directed by the Unit SCO, by withdrawing control rods in accordance with 1(2)OP-07 in the sequence designated by 0GP-10, Attachment 1 and increasing recirculation flow in accordance with Reactor Engineer's recommendation. _____

5.2.4 **IF** Digital Feedwater Level Control System is in 1-ELEM control, **THEN** swap to 3-ELEM control in accordance with 1(2)OP-32. _____

5.2.5 **IF** operating using *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177*, to stabilize feedwater flow, **THEN CLOSE** *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177.* _____

1. **WHEN** *FEEDWATER RECIRC TO CONDENSER VLV, FW-FV-177*, is closed, **THEN CLOSE** *FW-FV-177 ISOL VLV, FW-V10.* _____

5.0 PROCEDURAL STEPS

Initials

5.2.6 **PERFORM** 0PT-13.1, Reactor Recirculation Jet Pump Operability, prior to exceeding 25% reactor power. _____

5.2.7 Unit 1 Only: **WHEN** reactor power is between 23% and 28%, **THEN CONFIRM** APRM GAFs are less than or equal to 1.00. _____

5.2.8 **IF** reactor power was decreased to less than 23% (Unit 1) or 25% (Unit 2), **THEN PERFORM** 1(2)PT-01.11, Core Performance Parameter Check, within 12 hours after reaching or exceeding 23% RTP (Unit 1) or 25% RTP (Unit 2). _____

> **NOTE:** Heater drains recirculation should be conducted such that the system will be ready for forward pumping of the heater drains when turbine load reaches 200 MWe.

5.2.9 **IF** secured, **THEN PLACE** heater drains in the recirculation mode in accordance with 1(2)OP-35. _____

5.2.10 Unit 2 Only: **IF** recommended by the Reactor Engineer, **AND** the additional controls of Technical Specifications 3.2.3 and 3.3.1.3 for operating in the Restricted Region have been implemented, **THEN PERFORM** APRM Normal Trip Setpoint setup as follows:

1. **CONFIRM** FCBB is less than or equal to 1.0 using Core Monitor edit program. _____

2. **DEPRESS** the *NORMAL/SETUP* push-button on the FCTR card for each APRM. _____

3. **CONFIRM** NORMAL/SETUP LED is yellow. _____

5.2.11 **IF** SJAE Condensate Recirculation Valve, *CO-FV-49*, is open, **THEN THROTTLE CLOSED** as necessary to maintain Condensate Pump discharge pressure between 150 psig and 190 psig. _____

5.2.12 **IF** Condensate Booster Pump suction pressure approaches 114 psig during power increase **AND** it is desired to maintain HWC supply on liquid, **THEN CONTROL** HWC on liquid supply in accordance with 1(2)OP-59. _____

5.0 PROCEDURAL STEPS <u>Initials</u>

5.2.13 **NOTIFY** radwaste to perform the following:

1. **PLACE** CDDs, CFDs, and Master Flow Controllers in service as required. _____

2. **PLACE** Hotwell level control in feed and bleed in accordance with 1(2)OP-32, as desired. _____

5.2.14 **ADJUST** Low Load Valve Panel Loaders at IR-TB-13 and IR-TB-14, as main turbine load increases, to increase second stage tube bundle pressure at 15 minute intervals in accordance with 1(2)OP-36, Figure 1. _____

5.2.15 **WHEN** turbine load increases to between 200 MWe and 360 MWe, **THEN COMMENCE** forward pumping of the heater drains in accordance with 1(2)OP-35. _____

5.2.16 **WHEN** turbine load reaches approximately 240 MWe, **THEN ENSURE** *HP TURB 7TH STAGE EXHAUST DRAIN VLVS MVD-MOV-CA-4/3/1/2* are closed. _____

5.2.17 <u>Unit 2 Only</u>: **WHEN** turbine load reaches approximately 240 MWe, **THEN PERFORM** the following:

1. **OBTAIN** Select Rod Insert (SRI) control rod list from the Reactor Engineer. _____

> **NOTE:** The following step must be performed by a licensed person and independently verified by a Senior Reactor Operator.

2. **PLACE** the SRI control rods on the SRI bus recommended by the Reactor Engineer in accordance with 2OP-07. _____/_____ IV SRO

3. **RESET** the APRM setdown. _____

5.0 PROCEDURAL STEPS <u>Initials</u>

> **NOTE:** The Turbine Stop Valve/Control Valve Fast Closure Reactor Scram MUST be enabled **PRIOR** to exceeding 26% RTP (Unit 1) or 30% RTP (Unit 2). This may be accomplished by annunciator and relay confirmation of automatic enabling **OR** by manually enabling this function by removing fuses.

5.2.18 **PRIOR** to 26% RTP (760 MWT on Unit 1) or 30% RTP (767 MWT on Unit 2), **CONFIRM** Turbine Stop Valve/Control Valve Fast Closure Reactor SCRAM is enabled by performing the following for the applicable Unit:

1. <u>Unit 1 Only</u>: **CONFIRM** *TURB CV FAST CLOS/SV TRIP BYPASS* (A-05, 6-7) is clear. _____

2. <u>Unit 2 Only</u>: **CONFIRM** *TURB CV FAST CLOS/SV/RPT TRIP BYPASS* (A-05, 6-7) is clear. _____

> **NOTE:** The *K9A-D* relays are deenergized when they are at the stop screws.

3. **CONFIRM** relay *C71A(72A)-K9A* on Panel H12-P609 is deenergized. _____

4. **CONFIRM** relay *C71A(72A)-K9C* on Panel H12-P609 is deenergized. _____

5. **CONFIRM** relay *C71A(72A)-K9B* on Panel H12-P611 is deenergized. _____

6. **CONFIRM** relay *C71A(72A)-K9D* on Panel H12-P611 is deenergized. _____

5.0 PROCEDURAL STEPS

Initials

NOTE: Removing the following fuses will deenergize relays *C71A(72A)-K9A-D* and enable the Reactor SCRAM on Turbine Stop Valve/Control Valve Fast Closure. Confirmation of relay deenergization SHOULD be performed after each fuse is removed. (Reference prints- Unit 1: 1-FP-55046, Sh 6-9, 1-FP-55085, Sh 1,3, 1-FP-55086, Sh 1,3; Unit 2: 2-FP-50015, Sh 6-9, 2-FP-50607 Sh 1,3, 2-FP-50608, Sh 1,3.)

5.2.19 **IF** the Turbine Stop Valve/Control Valve Fast Closure Reactor SCRAM is **NOT** enabled, **THEN MANUALLY ENABLE** this function prior to 26% RTP (760 MWT on Unit 1) or 30% RTP (767 MWT on Unit 2) by performing the following for the applicable Unit:

1. Unit 1 Only:

   a. **REMOVE** fuse *C71-F9A* from Panel H12-P609. ____/____ Ind.Ver.

   b. **REMOVE** fuse *C71-F9C* from Panel H12-P609. ____/____ Ind.Ver.

   c. **REMOVE** fuse *C71-F9B* from Panel H12-P611. ____/____ Ind.Ver.

   d. **REMOVE** fuse *C71-F9D* from Panel H12-P611. ____/____ Ind.Ver.

   e. **CONFIRM** *TURB CV FAST CLOS/SV TRIP BYPASS* (A-05, 6-7) is clear. ____

NOTE: The *K9A-D* relays are deenergized when they are at the stop screws.

   f. **CONFIRM** relay *C71A-K9A* on Panel H12-P609 is deenergized. ____

   g. **CONFIRM** relay *C71A-K9C* on Panel H12-P609 is deenergized. ____

   h. **CONFIRM** relay *C71A-K9B* on Panel H12-P611 is deenergized. ____

   i. **CONFIRM** relay *C71A-K9D* on Panel H12-P611 is deenergized. ____

5.0 PROCEDURAL STEPS

Initials

2. Unit 2 Only:

   a. **REMOVE** fuse *C72-F9A* from Panel H12-P609. ____/____ Ind.Ver.

   b. **REMOVE** fuse *C72-F9C* from Panel H12-P609. ____/____ Ind.Ver.

   c. **REMOVE** fuse *C72-F9B* from Panel H12-P611. ____/____ Ind.Ver.

   d. **REMOVE** fuse *C72-F9D* from Panel H12-P611. ____/____ Ind.Ver.

   e. **CONFIRM** *TURB CV FAST CLOS/SV/RPT TRIP BYPASS* (A-05, 6-7) is clear. ____

NOTE:	The *K9A-D* relays are deenergized when they are at the stop screws.

   f. **CONFIRM** relay *C72A-K9A* on Panel H12-P609 is deenergized. ____

   g. **CONFIRM** relay *C72A-K9C* on Panel H12-P609 is deenergized ____

   h. **CONFIRM** relay *C72A-K9B* on Panel H12-P611 is deenergized. ____

   i. **CONFIRM** relay *C72A-K9D* on Panel H12-P611 is deenergized. ____

5.0 PROCEDURAL STEPS

Initials

NOTE: Installation of the *C71(72)F9A-D* fuses should **NOT** energize the *C71A(72A)K9A-D* relays at this power level. Confirmation of relays remaining deenergized should be performed as each fuse is installed. **IF** relay(s) energize, **THEN** the Unit SCO should be contacted immediately.

5.2.20 **IF** the Turbine Stop Valve/Control Valve Fast Closure Reactor SCRAM was manually enabled, **THEN**, at approximately 35% Reactor Power **PERFORM** the following for the applicable Unit:

1. Unit 1 Only:

   a. **INSTALL** fuse *C71-F9A* in Panel H12-P609. ___/___ Ind.Ver.

   b. **INSTALL** fuse *C71-F9C* in Panel H12-P609. ___/___ Ind.Ver.

   c. **INSTALL** fuse *C71-F9B* in Panel H12-P611. ___/___ Ind.Ver.

   d. **INSTALL** fuse *C71-F9D* in Panel H12-P611. ___/___ Ind.Ver.

2. Unit 2 Only:

   a. **INSTALL** fuse *C72-F9A* in Panel H12-P609. ___/___ Ind.Ver.

   b. **INSTALL** fuse *C72-F9C* in Panel H12-P609. ___/___ Ind.Ver.

   c. **INSTALL** fuse *C72-F9B* in Panel H12-P611. ___/___ Ind.Ver.

   d. **INSTALL** fuse *C72-F9D* in Panel H12-P611. ___/___ Ind.Ver.

NOTE: The *K9A-D* relays are deenergized when they are at the stop screws.

3. **CONFIRM** relay *C71A(72A)-K9A* on Panel H12-P609 is deenergized. _____

4. **CONFIRM** relay *C71A(72A)-K9C* on Panel H12-P609 is deenergized. _____

5. **CONFIRM** relay *C71A(72A)-K9B* on Panel H12-P611 is deenergized. _____

5.0 PROCEDURAL STEPS

Initials

6. **CONFIRM** relay *C71A(72A)-K9D* on Panel H12-P611 is deenergized. ____

7. Unit 1 Only: **CONFIRM** *TURB CV FAST CLOS/SV TRIP BYPASS* (A-05, 6-7) is clear. ____

8. Unit 2 Only: **CONFIRM** *TURB CV FAST CLOS/SV/RPT TRIP BYPASS* (A-05, 6-7) is clear. ____

5.2.21 **NOTIFY** Radwaste to place additional CDDs and CFDs in service as required. ____

5.2.22 **WHEN** reactor power approaches 40% **OR** Condensate Booster Pump discharge pressure decreases to near 340 psig, **THEN ENSURE** a second Condensate Booster Pump is in service as necessary in accordance with 1(2)OP-32. ____

5.2.23 **WHEN** reactor power exceeds 40%, **THEN CONFIRM** the following:

a. Circulating Water System operation is in conformance with NPDES restrictions in accordance with 1(2)OP-29, Figure 1. ____

b. Unit 1 Only: PSS *ACTIVE* light is on at Power System Stabilizer control cabinet, 1-GEN-PSS. ____

5.2.24 Unit 2 Only: **IF** APRM Trip Setpoints were *SETUP*, **THEN CONFIRM** that reactor power/flow is outside the Restricted Region in the appropriate Power/Flow Map in the COLR, **AND PERFORM** the following:

1. **CONFIRM** the correct Power/Flow Map is referenced. ____

2. **DEPRESS** *NORMAL/SETUP* push-button on the FCTR card for each APRM **AND CONFIRM** NORMAL/SETUP LED is green. ___/___ Ind.Ver.

5.2.25 **START** additional circulating water pumps as necessary in accordance with 1(2)OP-29 to maintain condenser vacuum. ____

CAUTION

High power operation with only a single Heater Drain Pump in service may cause the following problems: hotwell level tilts, Condensate Pump runout, and differential pressure problems with radwaste CFDs and CDDs.

5.2.26 **WHEN** the Heater Drain Tank level can **NOT** be maintained with only a single Heater Drain Pump in service, **THEN THROTTLE OPEN** *DEAERATOR FILL & DRAIN VLV, HD-V57,* as needed to permit additional power increase. _____

NOTE: As long as Heater Drain Tank level can be maintained with only a single Heater Drain Pump in service, it is acceptable to increase power.

5.2.27 **IF** desired, **WHEN** turbine load reaches approximately 400 MWe, **THEN PLACE** a second Heater Drain Pump in service in accordance with 1(2)OP-35. _____

5.2.28 Unit 1 Only: **WHEN** reactor power is between 58% and 63%, **THEN CONFIRM** APRM GAFs are less than or equal to 1.00. _____

NOTE: **IF** due, **THEN** Reactor Feed Pump turbine tests 0PT-37.2.1, 0PT-37.2.2, and 0PT-37.2.3 should be performed prior to placing the second Reactor Feed Pump in service.

5.2.29 **WHEN** reactor power is approximately 53% (Unit 1) or 60% (Unit 2), **THEN PLACE** a second Reactor Feed Pump in service in accordance with 1(2)OP-32. _____

5.2.30 **WHEN** reactor power exceeds 65%, **THEN ENSURE** *REHEAT STEAM HIGH LOAD VALVES, RSHLV-1 AND RSHLV-2*, open. _____

5.2.31 Unit 1 Only: **WHEN** reactor power is between 78% and 83%, **THEN CONFIRM** APRM GAFs are less than or equal to 1.00. _____

5.0 PROCEDURAL STEPS

5.2.32 Unit 2 Only: **ENSURE** APRM setdowns are reset. _____

R25

> **NOTE:** Control rod withdrawal to the Full Out position in a sequence other than that called for in 0GP-10 shall be documented on Attachment 1.

5.2.33 Unit 1 Only: **INCREASE** reactor power as directed by the Unit SCO, in accordance with the Reactor Engineer's recommendation, to the most limiting of the values stated in Step 3.18. _____

5.2.34 Unit 2 Only: **INCREASE** reactor power to 100% as directed by the Unit SCO, in accordance with the Reactor Engineer's recommendation. _____

5.2.35 **WHEN** unit is at 100% maximum achievable reactor power, **THEN ENSURE** reactor pressure is at rated pressure of 1030 psig utilizing narrow range indications (preferably Computer Point B015 if available). _____

5.2.36 **CONFIRM** core thermal limits are within the prescribed limits of Technical Specifications. _____

Date/Time Completed ____________________

Performed By (Print)	Initials

Reviewed By: ____________________
Unit SCO

COMMENTS:

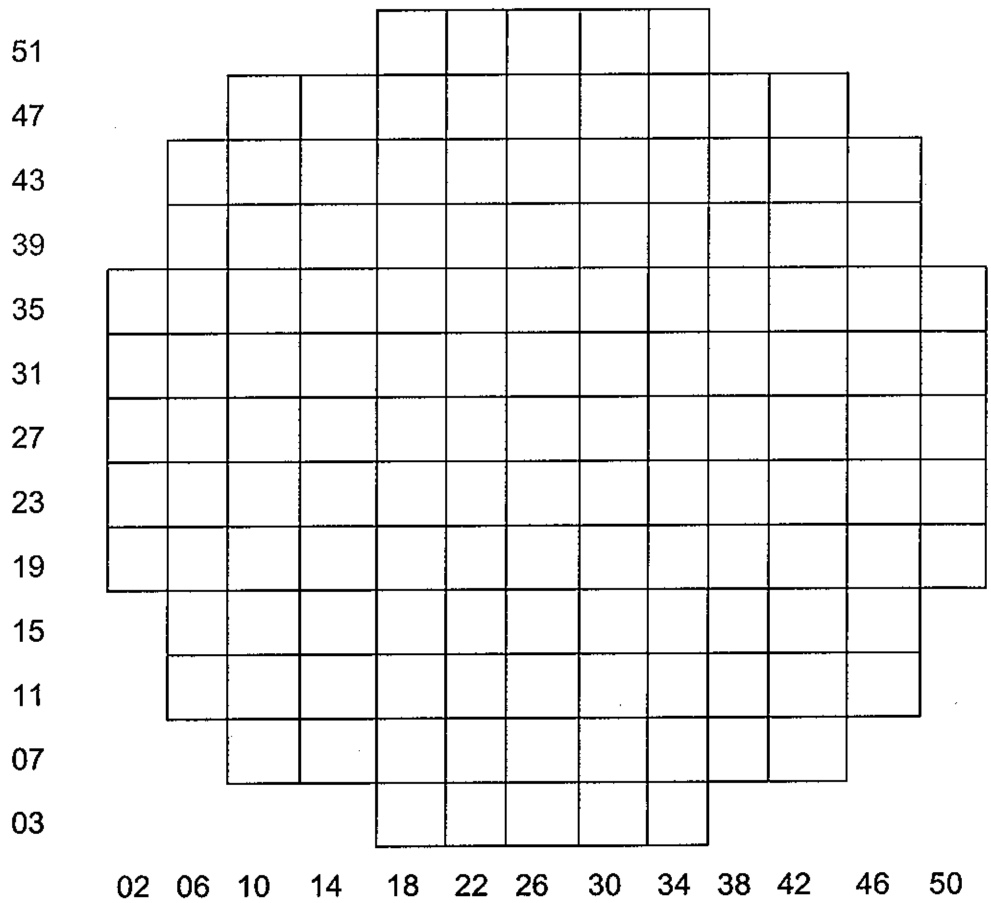
____

____

____

ATTACHMENT 1
Page 1 of 3
Control Rod Movement

Present Rod Pattern:
(or attach Display 810 edit)



Unit_____ Date___________ Time_________ Reactor Engineer________________

Control Rod Movement

Individual Rod Movement Instructions:

	Control Rod	If applicable, Complete 0PT-14.1**	Control Rod Position			Licensed Operator	Overtravel Check*	Second Licensed Operator or Qualified Monitor
1				to				
2				to				
3				to				
4				to				
5				to				
6				to				
7				to				
8				to				
9				to				
10				to				
11				to				
12				to				
13				to				
14				to				
15				to				
16				to				
17				to				
18				to				

***WHEN** a control rod is withdrawn to the Full Out position,**VERIFY** the control rod does **NOT** overtravel (Technical Specification SR 3.1.3.5).

Applicable for control rods moved from intermediate to fully withdrawn position. Technical Specification SR 3.1.3.2 must be completed for these rods if **NOT performed within the previous seven days. This surveillance requirement is **NOT** required to be performed until seven days after the control rod is withdrawn and thermal power is greater than the LPSP of RWM.

Control Rod Movement

Other Instructions ____

____

____

____

____

NOTE: **IF** control rods were withdrawn to "full out" position, **THEN** 0GP-12, Attachment 1, must be routed to Document Control for retention as a surveillance requirement.

Date/Time Completed ____________________

Performed By (Print)	Initials
______________________	______
______________________	______
______________________	______
______________________	______

Reviewed By: ______________________
Unit SCO

ATTACHMENT 2
Page 1 of 3
Verification of Reactor Power Level Using Alternate Indications

UNIT: ______________________________ DATE:______________________

> **NOTE:** This attachment is used to validate the heat balance at approximately 10% power increments.

1. **OBTAIN** valid Heat Balance (Display 820 or 0PT-01.8D, Core Thermal Power Calculation) **AND RECORD** heat balance % power in Table 1.

2. **OBTAIN** LPRM % PWR (Unit 1 - Display 861, Filtered LPRM Readings Edit; Unit 2 - Display 815, LPRM Readings Edit) **AND RECORD** in Table 1.

TABLE 1

TIME	APPROX. RX POWER	STEAM FLOW % POWER	LPRM % POWER	HEAT BALANCE % Power	APRM GAFs ≤1.00	INITIALS
N/A	TURBINE ON LINE	N/A	N/A	N/A		N/A
	30%					
	40%					
	50%					
	60%					
	70%					
	80%					
	90%					
	100%					

Definitions for Table 1:

HEAT BALANCE - A calculation of core thermal power obtained by solving an energy balance on the reactor vessel. Valid heat balance calculations may be obtained from Display 820 edit or manually by performing 0PT-01.8D, Core Thermal Power Calculation. Caution must be taken to ensure any failed sensors have valid substituted values.

LPRM % POWER - An alternate indication of reactor power calculated only on the process computer which is obtained by averaging calibrated LPRM readings.

STEAM FLOW - An alternate indication of reactor power obtained by correlating the total steam flow to a valid heat balance. Total steam flow can be obtained from process computer point B041, ERFIS points C32FA014, C32FA015, C32FA016, C32FA017, or RTGB indications C32-R603A, B, C, D on P603.

Verification of Reactor Power Level Using Alternate Indications

3. **PERFORM** the following to obtain Total Steam Flow (Mlb/hr):

Steam Line	(A)	(B)	(C)	(D)
(ERFIS)	C32FA014	C32FA015	C32FA016	C32FA017
(P603)	C32-R603A	C32-R603B	C32-R603C	C32-R603D
	______	______	______	______

Total Steam Flow = (A) + (B) + (C) + (D) = ___________

OR

USE computer point B041

4. **PERFORM** the following to log on to ERFIS at the ERFIS VT-200 terminal on the SRO's desk:

   a. **TYPE:** SET HOST EC01B (EC02B)

      OR

      SET HOST EC01A (EC02A)

   b. **TYPE:** GEPACUSER at USERNAME prompt

   c. **TYPE:** GEPAC at PASSWORD prompt

> **NOTE:** Typing MAN runs an interactive program called MAN_ALTDSP, which performs alternate power calculations based upon user supplied plant inputs. Decimal points must be entered for all values. The equivalent % power output from this program will be used for the comparison in the next step.

Verification of Reactor Power Level Using Alternate Indications

NOTE: Typing NE runs an automatic program called NE_MAIN, which reads ERFIS computer points and automatically calculates the alternate power correlations for display. There are 7 screens in the program. The user can type "A" to advance from one screen to the next or the user can enter the number of the screen (1-7) he wishes to display next. The Alternate Power Display is screen 6. The user can enter "H" for online HELP. The user must enter "E" to EXIT the program.

d. **TYPE:** MAN (for manual input and enter data at screen prompts)

OR

e. **TYPE:** NE (for automatic input) and select screen 6 (type: 6).

5. **RECORD** STEAM FLOW alternate indication (% power) in Table 1 of this attachment using the value obtained from MAN or NE programs.

6. **COMPARE** the Heat Balance (%) with the other alternate indications (%).

7. **IF** the heat balance is greater than all alternate indications (conservative as is) **OR** one or more alternate indications are within ± 5% of the heat balance (normal acceptance), **THEN** power ascension may continue.

8. **IF** power ascension is **NOT** permitted, **THEN CONTACT** Reactor Engineering to account for the differences in agreement.

9. **REPEAT** the above steps at 10% increments until the reactor is at 100% power.

REVISION SUMMARY

Revision 20 is an editorial change to correct the formatting contained in the paragraph mark at Step 5.2.20.1.a to allow the existing step to print (previously the existing step would not print when the document was printed), added clarifying information to Note at Step 5.2.20 and to convert the document to Word 2000 desktop software.

Revision 19 incorporated changes for EC 46861, Implement Extended Power Uprate, for Unit 1.

Revision 18 incorporated final turnover of EC 0000048588 (ESR 01-00209), Unit 1 Power System Stabilizer, by adding steps to confirm the PSS *ACTIVE* light is off below 25% power and on when returning above 40% power.

Revision 17 incorporated ESR 00-00417 which decomissioned the Unit 1 Automatic function of MSR 2nd Stage Reheat Pressure Transmitters, and ESR 00-00442, Unit 1 Power Range Neutron Monitoring Replacement.

Revision 16 incorporated ESR 99-00521 which decomissioned the Unit 2 Automatic function of MSR 2nd Stage Reheat Pressure Transmitters, and deleted Caution prior to Step 5.1.14 which is no longer applicable with Powerplex program.

Revision 15 added a column to Attachment 1 for completion of 0PT-14.1 for control rods repositioned from partial to fully withdrawn, as applicable for SR 3.1.3.2 (CR 20977).

Revision 14 incorporated ESR 95-01733 to reflect TS 3.2.1/3.3.1.1.3 compliance if power was reduced to less than 25% rated thermal power. (Added/revised steps to perform 0PT-01.11 and APRM GAFs within 12 hours of reaching or exceeding 25% RTP).

CONSUMERS' CHECKBOOK A NONPROFIT CONSUMER INFORMATION AND SERVICE RESOURCE

Guide to Health Plans (2003) - FEHBP Basics

General Procedures

The Federal Employees Health Benefits Program (FEHBP) is a unique government program. Instead of giving you one "take it or leave it" choice, the government licenses plans to compete for your premium dollar. It pays most of the premium cost—up to 75 percent for annuitants and most employees, and even more for Postal employees—for whatever health plan you choose. Taking into account tax advantages, the government pays between 80 and 90 percent of premium costs for almost all employees. Nationally, about 200 plan options are offered. Between twelve and twenty plans are available in most places. Unlike most other government-paid services, you decide which plan you want to buy. If you are n satisfied, you can switch in Open Season.

The FEHBP enrolls almost 9 million persons. Enrollees spend over $20 billion a year through their health plans. About five percent of enrollees will switch among plans in Open Season, based on past trends. The program is the largest "managed competition" system for harnessing consumer choices to contain health insurance costs. Studies have shown that it outperforms both Medicare and private employ plans in coverage, cost control, and consumer satisfaction.

The Office of Personnel Management (OPM) sets minimum financial, administrative, and bene terms and conditions for every plan participating in the program. Insurance companies and OPM agree ea year on contracts setting forth both benefits and costs. OPM publishes brochures and pamphlets to inform you of the employee share of premium costs, and of the benefits under each plan. It also publishes severa pamphlets giving detailed rules on enrolling, changing plans, and appealing a plan's failure to reimburse you. We do not explain all these details in our *Guide*, since you can get them from the OPM Web site, in OPM's annual insurance guide, and in the plan brochures. A few key points are:

- All employees and annuitants can enroll in whatever plan they choose, or not enroll at all.
- Everyone can change plans once per year in Open Season. You also may switch plans or options in circumstances such as marriage, birth of a child, or geographic transfer. If you belong to an HMO, you may enroll in a new plan if you move out of its service area.
- A family enrollment covers only immediate family members: your spouse and children. Coverage f children is only temporary after they marry or reach age 22, unless they are severely handicapped. I that case they may be eligible at any age.
- Plans cannot exclude coverage for any preexisting conditions or illnesses your family may have wh you switch plans. You may switch to gain the best coverage for your condition, and use the new pla without penalty.
- Each plan must pay for the medical and related costs explained in its brochure, and only that—no more and no less. Some brochures use open-ended language such as "including, but not limited to," and sometimes cover more than they list in a particular category. But if a brochure says that a particular category of service is limited or excluded, believe it.
- Plan brochures may word the same benefits differently. Sometimes the wording used is not clear to layperson. However, **you can often figure out what specific benefit language really means by comparing two or three brochures to see how they differ.**

Scenario 2, event 4

7.1.2 Procedural Steps

CAUTION

Momentarily depressing the raise or lower pushbuttons on *C32-SIC-R601A(B)* will cause pump demand to change in increments of 0.1%. Continually depressing the raise or lower pushbuttons will cause pump demand to change at an exponential rate.

8. **SLOWLY LOWER** RFP A(B) speed, using lower pushbutton on *C32-SIC-R601A(B),* **AND OBSERVE** B(A) pump increases speed to maintain reactor level between 182 and 192 inches.

9. **SLOWLY LOWER** RFP A(B) speed until RFP discharge pressure is slightly less than reactor pressure.

10. **IF** RFP A is to be completely shut down, **THEN DEENERGIZE** seal water solenoid valves by performing the following:

    a. **PLACE** 2C-TB1 Circuit 8, *COD-SV-1146*, breaker to *OFF*.

    b. **PLACE** 2C-TB1 Circuit 24, *COD-SV-1147*, breaker to *OFF*.

11. **IF** RFP B is to be completely shut down, **THEN DEENERGIZE** seal water solenoid valves by performing the following:

    a. **PLACE** 2D-TB1 Circuit 23, *COD-SV-1148*, breaker to *OFF*.

    b. **PLACE** 2D-TB1 Circuit 24, *COD-SV-1149*, breaker to *OFF*.

12. **CLOSE** *RFP A(B) DISCH VLV, FW-V3(FW-V4).*

The following are the Reactor Feed Pump Turbine trips:

- Over Speed $\geq$ 5940 rpm (2A,2B,1B) $\geq$ 5990 rpm (1A).
- Low bearing oil pressure $\leq$ 3.5 to 4.5 psig (RFPT and RFP).
- Thrust Bearing Wear 38 to 42 psig (Active and Inactive).
- Reactor vessel High Level 206.5" to 209.5".
- Low RFP suction pressure.
  "A" 30 to 40 psig for 20 seconds.
  "B" 55 to 65 psig for 15 seconds.
- Manual.

The feed pump and feedwater system controls are explained in System Description 0SD-32.2.

2.14 Feed Pump Suction and Discharge Isolation Valves

Each feed pump is equipped with a motor operated suction and discharge isolation valve. These valves are equipped with a three position spring return to neutral (CLOSE/NEUT/OPEN) control switch located on RTGB Panel XU-2.

The suction valve provides an interlock with the speed control circuit which prevents speed increase with the Motor Speed Changer (MSC) unless the valve limit switch indicates full open.

2.15 Reactor Feed Pump Recirculation (Figure 4)

The air operated recirculation valve for each Reactor Feed Pump is controlled by a control switch located on RTGB Panel XU-2. Each is a two-position (OPEN/AUTO) control switch.

Each recirculation valve is controlled by a normally energized air operated solenoid valve. As feedwater flow drops below 3000 gpm (1.4 Mlbm/hr), a flow switch deenergizes the associated solenoid valve and opens the RFP recirculation valve. As feedwater flow increases above 9000 gpm (4.17 Mlbm/hr), the RFP recirculation valve goes closed. The recirculation valve can be manually opened by placing the RTGB control switch in the OPEN position.

Scenario 2, event 5

COND BOOSTER PUMP B TRIP

AUTO ACTIONS

1. Condensate booster pump breaker trips.

CAUSE

1. Instantaneous or time overcurrent on any phase.
2. Circuit malfunction.

OBSERVATIONS

R3

> **NOTE:** A condensate booster pump trip at power could cause reactor water level transients or feedwater flow fluctuations. Monitor reactor water level, feedwater flow, and steam flow indications and take the appropriate actions of AOP-23.0, Condensate/Feedwater System Failure if necessary.

1. Condensate booster pump discharge header pressure is decreasing.
2. Standby condensate booster pump has automatically started.

ACTIONS

1. If Condensate Booster Pump B has tripped, verify auto start or start the standby condensate booster pump.
2. Determine the cause of the pump trip and ensure that a WR/JO is prepared.
3. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Overcurrent Device 74	Energized

POSSIBLE PLANT EFFECTS

1. Possible RFPT trip on low suction pressure.

REFERENCES

1. 9527-LL-9352 - 36
2. AOP-23.0, Condensate/Feedwater System Failure
3. LER 2-88-018

CP&L CAROLINA POWER & LIGHT COMPANY
BRUNSWICK NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME XXI

ABNORMAL OPERATING PROCEDURE

UNIT
0

0AOP-23.0

CONDENSATE/FEEDWATER SYSTEM FAILURE

REVISION 17

TITLE: **CONDENSATE/FEEDWATER SYSTEM FAILURE**

1.0 SYMPTOMS

1.1 *REACTOR WATER LEVEL HIGH/LOW* (A-07 2-2) annunciator.

1.2 Steam flow/feed flow mismatch as indicated by *C32-R607*.

1.3 Indicated reactor vessel level above +192 or below +182 inches.

1.4 Condensate, condensate booster, heater drain or RFP trip annunciators on UA-4 alarming.

1.5 Feedwater flow controller not responding to level changes.

1.6 If either substation E or F should lose power, RFP A or B respectively, will lose power to its *MGU* servo motor.

1.7 Recirculation pump runback.

2.0 AUTOMATIC ACTIONS

2.1 Recirculation pumps may runback.

2.2 The standby condensate pump may start on low condensate booster pump suction pressure.

2.3 The standby condensate booster pump may start on low RFP suction pressure.

2.4 The condensate booster pump(s) may trip on low suction pressure.

2.5 RFP(s) may trip on low suction pressure.

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

3.1.1 **IF** automatic level control will **NOT** restore level, **THEN TAKE** manual control of the feed pumps **AND RESTORE** normal level.

3.1.2 **IF** both reactor feed pumps are in service, **AND** reactor power must be reduced to stabilize level, **THEN REDUCE** reactor power as necessary in accordance with 0ENP-24.0.

3.0 OPERATOR ACTIONS

> **CAUTION**
>
> (Unit 2 Only): **IF**, while manually reducing core flow in the following step, APRM Upscale alarms are received, **THEN** manual core flow reduction shall be terminated.
>
> **CAUTION**
>
> **IF** core flow is being manually reduced in the following step, **AND** the automatic runback signal is received, **THEN** the automatic runback shall be allowed to control the flow reduction.

3.1.3 **IF** a reactor feed pump has tripped, **THEN PERFORM** the following:

1. **MANUALLY REDUCE** reactor recirculation pump SPEED DEMAND to 45%

OR

2. **CONFIRM** the automatic runback signal has been received **AND** is reducing reactor recirculation pump speed demand.

3.1.4 **IF** level is increasing due to a power loss to an RFP *MGU* servo motor, **THEN USE** the *MSC* to control the affected RFP speed.

3.1.5 **IF** vessel level approaches the RFP trip setpoint, **THEN TRIP** one feed pump.

3.2 Supplementary Actions

3.2.1 **IF** a single reactor feed pump has tripped from high power **AND** reactor power must be reduced following the recirculation runback, **THEN INSERT** control rods in accordance with ENP-24.0 to reduce reactor power to within the capacity of the remaining operating feed pump.

R2 3.2.2 **IF** a reactor power rise is observed while executing this procedure, **THEN REFER** to 1(2)AOP-03.0.

R2 3.2.3 **REFER** to the appropriate figure from COLR Power/Flow Map, attached to Form 2, 0ENP-24.0:

1. (Unit 1 Only): **IF** OPRM System is operable, **AND** the Scram Avoidance Region has been entered, **THEN REFER** to 1AOP-4.0.

3.0 OPERATOR ACTIONS

2. (Unit 1 Only): **IF** OPRM System is inoperable, **AND** there are indications of THI while operating in Region B, the 5% Buffer Region, **OR** the OPRM Enabled Region, **THEN INSERT** a manual reactor scram.

3. (Unit 1 Only): **IF** OPRM System is inoperable, **AND** Region A has been entered, **THEN INSERT** a manual reactor scram.

4. (Unit 1 Only): **IF** OPRM System is inoperable, **AND** Region B **OR** the 5% Buffer Region has been entered, **THEN REFER** to 1AOP-4.0.

5. (Unit 2 Only): **IF** the Monitored Region has been entered, **THEN ENSURE** one channel of PBDS is operable.

6. (Unit 2 Only): **IF** the Monitored **OR** Restricted Regions have been entered, **THEN REFER** to 2AOP-04.0.

3.2.4 **MAINTAIN** hotwell levels between -7 and +7 inches.

3.2.5 **RESTART** feed pumps as necessary to maintain level.

3.2.6 **IF** the condensate demineralizer bypass valves have opened, **THEN CLOSE** these valves as soon as possible **AND NOTIFY** E&RC to commence sampling every four hours.

R1 3.2.7 **IF** the Feedwater Pump Recirc, the Condensate Pump Recirc, or the Condensate Booster Pump Recirc Valves indicate open, **THEN CLOSE** any valve found in the open position **OR ISOLATE** that line with a manual valve.

4.0 GENERAL

High or low water level during power operation is an abnormal condition which could result in major damage to the HPCI, RCIC, RFP, and main turbines, jet pumps, and recirculation pumps.

High water level may result in excessive moisture carryover, causing erosion of turbine blading. Low water level may cause steam carryunder which could lead to cavitation in recirculation and jet pumps and excessive core internal vibration.

Automatic level control is preferred. Manual control should be taken only if operation in automatic is unsafe or would cause unnecessary transients. If two feed pumps are in operation and the feed pumps have opposite demand signals, take manual control of the pump whose demand signal coincides with the direction of the level change and attempt to control water level.

5.0 REFERENCES

5.1 LER 2-88-018

5.2 NEDO-32339, Reactor Stability Long-Term Solution: Enhanced Option 1-A

5.3 1(2) AOP-3.0, Positive Reactivity Addition

5.4 1(2) AOP-4.0, Low Core Flow

5.5 ESR 00-00442, Replace Unit 1 Power Range Neutron Monitoring System

6.0 ATTACHMENTS

1 - Condensate System Setpoints

ATTACHMENT 1
Page 1 of 1
Condensate System Setpoints

Recirculation pumps ramp down to 45% (limiter #2)	**UNIT 1 ONLY:** Either RFP's flow is less than 16.4% of rated, **AND** Level decreases to less than +182 inches **UNIT 2 ONLY:** Either RFP's flow is less than 20% of rated **AND** Level decreases to less than +182 inches
Recirculation pumps ramp down to 28% (limiter #1)	**UNIT 1 ONLY:** Total feedwater flow is less than 16.4% of rated **UNIT 2 ONLY:** Total feedwater flow is less than 20% of rated
Standby condensate pump starts	1. Associated suction valve is open, **AND** 2. Condensate booster pump suction pressure less than 20 psig
Standby condensate booster pump starts	RFP suction header pressure is less than 165 psig
Condensate booster pump trips	Condensate booster pump suction pressure is less than 22 psig for 28 seconds
RFP B trips	RFP suction header pressure is less than 60 psig for 15 seconds
RFP A trips	RFP suction header pressure is less than 35 psig for 20 seconds

REVISION SUMMARY

Revision 17 incorporates EC 46861, Extended Power Uprate setpoint changes and EC 49319, Condensate Pumps Setpoint/Auto-Start Logic Change, both for Unit 1.

Revision 16 incorporates ESR 00-00442, which modified Unit 1 Neutron Monitoring System and Power/Flow Map operating regions.

Revision 15 incorporates recommended action in accordance with A/R 25864-09, to add operator actions to reduce reactor power, within appropriate limitations, to mitigate the consequences of a single reactor feed pump trip from high power.

Revision 14 revised to add necessary instruction to meet the changes to the plant from ESRs 95-00080, 95-00081, and 96-00499 which implement Reactor Stability Long Term Solution: Option 1-A. Upgrade to 0AP-005 format per the writers guide.

Scenario 3- ATWS with MSIV closure

Description:

The crew assumes the shift with power at 90%? The previous shift initiated 0GP-13 "Increasing Unit Capacity at End of Core Cycle", Section 5.1 (Bypassing Feedwater Heaters #4 and #5).

(N) Night orders include direction to complete a Control Rod Operability Check on rod 42-39 per 0PT-14.1 (*all rods but 42-39 were already completed*) and then to increase power to 100%. AT LEAST 2 RODS.

(R) The RO completes 0PT-14.1 and commences power increase to 100%.

(I) Following power increase to 100% APRM 1 fails HI (MNI031F). *This is intended to simply create an instrument malfunction for the RO, but may be interpreted by the crew as being associated with 0GP-13.* The RO is expected to bypass the APRM and reset the half scram. SRO will check TS.

(I) One channel of MS Radiation fails low (MRM001F) . The crew is expected to take appropriate actions per Alarm Response Procedures and TS and may again attribute the failure to actions taken for 0GP-13. ARE THERE ANY ACTIONS. -

The SDV vent and drain valves are failed closed to provide a high level alarm on Scram Discharge Volume(s). (Malfunction MDR036F). *This is intended to be a precussor to an ATWS following SCRAM from MS Line Hi Rad (MNB005F)*. While a AO is sent to investigate the SDV problem (vent and drain valves fail closed) the fuel failure is increased and is readily detectable on the three remaining MS Rad Monitors as well as the Off Gas Monitor(s).

The crew may attempt to reduce power, but MS radiation will continue to increase (MRM0011F through 13F). *The increase is intended to be slow enough to allow the SRO to make the decision to manually Scram and close the MSIVs before automatic action occurs. With the feedwater heaters bypassed there are restrictions to how low the power may be reduced. This may complicate the decision and may "push" the SRO to manually scram earlier rather than later.*

When the SCRAM occurs MRP009F will keep sufficient rods out to result in power between 5% and 10%. The crew is expected to enter Power/Level control and may enter EOP-02-PCCP if suppression pool temperature exceeds 95 F. Once power is <1%, rod motion will be allowed by driving. Once rod motion is started the scenario is terminated.

Event No.	Malf. No.	Event Type	Description
1	N/A	N(RO)	Control Rod Operability Check (42-39 only)
2	N/A	R(RO)	Increase power to 100%
3	MRM001F	I(BOP)	One channel of MS Rad Monitors fails low
4	MNI031F	I(RO)	APRM 1 Fails HI
5	MDR036F	C(RO)	Hi level on Scram Discharge Volume
6	MRM0011F , 0012F and 0013F	C(BOP)	Fuel failure causes increasing radiation levels on remaining MS Rad Monitors
7	MRP009F	M(ALL)	MSIV closure on Hi Rad (may be manually initiated) with ATWS following scram

Note: SRO(I) in RO position

Time (Minutes)	Position	Applicant's Actions or Behavior
T=0	All	Accept shift, review panels. The shift may review 0GP-13 to determine configuration/limits.
T=5	RO	Performs operability check on 42-39 per 0PT-14.1
T=10	RO	Commences power increase to 100% per 0GP-12. RO should coordinate with BOP to ensure Feedwater and Turbine-Generator are following the reactor power increase. SRO may consult 0GP-13 and call the Reactor Engineer to ensure thermal-hydraulic limits are met with the feedwater heaters bypassed.
T=30	RO	Observes APRM HI, consults with SRO and bypasses the APRM. Then he will reset the half scram. SRO will ensure applicable TS are satisfied. There may be some discussion on the possible effects from 0GP-13.
T=35	RO	Observes MS Rad low alarm and takes appropriate action per Alarm Response Procedure. The BOP may be used to take actions on the "back panels" (may have to be simulated). There may be some discussion on the possible effects from 0GP-13.
T=40	RO	Will respond to Hi SDV level alarm (or may observe the vent/drain valves going closed). If an AO is dispatched he will report that "the valve solenoids are cool to the touch and the SDV level is slowly increasing but everything else looks normal".
T=45	BOP	Observes MS rad levels are increasing (or responds to alarms). The SRO may order the power reduced. However, rad levels will continue to increase (will reach scram setpoint in 5 minutes).
T=50	RO	Following Scram he will observe (and announce) that all rods are NOT at 04 and power is above 5% (should be between 5% and 10%). He will continue in the Reactor Scram procedure (EOP-01-RSP) until directed to exit by the SRO.
T=51	SRO	Will exit RSP and enter EOP-01-LPC "Level/Power Control". The BOP may be used to get suppression pool cooling on. If/when the suppression pool approaches 95 F the SRO will enter 0EOP-02-PCCP, "Primary Containment Control"
T=55	SRO	Will either direct the RO to initiate SLC (if power remains above 4% when recirc pumps are tripped) or attempt to drive rods per EOP-01-LEP-02 (if power is <4%). RO will be inserting IRMs or using other means to determine power is <4%. BOP will be assisting RO to maintain containment parameters within limits while attempting to get rods inserted.