FARLEY JAN. 2005 EXAM
50-348 & 50-364/2005301

JANUARY 10 - 14, 2005
JANUARY 18, 2005 (written)

Farley SRO Written Draft

Draft Submittal

FARLEY JAN. 2005 EXAM
50-348 & 50-364/2005301

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1. Senior Reactor Operator Written Exam

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002A2.03 001/2/2/NATURAL CIRC/C/A 4.3/NEW/FA011005/S/GTO

Unit 2 is at 8% power with a startup in progress with the following conditions:

- An electrical fault occurs due to a grid perturbation.
- 2A, 2B & 2C 4160V busses are de-energized.
- All other 4160V busses remain energized after the perturbation and voltage is stable.

Which one of the following is the correct plant response and then procedural flow path for the above conditions?

A. An automatic reactor trip will occur; enter EEP-0, Reactor Trip or Safety Injection, ESP-0.1, Reactor Trip Response, ESP-0.2, Natural Circulation Cooldown to prevent Vessel Head Steam Voiding.

B. An automatic reactor trip will occur; enter EEP-0, ESP-0.1, UOP-2.3, Shutdown of Unit Following Reactor Trip.

C. An automatic reactor trip will NOT occur; enter AOP-4.0 and trip the reactor, then go to EEP-0, ESP-0.1, and UOP-2.1.

D✓ An automatic reactor trip will NOT occur; enter AOP-4.0, Loss of Reactor Coolant Flow, then shutdown per UOP-2.1, Shutdown of Unit from Minimum Load to Hot Standby, and return to AOP-4.0 for additional guidance.

A. Incorrect. Below P-7 (10%), loss of all RCPs does not initiate Auto Reactor trip. If it did, since there is no reason given to require a cooldown, ESP-0.2 would be incorrect anyway.

B. Incorrect. Below P-7 (10%), loss of all RCPs does not initiate Auto Reactor trip. If it did, such as if this occured above 10%, the procedure flow path would be correct.

C. Incorrect. Even though manually securing RCPs at power without tripping the Reactor first is prohibited by plant policy, Tech Specs allows controlled shutdown after RCPs trip at power as long as the permissives are met (<P-7 for loss of all RCPs & <P-8 for loss of up to two RCPs). UOP-2.1 would not be transitioned to from ESP-0.1 unless 1 RCP was running.

D. Correct. **An automatic reactor trip will NOT occur; enter AOP-4.0, Loss of Reactor Coolant Flow, then shutdown per UOP-2.1 and return to AOP-4.0 for additional guidance.**

An automatic reactor trip will not occur. AOP-4.0 is entered when no auto trip is supposed to occur or does not occur whenever any RCP trips in modes 1-4. The procedure directs shutdown per UOP-3.1 & 2.1 within 6 hours (which is also the Tech Spec requirement). AOP-4 is being completed along with UOP guidance and UOP-2.1 is set up for having at least 1 RCP running. Since no RCP is running, AOP-4 will direct transition to ESP -0.2 or to start 1 RCP prior to transitioning out.

002A2.03 Reactor Coolant
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.5)
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A2.03 – Loss of forced circulation.

Evaluate abnormal plant or equipment conditions associated with the Reactor Coolant Pumps and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52101D02).

Evaluate abnormal plant or equipment conditions associated with the Reactor Coolant System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52101A02).

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-4.0

LOSS OF REACTOR COOLANT FLOW

SAFETY RELATED

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	ALL
Reference Use	
Information Use	

Approved:

TODD YOUNGBLOOD
Operations Manager

Date Issued: 05/05/2003

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A. **Purpose**

This procedure provides actions for response to a loss of forced RCS flow in one or more loops when a reactor trip is not required.

This procedure is applicable in Modes 1, 2, 3 and 4.

B. **Symptoms or Entry Conditions**

1. This procedure is entered when forced RCS flow is lost in one or more loops and no reactor trip is required.

Step	Action/Expected Response	Response NOT Obtained

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CAUTION: To prevent a possible reactor power excursion, no RCP should be restarted while the reactor is critical.

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NOTE:
- Loss of flow in one loop with reactor power greater than 30% will cause a reactor trip.
- Loss of flow in two loops with reactor or turbine power greater than 10% will cause a reactor trip.
- SG level in any affected loop will tend to shrink.

___ 1 **Stop any load change in progress.**

___ 2 **Maintain SG narrow range level stable at 65%.**

2.1 IF SGs supplied from Main Feedwater, THEN manually control Feedwater Flow to the affected SG.

Affected SG	1A	1B	1C
1A(1B,1C) SG FW FLOW FK	[] 478	[] 488	[] 498

OR

2.2 Manually control Main Feedwater Bypass flow to affected SG.

Affected SG	1A	1B	1C
1A(1B,1C) SG FW BYP FLOW FK	[] 479	[] 489	[] 499

Response NOT Obtained:

2 IF SG level rise cannot be controlled, THEN close the affected SG main feedwater stop valve.

Affected SG	1A	1B	1C
MAIN FW TO 1A(1B,1C) SG STOP VLV Q1N21MOV	[]3232A	[]3232B	[]3232C

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

___ 3 **Check 1A and 1B RCPs - RUNNING.**

3 Manually close pressurizer spray valve for affected RCP.

Affected RCP	1A	1B
1A(1B) LOOP SPRAY VALVE PK	[] 444C	[] 444D

4 **Maintain PRZR pressure 2200 psig-2300 psig.**

4.1 IF available, THEN use non-affected pressurizer spray valve(s).

Step 4 continued on next page.

___ Page Completed

Step	Action/Expected Response	Response NOT Obtained

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<u>CAUTION</u>: Thermal shock to the spray nozzle will occur if auxiliary spray is initiated without normal letdown in service.

****

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<u>CAUTION</u>: To prevent PRZR PORV failure, cycling of PRZR PORVs should be minimized.

****

****

<u>CAUTION</u>: The PRT may rupture causing abnormal containment conditions while using PRZR PORVs.

****

4.2 <u>IF</u> normal pressurizer spray unavailable <u>OR</u> ineffective, <u>THEN</u> control RCS pressure with auxiliary spray.

4.2.1 Verify normal letdown established.

4.2.2 Establish auxiliary spray by performing the following.

1A(1B) LOOP
SPRAY VLV
[] PK 444C manually closed
[] PK 444D manually closed

RCS PRZR
AUX SPRAY
[] Q1E21HV8145 open

RCS NORMAL
CHG LINE
[] Q1E21HV8146 closed

RCS ALT
CHG LINE
[] Q1E21HV8147 closed

Response NOT Obtained:

4.2 Perform the following.

a) Open only one PRZR PORV for any RCS pressure reduction.

b) Maintain PRT parameters normal using FNP-1-SOP-1.2, REACTOR COOLANT PRESSURE RELIEF SYSTEM.

c) <u>IF</u> any PRZR PORV fails to reclose, <u>THEN</u> close associated PRZR PORV ISO.

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

4.2.3 Control PRZR heaters.

PRZR HTR GROUP
VARIABLE
[] 1C

PRZR HTR GROUP
BACKUP
[] 1A
[] 1B
[] 1D
[] 1E

NOTE: Changes in charging flow will cause PRZR pressure fluctuations when auxiliary spray is established.

4.2.4 Operate the following valves as required to control RCS pressure.

CHG FLOW
[] FK 122 adjusted

1A(1B) LOOP
SPRAY VLV
[] PK 444C manually open/closed
[] PK 444D manually open/closed

RCS PRZR
AUX SPRAY
[] Q1E21HV8145 open/closed

RCS NORMAL
CHG LINE
[] Q1E21HV8146 open/closed

RCS ALT
CHG LINE
[] Q1E21HV8147 open/closed

___Page Completed

Step	Action/Expected Response		Response NOT Obtained
___ 5	**Verify normal letdown - ESTABLISHED.**	5	Establish excess letdown using FNP-1-SOP-2.7, CHEMICAL AND VOLUME CONTROL SYSTEM EXCESS LETDOWN.
___ 6	**Maintain PRZR level 19%-24%.**		

NOTE: Steps 7 and 8 must be completed within six hours of the loss of RCS flow.

Step	Action/Expected Response		Response NOT Obtained
___ 7	**IF unit in Mode 1 or 2, THEN place unit in Mode 3 using FNP-1-UOP-3.1, POWER OPERATION and FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.**		
___ 8	**WHEN unit in Mode 3 or 4, THEN verify all reactor trip and reactor trip bypass breakers open.** [] Reactor trip breaker A [] Reactor trip breaker B [] Reactor trip bypass breaker A [] Reactpr trip bypass breaker B	8	Secure both CRDM MG sets using FNP-1-SOP-41.0, CONTROL ROD DRIVE AND POSITION INDICATION SYSTEM.

NOTE: Step 8 must be complete before continuing with this procedure.

Step	Action/Expected Response		Response NOT Obtained
___ 9	**Check at least one RCP - RUNNING.**	9	Perform the following. 9.1 Secure any dilution in progress. 9.2 Start one RCP using FNP-1-SOP-1.1, REACTOR COOLANT SYSTEM.

__ Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- As natural circulation develops, RCS ΔT should stabilize at less than 65°F. This will take 20 to 30 minutes.
- During natural circulation, loop transit time will be on the order of 10 minutes. Temperature trends will be of more value than actual temperatures.
- Changes in SG feeding or steaming rates must be made slowly to prevent rapid RCS pressure changes.

Step	Action/Expected Response	Response NOT Obtained
___10	**IF at least one RCP running, THEN go to FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.**	10 Verify adequate natural circulation. 10.1 Check SG pressures stable or falling. 10.2 Check SUB COOLED MARGIN MONITOR indication greater than 16°F subcooled in CETC mode. 10.3 Check RCS hot leg temperatures stable or falling. RCS HOT LEG TEMP [] TR 413 10.4 Check core exit T/Cs stable or falling. 10.5 IF natural circulation NOT adequate, THEN dump steam at a faster rate. 10.6 Begin taking natural circulation logs.

___Page Completed

Step | **Action/Expected Response** | **Response NOT Obtained**

****

CAUTION: During natural circulation with low decay heat loads, overfeeding the SGs will result in excessive cooldown with no apparent rise in SG level.

****

___11 **Maintain SG narrow range levels 61-69%.**

- Control MDAFWP flow.

MDAFWP TO 1A(1B,1C) SG
FLOW CONT
[] HIC 3227AA adjusted
[] HIC 3227BA adjusted
[] HIC 3227CA adjusted

- Control TDAFWP flow.

TDAFWP TO 1A(1B,1C) SG
FLOW CONT
[] HIC 3228AA adjusted
[] HIC 3228BA adjusted
[] HIC 3228CA adjusted

___12 **Monitor CST level.**

CST
LVL
[] LI 4132A
[] LI 4132B

12.1 IF CST level less than 5.3 ft, THEN align AFW pump suction to SW using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___13	**Verify reactor makeup system aligned.**	13 Manually control reactor makeup system using FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM.
13.1	Verify BORIC ACID MKUP FLOW FK 113 - ADJUSTED TO DELIVER GREATER THAN EXISTING RCS BORON CONCENTRATION.	
13.2	Verify reactor makeup system - IN AUTOMATIC MODE. MKUP MODE SEL SWITCH [] N1E21HS2100Q in AUTO MKUP MODE CONT SWITCH [] N1E21HS2100P to START	
___14	**Maintain shutdown margin adequate.**	
14.1	Direct Chemistry to sample RCS for boron concentration using FNP-1-CCP-651, SAMPLING THE REACTOR COOLANT SYSTEM at least once per two hours.	

NOTE: Natural circulation may cause inadequate RCS mixing. Any boration should be at a continuous rate not exceeding 65 gpm.

Step	Action/Expected Response	Response NOT Obtained
14.2	Verify shutdown margin using FNP-1-STP-29.1, SHUTDOWN MARGIN CALCULATION (TAVG 547°) or FNP-1-STP-29.2, SHUTDOWN MARGIN CALCULATION (TAVG < 547°F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING).	14.2 Borate RCS using FNP-1-AOP-27.0, EMERGENCY BORATION.
14.3	Monitor source range count rate stable or falling.	14.3 Borate RCS using FNP-1-AOP-27.0, EMERGENCY BORATION.

___Page Completed

Step	Action/Expected Response		Response NOT Obtained
___15	**Maintain hot standby conditions.** • Maintain steam header pressure less than or equal to 1005 psig. • Maintain average core exit T/C temperature less than 600°F.		
___16	**<u>IF</u> RCS cooldown <u>NOT</u> required, <u>THEN</u> return to step 9.**	16	Go to FNP-1-ESP-0.2, NATURAL CIRCULATION COOLDOWN TO PREVENT REACTOR VESSEL HEAD STEAM VOIDING to initiate RCS cooldown.

-END-

Step	Action/Expected Response	Response NOT Obtained
17.4	WHEN emergency bus(es) re-energized from offsite power, THEN perform the following for affected bus(es):	
17.4.1	Locally reset the loss of voltage indicating lamp on the B1F (B1G) sequencer auxiliary panel.	
17.4.2	At the EPB for the B1F (B1G) sequencer, push the LAMP RESET pushbuttons.	
___18	**Determine controlling procedure.**	
18.1	Check at least one RCP - STARTED.	18.1 Perform the following.
		18.1.1 Notify Chemistry to secure the zinc addition system (ZAS).
		18.1.2 IF cooldown required, THEN go to FNP-1-ESP-0.2, NATURAL CIRCULATION COOLDOWN TO PREVENT REACTOR VESSEL HEAD STEAM VOIDING.
18.2	Go to FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.	

-END-

Step	Action/Expected Response	Response NOT Obtained

NOTE: To ensure adequate pressurizer spray, the priority for establishing RCP support conditions is 1B, 1A and then 1C.

14 Verify RCP support conditions established.

****

CAUTION: To prevent potential seal damage, neither seal injection nor CCW cooling should be restored to a RCP which has lost both seal injection and CCW cooling.

****

Action/Expected Response	Response NOT Obtained
14.1 Verify No. 1 seal support conditions established.	14.1 IF any No. 1 seal support condition NOT established, THEN stop associated RCP.
14.1.1 Maintain seal injection flow - GREATER THAN 6 gpm.	
14.1.2 Verify No. 1 seal leakoff flow - WITHIN FIGURE 1 LIMITS.	
14.1.3 Verify No. 1 seal differential pressure - GREATER THAN 210 psid.	
14.2 Verify CCW - ALIGNED. CCW FROM RCP THRM BARR [] Q1P17HV3045 open [] Q1P17HV3184 open	
14.3 Check RCP thermal barrier - INTACT. RCP THRM BARR CCW FLOW HI [] Annunciator DD2 clear	14.3 Verify CCW flow isolated. CCW FROM RCP THRM BARR [] Q1P17HV3045 closed [] Q1P17HV3184 closed

Step 14 continued on next page.

___Page Completed

Step / Action/Expected Response	Response NOT Obtained
14.4 Verify at least one RCP bus - ENERGIZED. [] 1A 4160 V bus [] 1B 4160 V bus [] 1C 4160 V bus	14.4 Proceed to step 15.
14.5 Check CCW to RCP oil coolers - SUFFICIENT. CCW FLOW FROM RCP OIL CLRS LO [] Annunciator DD3 clear	14.5 Perform the following. 14.5.1 Verify CCW aligned. CCW TO RCP CLRS [] Q1P17MOV3052 open CCW FROM RCP OIL CLRS [] Q1P17MOV3046 open [] Q1P17MOV3182 open 14.5.2 <u>IF</u> annunciator DD3 clear, <u>THEN</u> proceed to step 14.6. <u>IF NOT</u>, stop all RCPs. 14.5.3 Proceed to step 15.
14.6 Check RCP oil level - SUFFICIENT. RCP 1A(1B,1C) BRG UPPER/LOWER OIL RES LO LVL [] Annunciator HH1 clear [] Annunciator HH2 clear [] Annunciator HH3 clear	14.6 <u>IF</u> any RCP oil level <u>NOT</u> sufficient, <u>THEN</u> stop associated RCP.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: To ensure adequate pressurizer spray, the priority for operating RCPs is 1B, 1A and then 1C.

15	**Check at least one RCP - STARTED.**	15 Perform one of the following. • IF support conditions exist to start an RCP, THEN start only one RCP. a) Start bearing oil lift pump. RCP OIL LIFT PUMP [] 1B(1A,1C) b) Check oil lift pressure indicating light - LIT. c) Start RCP. RCP [] 1B(1A,1C) d) WHEN RCP has operated for one minute, THEN stop bearing oil lift pump. RCP OIL LIFT PUMP [] 1B(1A,1C) OR

Step 15 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

- IF support conditions do NOT exist, THEN verify adequate natural circulation.

  a) Check SG pressure stable or falling.

  b) Check SUB COOLED MARGIN MONITOR indication greater than 16°F subcooled in CETC mode.

  c) Check RCS hot leg temperatures stable or falling.

  RCS HOT LEG TEMP
  [] TR 413

  d) Check core exit T/Cs stable or falling.

  e) IF natural circulation NOT adequate, THEN dump steam at a faster rate.

  f) Begin taking natural circulation logs.

Page Completed

LOCATION EF1

F1
1A RCS LOOP FLOW LO OR 1A RCP BKR OPEN

SETPOINT: 1. Flow: 90% of Normal
2. Breaker open

ORIGIN: 1. Loop 1A Flow:
a) Flow Transmitter (Q1B21FT414)
b) Flow Transmitter (Q1B21FT415)
c) Flow Transmitter (Q1B21FT416)
2. 4KV Breaker DA04 Auxiliary Contact

PROMPT OPERATOR ACTION REQUIRED

PROBABLE CAUSE

1. Loss of power to RCP 1A.
2. RCP 1A breaker tripped due to electrical or mechanical fault.

AUTOMATIC ACTION

1. None if below P-8 setpoint (30% Power) and loops 1B and 1C are normal.

NOTE: Reactor trip will not occur when above P-8 setpoint if alarm is due to failure of a single flow transmitter.

2. Reactor Trip will occur if above P-8 setpoint (30% power).

IMMEDIATE ACTION

1. VERIFY LOSS OF FLOW FROM MCB INDICATION (FI-414/FI-415/FI-416 AND FROM TSLB-2).
2. PERFORM THE ACTIONS REQUIRED BY FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW
3. IF A REACTOR TRIP HAS NOT OCCURRED, THEN IMMEDIATELY CHECK STEAM GENERATOR LEVELS. IF NECESSARY, THEN MANUALLY CONTROL THE FEED REG. VALVES AND/OR THE FEED REG. BYPASS VALVES TO RE-ESTABLISH NORMAL LEVEL IN ALL STEAM GENERATORS.

SUPPLEMENTARY ACTION

1. Place the plant in Hot Standby in accordance with FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY, and restart the affected RCP per FNP-1-SOP-1.1, REACTOR COOLANT SYSTEM.
2. IF the tripped RCP can NOT be immediately restarted, THEN make the rod control system incapable of rod withdrawal.
3. Refer to Technical Specifications LCOs 3.4.4, 3.4.5, and 3.4.6.

References: A-177100, Sh. 256; U-211024; U-260610; PLS Document

77. 004A2.14 001/2/1/EMER BORATE/C/A 3.9/NEW/FA011005/S/CVR/GTO

Given the following conditions during a ramp down IAW AOP-17.0, Rapid Load Reduction:

- 2A BAT is on service.
- 2B BAT is on standby.
- FE2, CONT ROD BANK POSITION LO-LO, is in Alarm.

The OATC initiates an emergency boration using MOV-8104, EMERG BORATE TO CHG PUMP SUCT, when the 2A BAT Pump trips and cannot be restarted.

Which one of the following actions will meet AOP-27, Emergency Boration, requirements for this situation?

A. Start the 2B BAT Pump, verify one letdown orifice on service, verify boration flow of 35 gpm and charging flow of 35 gpm.

B✓ Start the 2B BAT Pump, verify two letdown orifices on service, verify boration flow of 40 gpm and charging flow of 45 gpm.

C. Open LCV115B, RWST TO CHG PUMP and Close LCV115C, VCT OUTLET ISO, verify one letdown orifice on service, and verify charging flow of 90 gpm.

D. Open LCV115D, RWST TO CHG PUMP and Close LCV115C, VCT OUTLET ISO, verify two letdown orifices on service, and verify charging flow of 45 gpm.

A. INCORRECT. The AOP directs the emergency boration flowpath to be aligned from the Bat pump and this also requires two orifices on service and charging flow >40gpm, which is clearly not the case here.

B. CORRECT. **Start the 2B BAT Pump, two letdown orifices on service, boration flow of 40 gpm and charging flow of 45 gpm.**
Since the 2A Bat pump is available, the procedure has the team start that pump and check for certain flow rates with 2 letdown orifices aligned. Charging flow rates are >40 gpm and normal boration flow is >30 gpm.

C. INCORRECT. If this path were to be used which is not the proper intent of this part of the procedure, Two orifices must be aligned, not one, and charging flow must be > 92 gpm.

D. Incorrect - If this path were to be used which is not the proper intent of this part of the procedure, Two orifices must be aligned which is correct, however charging flow must be > 92 gpm.

004A2.14 Chemical and Volume Control Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those malfunctions or operations:(CFR: 41.5/ 43/5 / 45/3 / 45/5)
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A2.14 – Emergency Boration

Evaluate abnormal plant or equipment conditions associated with the Reactor Makeup and Chemical Addition System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52101G02).

LOCATION FE2

E2
CONT ROD BANK POSITION LO-LO

SETPOINT: Variable with Reactor Power as measured by ΔT and TAVG.

ORIGIN: Rod Insertion Limit Computer

PROBABLE CAUSE

NOTE:
- **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**
- **This annunciator has REFLASH capability.**

Reactor Coolant System Boric Acid Concentration too low to ensure Reactor Protection under Accident conditions due to;

a) Plant Transient
b) Xenon Transient
c) Dilution of RCS

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. CHECK INDICATIONS AND DETERMINE THAT ACTUAL CONTROL BANK ROD POSITION IS AT THE LOW-LOW INSERTION LIMIT.
2. EMERGENCY BORATE THE REACTOR COOLANT SYSTEM IN ACCORDANCE WITH FNP-1-AOP-27.0, EMERGENCY BORATION. **{CMT 0008555, 0008900}**
3. IF A PLANT TRANSIENT IS IN PROGRESS, THEN PLACE TURBINE LOAD ON "HOLD".

SUPPLEMENTARY ACTION

1. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
2. Refer to the Technical Specifications section on Reactivity Control.

References: A-177100, Sh. 292; U-260610; U266647 PLS Document; Technical Specifications; DCP 93-1-8587; **{CMT 0008887}**

PROMPT OPERATOR ACTION REQUIRED

UNIT 1

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-27.0

EMERGENCY BORATION

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	
Reference Use	ALL
Information Use	

SAFETY RELATED

Approved:

RAY MARTIN
Operations Manager

Date Issued: 02/17/2004

A. Purpose

This procedure provides actions to emergency borate the RCS when a reactor trip is not required.

This procedure is applicable in Modes 1, 2, 3, 4, 5 and 6.

B. Symptoms or Entry Conditions

I. This procedure is entered when emergency boration is required by any of the following:

a. Shutdown margin is determined to be less than required by Technical Specifications (or the TRM)

b. Unexplained or uncontrolled reactivity insertion

c. Actuation of CONT ROD BANK POSITION LO-LO annunciator FE2

d. Inadvertent cooldown below 525°F with critical boron concentration established

Step	Action/Expected Response	Response NOT Obtained
___ 1	Start a boric acid transfer pump. BATP [] 1A [] 1B	1 Perform the following. 1.1 Align charging pump suction to RWST. RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed 1.2 Proceed to step 3.

NOTE: IF emergency boration is being aligned to the manual emergency boration flow path, THEN consideration should be given to starting a boration through the blender via FCV113A & B in accordance with FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, while personnel are being dispatched to locally open Q1E21V185.

Step	Action/Expected Response	Response NOT Obtained
___ 2	Align normal emergency boration flow path. EMERG BORATE TO CHG PUMP SUCT [] Q1E21MOV8104 open	2 Align manual emergency boration flow path. BORIC ACID TO BLENDER [] Q1E21FCV113A open MAN EMERG BORATION [] Q1E21V185 open (100 ft. AUX BLDG rad-side chemical mixing tank area)
___ 3	Verify at least one CHG PUMP - STARTED.	

___ Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 4	**Establish adequate letdown.**	
	4.1 Verify 45 gpm letdown orifice - IN SERVICE. LTDN ORIF ISO 45 GPM [] Q1E21HV8149A open	
	4.2 Verify at least one 60 gpm letdown orifice - IN SERVICE. LTDN ORIF ISO 60 GPM [] Q1E21HV8149B open [] Q1E21HV8149C open	
___ 5	**Establish adequate charging flow.**	
	• IF boration is from boric acid storage tank, THEN verify charging flow - GREATER THAN 40 gpm. OR • IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 gpm.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 6	Verify emergency boration flow adequate. • IF normal emergency boration flow path aligned, THEN check emergency boration flow greater than 30 gpm. BORIC ACID EMERG BORATE [] FI 110 OR • IF manual emergency boration flow path aligned, THEN check boric acid flow greater than 30 gpm. MAKEUP FLOW TO CHG/VCT [] BA FI 113 OR • IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 gpm.	6 Verify boration flow path using ATTACHMENT 1.
___ 7	Direct Chemistry to secure the zinc addition system (ZAS).	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
8	Check emergency boration complete.	
8.1	Check reactor - NOT CRITICAL.	8.1 Perform the following. 8.1.1 IF control rod insertion below rod insertion limit, THEN continue emergency boration and return to step 5. IF NOT, proceed to RNO step 8.1.2. 8.1.2 IF emergency borating as a result of inoperable {untrippable} control rods per Tech. Spec. 3.1.3.1 {3.1.4.A}, THEN verify shutdown margin greater than Technical Specification requirement using FNP-1-STP-29.5, SHUTDOWN MARGIN CALCULATION IN MODES 1 AND 2 (TAVG ≥ 547°F) WITH INOPERABLE OR IMMOVABLE CONTROL RODS(S) {WITH UNTRIPPABLE CONTROL ROD(S)}. IF NOT, proceed to step 9. 8.1.3 WHEN shutdown margin greater than Technical Specification requirement, THEN proceed to step 9. 8.1.4 Continue emergency boration and return to step 5.

Step 8 continued on next page.

____Page Completed

Step | Action/Expected Response | Response NOT Obtained

NOTE: In response to an uncontrolled cooldown below 525°F, the cold shutdown boron concentration is the maximum boron concentration required regardless of the extent of the cooldown.

8.2 Check RCS TAVG - LESS THAN 525°F.

8.2.1 IF RCP's are running, THEN use the Tavg indication.

TAVG
1A,(1B,1C) RCS LOOP
[] TI 412D
[] TI 422D
[] TI 432D

8.2.2 IF RCP's are not running, THEN use RCS cold leg temperature indication.

RCS COLD LEG TEMP RECORDER
[] TR 410

8.2 Perform the following.

a) Verify shutdown margin greater than Technical Specification requirement using FNP-1-STP-29.1, SHUTDOWN MARGIN CALCULATION (TAVG 547°F) or FNP-1-STP-29.2, SHUTDOWN MARGIN CALCULATION (TAVG < 547°F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING).

b) WHEN shutdown margin greater than Technical Specification requirement, THEN proceed to step 9.

c) Continue emergency boration and return to step 5.

8.3 Continue emergency boration based on initial boron concentration and RCS TAVG.

Approximate Boration (4 wt% boric acid)

Initial RCS Boron Concentration	Each °F TAVG Is Less Than 525°F
0 ppm	50 gal
300 ppm	52 gal
600 ppm	55 gal
1200 ppm	60 gal
1500 ppm	64 gal
1800 ppm	68 gal

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: The intent of the following step is to optimize the effectiveness of an emergency boration when no RCP is running and RHR is in operation.

___ 9 IF no RCP is running AND RHR is aligned for cooldown operation, THEN perform the following.

9.1 Verify alternate charging path in service.

RCS ALT
CHG LINE
[] Q1E21HV8147 open

RCS NORMAL
CHG LINE
[] Q1E21HV8146 closed

NOTE: Step 8 must be complete before continuing with this procedure.

___ 10 Stop running boric acid transfer pump.

BATP
[] 1A
[] 1B

10 Perform the following.

10.1 Align charging pump suction to VCT.

VCT
OUTLET ISO
[] Q1E21LCV115C open
[] Q1E21LCV115E open

RWST
TO CHG PUMP
[] Q1E21LCV115B closed
[] Q1E21LCV115D closed

10.2 Proceed to step 12.

___ Page Completed

Step	Action/Expected Response	Response NOT Obtained
___11	IF normal emergency boration flow path aligned, THEN secure normal emergency boration flow path. EMERG BORATE TO CHG PUMP SUCT [] Q1E21MOV8104 closed	11 Secure manual emergency boration flow path. BORIC ACID TO BLENDER [] Q1E21FCV113A closed MAN EMERG BORATION [] Q1E21V185 closed (100 ft, AUX BLDG rad-side chemical mixing tank area)
___12	Direct Chemistry to sample RCS for boron concentration using FNP-1-CCP-651, SAMPLING THE REACTOR COOLANT SYSTEM.	
___13	Align reactor makeup system. 13.1 Adjust BORIC ACID MKUP FLOW FK 113 to deliver greater than new RCS boron concentration. 13.2 Verify reactor makeup system IN AUTOMATIC MODE. MKUP MODE SEL SWITCH [] N1E21HS2100Q in AUTO MKUP MODE CONT SWITCH [] N1E21HS2100P to START	13 Manually control reactor makeup system using FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___14	Verify shutdown margin greater than Technical Specification requirement using applicable procedure: • FNP-1-STP-29.1, SHUTDOWN MARGIN CALCULATION (TAVG 547°F) OR • FNP-1-STP-29.2, SHUTDOWN MARGIN CALCULATION (TAVG < 547°F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING) OR • FNP-1-STP-29.5, SHUTDOWN MARGIN CALCULATION IN MODES 1 AND 2 (TAVG ≥ 547°F) WITH INOPERABLE OR IMMOVABLE CONTROL RODS(S) (WITH UNTRIPPABLE CONTROL ROD(S))	

NOTE: After a completion of any fast ramp or emergency boration, the suction piping of any idle charging pump could have a significantly higher boron concentration than the existing RCS. (CE-17609 & AI 2004200233)

Step	Action/Expected Response	Response NOT Obtained
___15	Go to procedure and step in effect.	

-END-

Step | Action/Expected Response | Response NOT Obtained

ATTACHMENT 1

___ 1 IF normal emergency boration flow path aligned, THEN verify running charging pump header valves open.

Running CHG PUMP	1A	1B	1C
CHG PUMP SUCTION HDR ISO Q1E21MOV	[]8130A []8130B []8131A []8131B	[]8131A []8131B	
CHG PUMP DISCH HDR ISO Q1E21MOV		[]8132A []8132B	[]8132A []8132B []8133A []8133B

1 IF manual emergency boration flow path aligned, THEN verify running charging pump header valves open.

Running CHG PUMP	1A	1B	1C
CHG PUMP SUCTION HDR ISO Q1E21MOV		[]8130A []8130B	[]8130A []8130B []8131A []8131B
CHG PUMP DISCH HDR ISO Q1E21MOV		[]8132A []8132B	[]8132A []8132B []8133A []8133B

___ 2 Check boration flow adequate.

2.1 IF normal emergency boration flow path aligned, THEN check emergency boration flow greater than 30 gpm.

BORIC ACID
EMERG BORATE
[] FI 110

2.2 IF manual emergency boration flow path aligned, THEN check boric acid flow greater than 30 gpm.

MAKEUP FLOW
TO CHG/VCT
[] BA
FI 113

2 Align charging pump suction to RWST.

RWST
TO CHG PUMP
[] Q1E21LCV115B open
[] Q1E21LCV115D open

VCT
OUTLET ISO
[] Q1E21LCV115C closed
[] Q1E21LCV115E closed

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

___ 3 Verify charging flow path aligned.

3.1 Verify charging pump discharge flow path - ALIGNED.

CHG PUMPS TO
REGENERATIVE HX
[] Q1E21MOV8107 open
[] Q1E21MOV8108 open

3.2 Verify only one charging line valve - OPEN.

RCS NORM
CHG LINE
[] Q1E21HV8146

RCS ALT
CHG LINE
[] Q1E21HV8147

3.3 IF boration is from the boric acid storage tank, THEN verify charging flow - GREATER THAN 40 gpm.

CHG FLOW
[] FK 122 manually adjusted

Response NOT Obtained: 3.3 IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 gpm.

CHG FLOW
[] FK 122 manually adjusted

___ 4 Notify control room of boration status.

___ 5 Return to step 7.

-END-

LOCATION FD5

SETPOINT: 220 Steps on Control Bank D

ORIGIN: Auxiliary Relay Contact 1-ZY409X actuated by Rod Position Indication (C-11).

D5
BANK D FULL ROD WTHDRL AUTO ROD STOP

PROBABLE CAUSE

1. Bank D control rods withdrawn to 220 steps due to plant transient (i.e. startup; ramp up) or for Xenon oscillation transient.

NOTE: Following C-11 actuation this alarm will remain in for approximately three minutes at which time the alarm will clear and the associated bypass and permissive light box status will illuminate.

AUTOMATIC ACTION

NOTE: Automatic Rod Insertion capabilities still exist.

1. Control Bank D Automatic Rod Withdrawal is Inhibited.

IMMEDIATE ACTION

1. CHECK INDICATIONS AND DETERMINE ACTUAL CONTROL BANK D ROD POSITION AT 220 ON STEP COUNTERS.
2. IF ROD CONTROL IS IN AUTO, THEN PLACE IN MAN IF FURTHER WITHDRAWAL REQUIRED.

SUPPLEMENTARY ACTION

1. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.

References: A-177100, Sh. 290; U-260610; D-177352; D-177398; U-260387; U-260388; U-260389; PCN B91-1-7687; U266647 PLS Document; Technical Specifications.

LOCATION FE1

E1
CONT ROD BANK POSITION LO

SETPOINT: Variable; 10 Steps Greater than LO-LO Alarm Setpoint.
$Z_{LO} = Z_{LO\text{-}LO} + K_4$
Where $K_4 = 10$ Steps (6.25 inches)

ORIGIN: Rod Insertion Limit Computer

PROBABLE CAUSE

NOTE: • **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**

• **This annunciator has REFLASH capability.**

Reactor Coolant System Boric Acid Concentration too low for Reactor Power Level due to:

a) Plant Transient
b) Xenon Transient
c) Dilution of RCS

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. CHECK INDICATIONS AND DETERMINE THAT ACTUAL CONTROL BANK ROD POSITION IS AT LOW INSERTION LIMIT.
2. IF REACTOR COOLANT SYSTEM DILUTION IS IN PROGRESS, THEN STOP DILUTION.
3. IF A PLANT TRANSIENT IS IN PROGRESS, THEN PLACE TURBINE LOAD ON "HOLD".

SUPPLEMENTARY ACTION

1. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
2. Borate the Control Bank "OUT" as necessary using the Boron Addition Nomographs. **{CMT 0008900}**
3. Refer to the Technical Specifications section on Reactivity Control.

References: A-177100, Sh. 291; U-260610; U266647 PLS Document; Technical Specifications DCP 93-1-8587; **{CMT 0008554, 0008887}**

PROMPT OPERATOR ACTION REQUIRED

LOCATION FE2

E2
CONT ROD BANK POSITION LO-LO

SETPOINT: Variable with Reactor Power as measured by ΔT and TAVG.

ORIGIN: Rod Insertion Limit Computer

PROBABLE CAUSE

> **NOTE:**
> - **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**
> - **This annunciator has REFLASH capability.**

Reactor Coolant System Boric Acid Concentration too low to ensure Reactor Protection under Accident conditions due to;

a) Plant Transient
b) Xenon Transient
c) Dilution of RCS

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. CHECK INDICATIONS AND DETERMINE THAT ACTUAL CONTROL BANK ROD POSITION IS AT THE LOW-LOW INSERTION LIMIT.
2. EMERGENCY BORATE THE REACTOR COOLANT SYSTEM IN ACCORDANCE WITH FNP-1-AOP-27.0, EMERGENCY BORATION. **{CMT 0008555, 0008900}**
3. IF A PLANT TRANSIENT IS IN PROGRESS, THEN PLACE TURBINE LOAD ON "HOLD".

SUPPLEMENTARY ACTION

1. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
2. Refer to the Technical Specifications section on Reactivity Control.

References: A-177100, Sh. 292; U-260610; U266647 PLS Document; Technical Specifications; DCP 93-1-8587; **{CMT 0008887}**

PROMPT OPERATOR ACTION REQUIRED

LOCATION FE1

E1
CONT ROD BANK POSITION LO

SETPOINT: Variable; 10 Steps Greater than LO-LO Alarm Setpoint.
$Z_{LO} = Z_{LO\text{-}LO} + K_4$
Where K_4 = 10 Steps (6.25 inches)

ORIGIN: Rod Insertion Limit Computer

PROBABLE CAUSE

NOTE:
- **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**
- **This annunciator has REFLASH capability.**

Reactor Coolant System Boric Acid Concentration too low for Reactor Power Level due to:
a) Plant Transient
b) Xenon Transient
c) Dilution of RCS

<u>AUTOMATIC ACTION</u>

NONE

<u>IMMEDIATE ACTION</u>

1. CHECK INDICATIONS AND DETERMINE THAT ACTUAL CONTROL BANK ROD POSITION IS AT LOW INSERTION LIMIT.
2. <u>IF</u> REACTOR COOLANT SYSTEM DILUTION IS IN PROGRESS, <u>THEN</u> STOP DILUTION.
3. <u>IF</u> A PLANT TRANSIENT IS IN PROGRESS, <u>THEN</u> PLACE TURBINE LOAD ON "HOLD".

SUPPLEMENTARY ACTION

1. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
2. Borate the Control Bank "OUT" as necessary using the Boron Addition Nomographs. **{CMT 0008900}**
3. Refer to the Technical Specifications section on Reactivity Control.

References: A-177100, Sh. 291; U-260610; U266647 PLS Document; Technical Specifications DCP 93-1-8587; **{CMT 0008554, 0008887}**

PROMPT OPERATOR ACTION REQUIRED

78

011FA2.13 001/1/1/LBLOCA/C/A 3.7/NEW/FA011005/S/GTO

Unit 2 has experienced a Reactor Trip and Safety Injection. Conditions are as follows:

- EEP-1.0, Loss of Reactor or Secondary Coolant, is in progress.
- The STA announces that there is a Red path on the RCS integrity CSF.
- Peak Containment pressure **was** 33 psig, but now **is** 3 psig and dropping.
- RCS Cold leg temperatures are 225°F.
- RCS Pressure is 15 psig.
- R-2, R-7, & R-11 are in alarm.
- LHSI flow is 3300 gpm.

Which one of the following procedural flowpaths is correct and the reason?

A. Go to FRP-P.1, Response to Imminent Pressurized Thermal Shock Conditions, and it **WILL** direct actions to mitigate the cooldown event.

B. Continue in EEP-1.0, and it **WILL** direct actions to mitigate the cooldown event.

C. Continue in EEP-1.0, but it will **NOT** direct actions to mitigate the cooldown event because there is no concern for PTS during this plant condition.

D. Go to FRP-P.1, but it will **NOT** direct actions to mitigate the cooldown event because there is no concern for PTS during this plant condition.

A. INCORRECT. FRP-P.1 step 1 will direct to procedure and step in effect.

B. INCORRECT. FRPs must be addressed when they are in effect. In this case, the FRP-P.1 directs to discontinue use of the mitigation strategy at step one. PTS would be a concern during this large of a cooldown if it was not caused by a large break LOCA.

C. INCORRECT. FRPs must be addressed when they are in effect even though PTS is not a concern during a LBLOCA.

D. CORRECT. **Go to FRP-P.1, but it will NOT direct actions to mitigate the cooldown event because there is no concern for PTS during this plant condition.** FRPs must be addressed when they are in effect. In this case, the FRP-P.1 directs to discontinue use of the mitigation strategy for PTS Potential at step one. PTS is not a concern during a large break LOCA.

011EA2.13 Large Break LOCA / 3 Ability to determine or interpret the following as they apply to a Large Break LOCA:(CFR 43.5 / 45.13):
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

EA2.13 – Difference between overcooling and LOCA indications

State the basis for all cautions, notes, and actions associated with FRP-P.1/2 (OPS52533K03).
Assess plant conditions and determine if transition to another section of FRP-P.1/2 or to another procedure is required (OPS52533K08).

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Check RCS pressure - GREATER THAN 265 psig(430 psig).** 2C(2A) LOOP RCS NR PRESS [] PI 402B [] PI 403B	1 <u>IF</u> LHSI flow greater than $1.5x10^3$ gpm, <u>THEN</u> return to procedure and step in effect. 2A(2B) RHR HDR FLOW [] FI 605A [] FI 605B
___ 2	**Monitor CST level.**	
	2.1 Check CST level greater than 5.3 ft. CST LVL [] LI 4132A [] LI 4132B	2.1 Align AFW pumps suction to SW using FNP-2-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.
	2.2 Align makeup to the CST from water treatment plant <u>OR</u> demin water system using FNP-2-SOP-5.0, DEMINERALIZED MAKEUP WATER SYSTEM, as necessary.	

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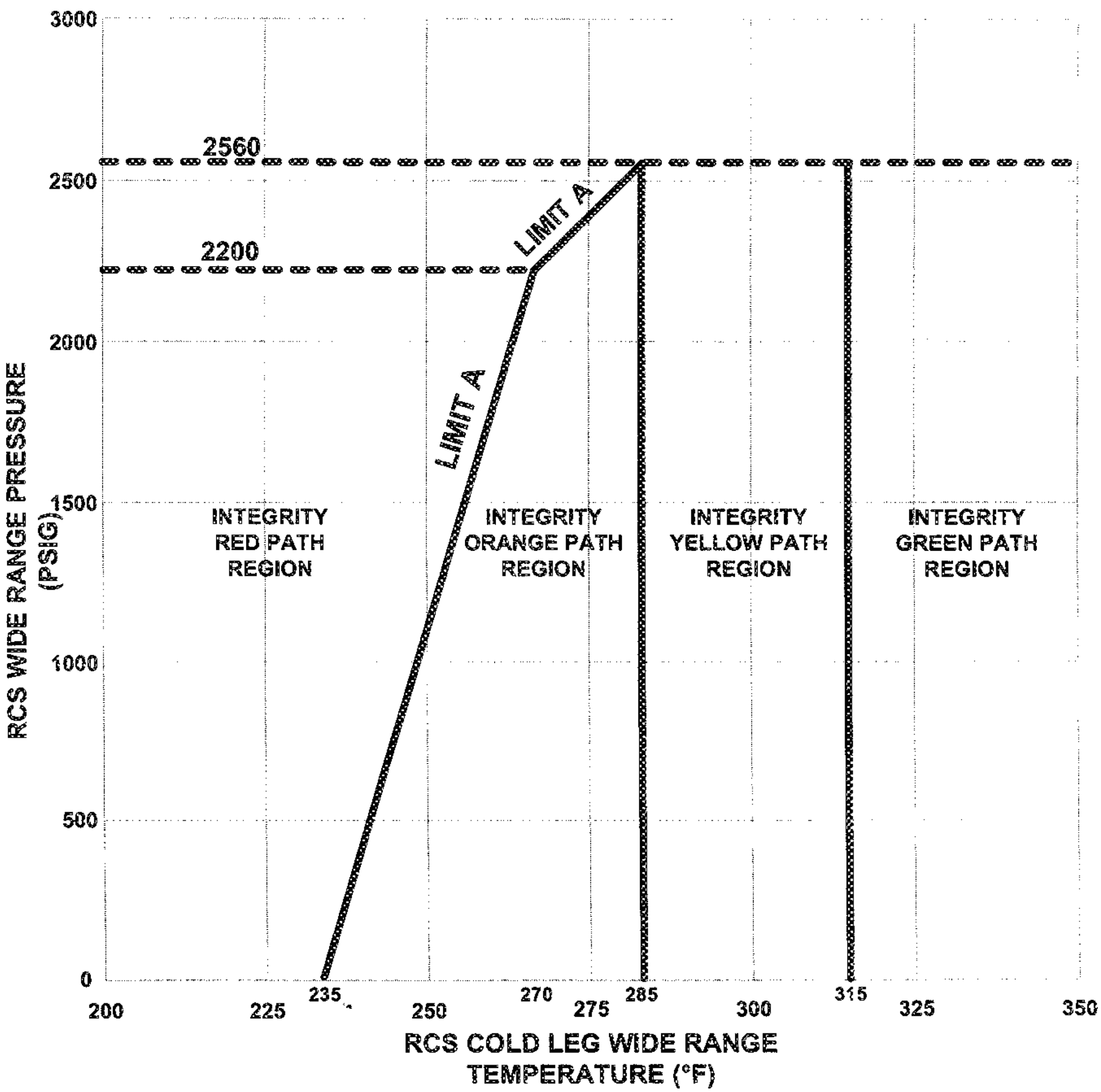
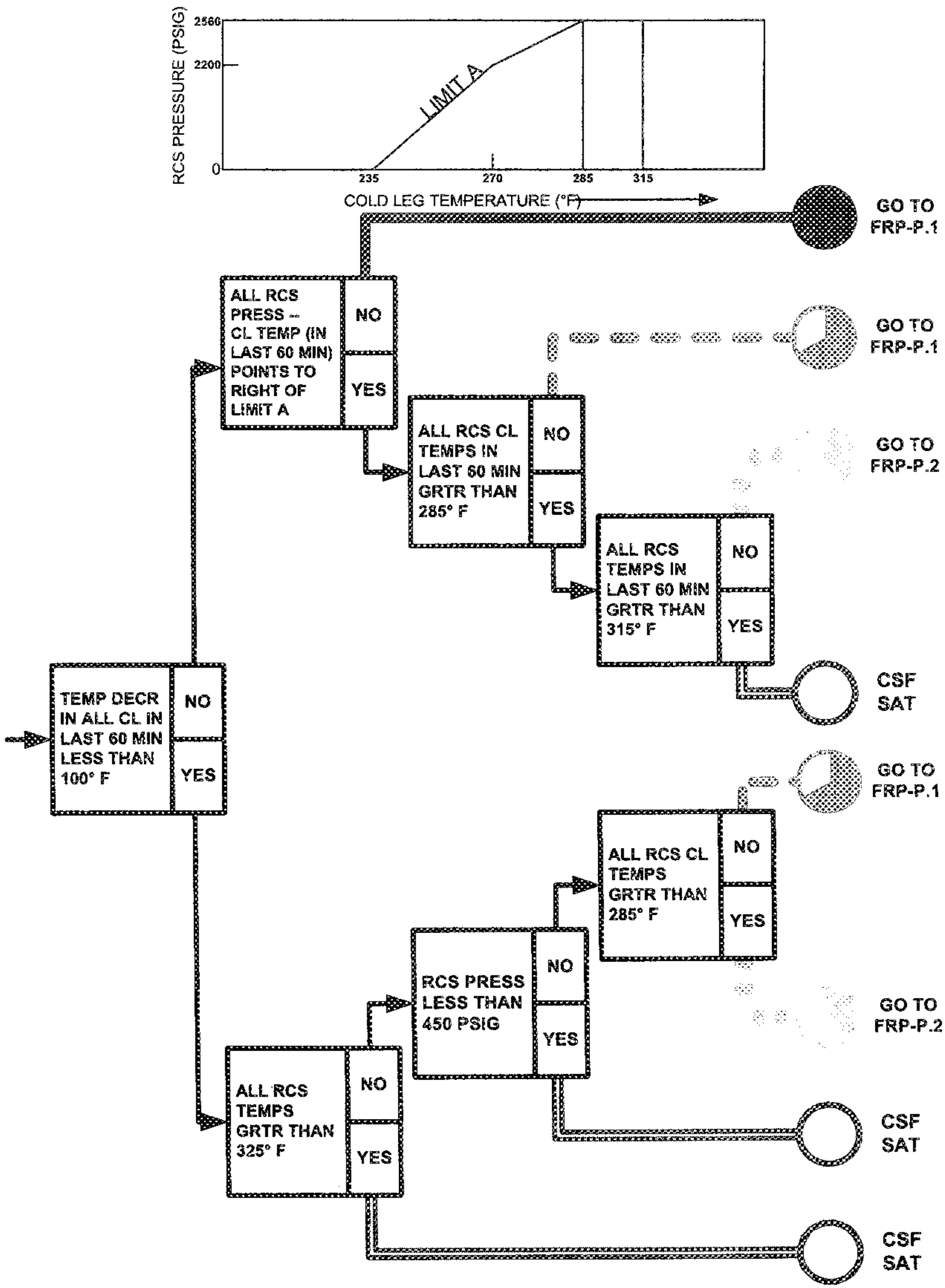
INTEGRITY

RCS PRESSURE - TEMPERATURE CRITERIA




RCS PRESSURE (PSIG)
2560
2200
0
LIMIT A
235
270
285
315
COLD LEG TEMPERATURE (°F)
TEMP DECR IN ALL CL IN LAST 60 MIN LESS THAN 100° F
NO
YES
ALL RCS PRESS -- CL TEMP (IN LAST 60 MIN) POINTS TO RIGHT OF LIMIT A
NO
YES
GO TO FRP-P.1
ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 285° F
NO
YES
GO TO FRP-P.1
ALL RCS TEMPS IN LAST 60 MIN GRTR THAN 315° F
NO
YES
GO TO FRP-P.2
CSF SAT
ALL RCS TEMPS GRTR THAN 325° F
NO
YES
RCS PRESS LESS THAN 450 PSIG
NO
YES
ALL RCS CL TEMPS GRTR THAN 285° F
NO
YES
GO TO FRP-P.1
GO TO FRP-P.2
CSF SAT
CSF SAT

would correspond to an event which is approaching the design basis assumptions. It should be noted that more probable delivery rates for the safety injection system (e.g., maximum safeguards or minimum safeguards with no spilling line) will yield less core uncovery or no core uncovery.

For break locations other than at the cold leg, little or no core uncovery is calculated. For breaks in the crossover leg there could be an uncovery similar to the uncovery experienced for the cold leg break when steam was vented through the crossover legs. Since for crossover leg breaks safety injection is injected into all RCS cold legs and steam does not have to pass through the broken loop RCP, there is no subsequent core uncovery. For breaks in the RCS hot leg or pressurizer vapor space, steam is vented earlier then for other locations (immediately for vapor space breaks) so that essentially no core uncovery is experienced.

The method used for long-term plant recovery for breaks in this category depends upon the RCS pressure at the time that the operator determines if further RCS cooldown and depressurization is required (step 13 in the E-1 guideline) . If at this time the RCS pressure is greater than the low-head SI pumps shutoff head pressure, ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, is used for long-term plant recovery. Even if the RCS pressure is less than the shutoff head pressure of the low-head SI pumps but it cannot be verified that the low-head SI pumps are injecting flow into the RCS, the operator should transfer to ES-1.2 for long-term plant recovery. The operator would stay in E-1 if the RCS pressure is less than the shutoff head pressure of the low-head SI pumps and flow into the RCS from the low-head SI pumps has been verified.

<u>Large Break LOCA, 1 FT^2 < Area ≤ Double-Ended, Minimum Safeguards</u>

A large break LOCA is the design basis for many aspects of the NSSS design. Some of the major design considerations impacted by the large break LOCA are peak core power, containment sizing, and loop/vessel internal forces. In order to describe the large break LOCA hydraulic transient a typical SAR safety analysis for a 4-loop plant will be utilized. The phenomena described here are similar for all Westinghouse plants.

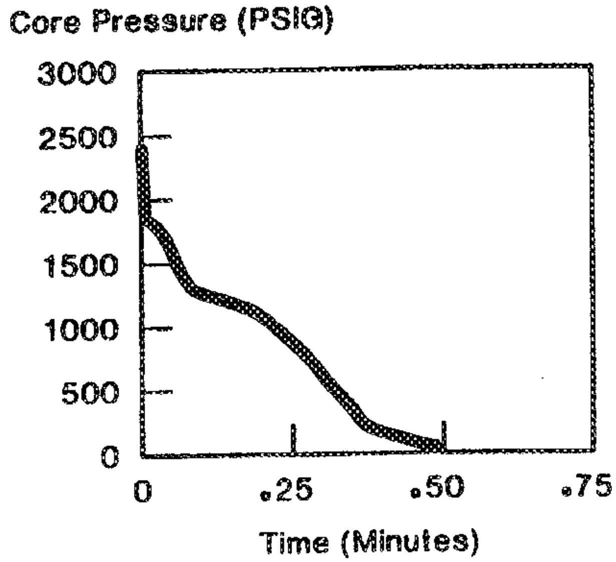
A large break LOCA (one square foot total area up to the double-ended break) has four characteristic stages: blowdown, refill, reflood, and long-term recirculation. Blowdown starts with the assumed initiation of the LOCA and ends when the reactor coolant system pressure (initially 2250 psig) falls to essentially that of the containment atmosphere. Refill starts at the end of blowdown and ends when the addition of emergency core cooling water fills the bottom of the reactor vessel and reaches the elevation of the bottom of the fuel rods. Reflood is defined as the time from the end of refill until the reactor vessel has been filled with water to the extent that core temperature rise has been terminated and core temperatures subsequently have been reduced to their long-term steady-state levels associated with dissipating decay heat. These time divisions are established mainly for analytical convenience.

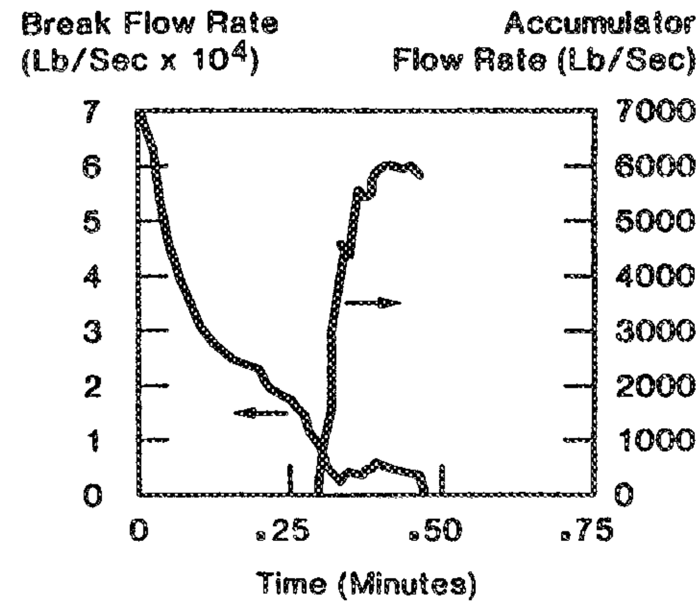
As contrasted with the large break, the blowdown phase of the small break occurs over a longer period and does not result in reduction of the effective water level in the reactor vessel below the bottom of the core. Thus, for the small break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked before the bottom of the core is uncovered (and before pressure equilibrium between reactor and containment is reached), reflood, and long-term recirculation.

Figure 11 illustrates the primary system pressure transient occurring during the blowdown phase of the LOCA for a double-ended cold leg guillotine break. The primary pressure rapidly drops from an initial value of 2250 psig to a low value of 40-50 psig by the end of blowdown (~1/2 minute). Also shown on Figure 11 is the break flow transient occurring during blowdown. The break flow starts at a very high value (critical flow, ~70,000 lbm/sec) and is reduced to zero by the end of blowdown.

Figure 11 also includes a plot of the SI accumulators mass flow rate. Note that accumulator flow is initiated approximately 16 seconds after the break occurs. This corresponds to the time when the RCS pressure has decreased to 600 psig, which corresponds to a minimum accumulator pressure set point.

Figure 11. BLOWDOWN TRANSIENTS FOR DOUBLE ENDED COLD LEG GUILLOTINE BREAK ($C_D = 0.6$)





The containment pressure transient is shown on Figure 12. As shown, the containment pressure reaches a peak value early in the transient during the blowdown phase of the transient. A safety injection signal will be initiated on a containment High-1 pressure signal in a matter of seconds after the break and containment spray may be initiated on a containment High-3 pressure signal depending on the magnitude of the break and the specific plant's containment design.

The important hydraulic transient parameters during the reflood phase are downcomer water level (ZD), core water level (ZC), and the core inlet flooding rate (VIN) as shown in Figure 12.

During refill the ECCS cooling water from the SI accumulators and safety injection pumps enters the top of the reactor vessel downcomer annulus and starts to fill the reactor vessel lower plenum, which is filled after 45 seconds. This is commonly called bottom of core (BOC) recovery time. After BOC occurs, the downcomer annulus starts to fill rapidly and thus provides a static head for pushing cooling water into the core. A core inlet flooding rate (inches/second) is established and the water starts to move up into the core, thus providing the mechanism for core cooling during reflood.

Table 1 presents a time sequence of events for the double-ended cold leg guillotine break.

After successful initial operation of the ECCS, the reactor core is once again covered with borated water. This water has enough boron concentration to maintain the core in a shutdown condition. Decay heat is removed by a continuous supply of water from the ECCS. This supply initially comes from the refueling water storage tank (RWST). When the RWST level reaches the switchover setpoint the ECCS pumps are transferred into the recirculation mode (using ES-1.3, TRANSFER TO COLD LEG RECIRCULATION) wherein water is drawn from the containment sump and is cooled in the residual heat removal heat exchangers. Thus, long-term cooling of the core is maintained by the ECCS in sump recirculation mode. The core is maintained in a shutdown state by borated water.

Figure 12. REFLOOD TRANSIENTS FOR DOUBLE ENDED COLD LEG GUILLOTINE BREAK ($C_D = 0.6$)

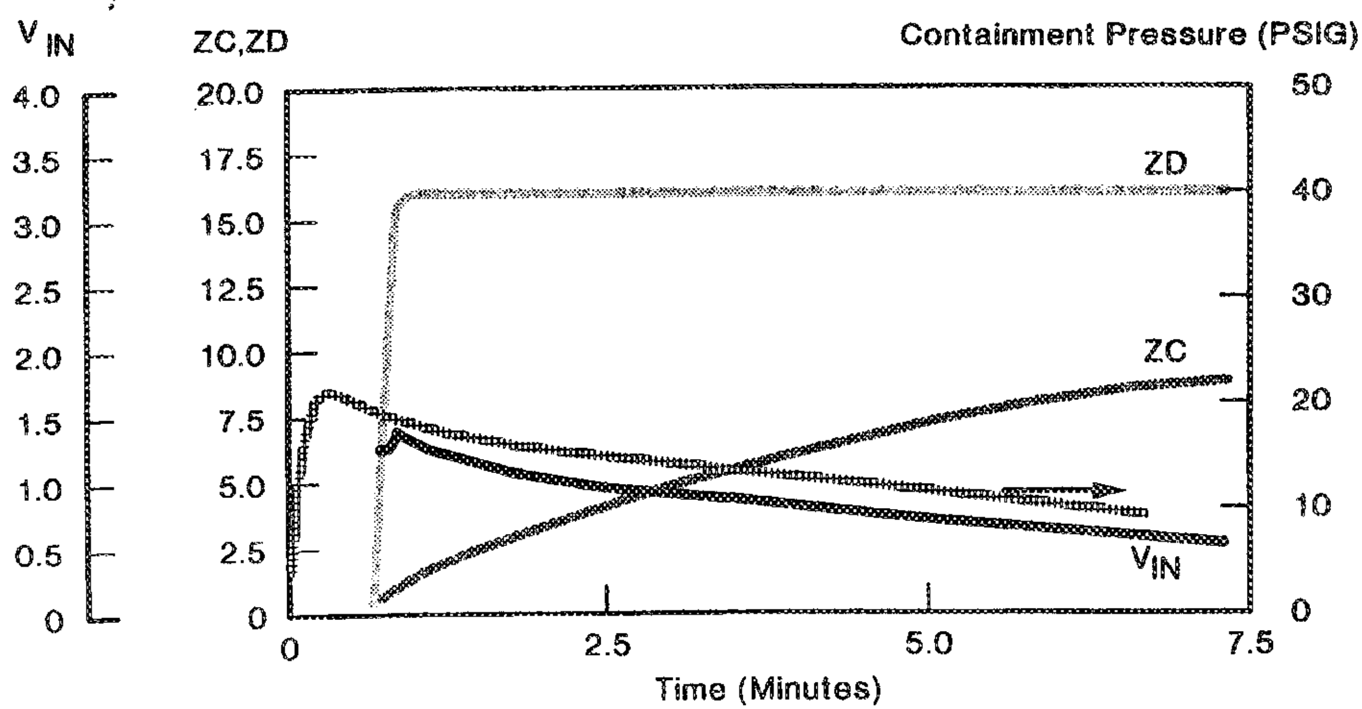


TABLE 1
TIME SEQUENCE OF EVENTS FOR LARGE BREAK*

Event	Time (Sec)
Start	0.00
Reactor Trip Signal	1.10
Safety Injection Signal	1.10
Accumulator Injection	15.8
End of Blowdown	24.9
Pump Injection	26.10
Bottom of Core Recovery	39.5
Accumulator Empty	58.8

*Double-ended cold leg guillotine break (C_D=0.6)

79. 015/17AA2.08 001/1/1/RCP/C/A 3.5/NEW/FA011005/S/CVR/GTO

Unit 2 is at 100% power when the following annunciators are received:

- DC2, RCP #1 SEAL LKOF FLOW HI
- DB5, 1B RCP #2 SEAL LKOF FLOW HI

The unit operator reports the following plant indications:

- 1B RCP #1 Seal leakoff Flow is 6.8 gpm.
- RCDT Level rise trend is 0.8 gpm.
- 1B RCP Seal injection is 7.9 gpm.
- 1B RCP Lower Seal Water Bearing temperature is 227°F and stable.

Which one of the following is the correct procedurally directed action?

A. Trip the reactor, secure 1B RCP; due to high Lower Seal Water Bearing temp.

B. Trip the reactor, secure 1B RCP; due to high combined #1 & #2 seal leakoff flowrates.

C. Commence a shutdown to be in mode 3 in 6 hours; due to the #1 seal leakoff flow rate and Lower Seal Water Bearing temperature.

D. No shutdown is required, contact the OPS Manager to obtain engineering support; due to the #1 seal leakoff flow rate.

A. CORRECT. **Trip the reactor, secure 1B RCP; due to high Lower Seal Water Bearing temp.**
ARP 1.4 DC2 Step 6 RNO of immediate action directs this due to seal water bearing temp >225°F.

B. INCORRECT. This would be true if the #1 seal leakoff was ≥ 8 gpm or the combined #1 & #2 seal leakoff flowrate was >8 gpm. In this plant condition, the given combined leak rate is assumed to be no more than RCDT level rise plus the #1 seal leakoff, or 6.8+0.8=7.6. (seal injection is not used in this calculation) DC2 steps 4 RNO & 5 RNO.

C. INCORRECT. This would be true if the seal bearing was <225 and stable, based on the #1 seal leakoff flow being >6 but <8 gpm.

D. INCORRECT. This would be true if the #1 seal leakoff was < 6 AND DB5 was NOT in alarm (DC2 step 3 RNO).

015/17AA2.08 RCP Malfunctions / 4 Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):
(CFR: 43.5 / 45.13)
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

AA2.08 – When to secure RCPs on high bearing temperatures

Evaluate abnormal plant or equipment conditions associated with the Reactor Coolant Pumps and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52101D02).

LOCATION DC2

SETPOINT: 5 GPM

C2	RCP #1 SEAL LKOF FLOW HI

ORIGIN:
1. RCP 1A #1 Seal leakoff Flow Transmitter (N1E21FT156A-N)
2. RCP 1B #1 Seal Leakoff Flow Transmitter (N1E21FT155A-N)
3. RCP 1C #1 Seal Leakoff Flow Transmitter (N1E21FT154A-N)

PROMPT OPERATOR ACTION REQUIRED

PROBABLE CAUSE

NOTE: This annunciator has REFLASH capability.

1. Loss of Injection Water followed by High Seal Temperature.

NOTE: High Temperature water will cause expansion of the seal and allow a higher flow.

2. High Temperature of the injection water supply.
3. Damage to the #1 Seal.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. DETERMINE THE CAUSE OF THE ALARM.
2. IF THE ALARM IS DUE TO INSTRUMENT FAILURE,
THEN INITIATE STEPS TO HAVE THE INSTRUMENTATION REPAIRED AND THE REMAINDER OF THIS PROCEDURE IS N/A.

LOCATION DC2

IMMEDIATE ACTION

NOTE:
- Steps 3, 4, 5 and 6 are continuing action steps.
- DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

A/ER	RNO
3. CHECK #1 SEAL LEAKOFF - GREATER THAN 6 GPM.	3. IF #2 SEAL LKOF HI ANNUNCIATOR (DA5, DB5 or DC5) FOR AFFECTED RCP IN ALARM, THEN PROCEED TO STEP 4, IF NOT PROCEED TO THE SUPPLEMENTARY ACTION SECTION.
4. CHECK #1 SEAL LEAKOFF - LESS THAN 8 GPM.	4. PERFORM THE FOLLOWING: 4.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 4.2 SECURE THE AFFECTED RCP. {CMT 0003908} 4.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 4.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 4.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

IMMEDIATE ACTION

A/ER	RNO
5. CHECK #2 SEAL LKOF HI ANNUNCIATOR (DA5, DB5 or DC5) FOR AFFECTED RCP -CLEAR	5. PERFORM THE FOLLOWING: 5.1 IF #1 SEAL LEAKOFF GREATER THAN 7.25 GPM, THEN PERFORM THE FOLLOWING, IF NOT PROCEED TO RNO STEP 5.2. 5.1.1 MANUALLY TRIP THE REACTOR 5.1.2 SECURE THE AFFECTED RCP. 5.1.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 5.1.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 5.1.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION. 5.2 MONITOR RCDT LEVEL INCREASE TO APPROXIMATE #2 SEAL LEAKOFF RATE, WHILE CONTINUING WITH THIS PROCEDURE.

IMMEDIATE ACTION

A/ER	RNO
	5.3 IF THE COMBINATION OF #1 SEAL LEAKOFF AND #2 SEAL LEAKOFF GREATER THAN 8 GPM, THEN PERFORM THE FOLLOWING, IF NOT PROCEED TO STEP 6. 5.3.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 5.3.2 SECURE THE AFFECTED RCP. {CMT 0003908} 5.3.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 5.3.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 5.3.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

NOTE: REFER TO RCS & RHR GROUP DISPLAY ON PLANT COMPUTER FOR RCP SEAL WATER TEMPERATURES.

6. CHECK RCP LOWER SEAL WATER BEARING AND SEAL WATER OUTLET TEMPERATURES -STABLE AND LESS THAN 225° F.	6. PERFORM THE FOLLOWING: 6.1. MANUALLY TRIP THE REACTOR. {CMT 0003908} 6.2 SECURE THE AFFECTED RCP. {CMT 0003908}

LOCATION DC2

IMMEDIATE ACTION

A/ER	RNO
	6.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 6.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 6.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
7. PERFORM A CONTROLLED SHUTDOWN AS FOLLOWS: 7.1 PERFORM A CONTROLLED SHUTDOWN TO HAVE THE RCP SECURED AND SEAL LEAKOFF ISOLATED WITHIN EIGHT HOURS. 7.2 MAINTAIN > 9 GPM SEAL INJECTION FLOW TO THE AFFECTED RCP WHILE THE PUMP IS RUNNING. 7.3 WHEN THE REACTOR IS SHUTDOWN, THEN SECURE THE AFFECTED RCP. 7.4 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP.	

IMMEDIATE ACTION

A/ER	RNO
7.5 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP).	

SUPPLEMENTARY ACTION

1. IF #1 Seal leakoff less than 6 gpm, AND #2 SEAL LKOF HI ANNUNCIATOR (DA5, DB5 or DC5) for affected RCP is clear, AND affected RCP lower seal water bearing and seal water outlet temperatures are stable,
   THEN perform the following:

   1.1 Monitor #1 Seal leakoff flow for the affected RCP.

   1.2 Minimize CCW temperature transients which will affect seal injection temperature.

   1.3 Minimize VCT pressure transients which will affect #1 Seal leakoff backpressure.

   1.4 Minimize evolutions which could affect seal injection such as shifting charging pumps.

   1.5 Contact OPS Manager to initiate engineering and/or vendor support.

NOTE: Consideration should be given to using the seal injection throttle valves (Q1E21V116A, B and C) instead of HIK 186 for finer control of temperature in the following steps.

2. IF the High Flow Alarm was caused by a loss of injection water followed by high seal temperature AND the RCP lower seal water bearing temperature is below 225°F, THEN carefully Re-establish injection water flow, reducing the bearing temperature at a maximum rate of 1°F per minute using SEAL WTR INJECTION HIK 186.

3. IF the High Flow Alarm was caused by high temperature of the injection water supply AND the RCP lower seal water bearing temperature is below 225°F, THEN slowly cool the Injection Water Supply while reducing the bearing temperature at a maximum rate of 1°F per minute .

4. IF the affected pump removed from service THEN refer to FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW.

SUPPLEMENTARY ACTION

5. Refer to Technical Specifications, LCO 3.5.5 condition A, for LCO requirements.

6. DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

References: A-177100, Sh. 192; D-175039, Sh. 1; U-175986; U-176032; U258242; PLS Document; Technical Specifications; Westinghouse letter ALA-88-811; Westinghouse Tech Bulletin ESBU-TB-93-01-R1; PCN B93-1-8652

LOCATION DA5

SETPOINT: 0.75 GPM

ORIGIN: RCP 1A #2 Seal Leakoff Flow Switch (N1E21FSH160-N)

A5	1A RCP #2 SEAL LKOF FLOW HI

PROBABLE CAUSE

1. #2 Seal hanging open.
2. Damage to #2 Seal.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

NOTE: DO NOT RESTART THE AFFECTED RCP UNTIL THE CAUSE OF THE SEAL MALFUNCTION HAS BEEN DETERMINED AND CORRECTED.

A/ER	RNO
1 IF ANNUNCIATOR DC2, RCP #1 SEAL LKOF FLOW HI IS IN ALARM, THEN REFER TO ANNUNCIATOR DC2 RESPONSE PROCEDURE FOR GUIDANCE.	
2. ENSURE PROPER SETTING OF SEAL WTR INJECTION HIK 186.	

NOTE:
- **The following step determines total flow through the # 1 seal (sum of the #1 and #2 leakoffs) to ensure degradation of #1 seal is not being masked by a simultaneous failure of #2 seal.**
- **Trending of RCDT level increase over time while only an approximation, is the only available method of determining #2 seal leakoff flow for the calculation below.**

3. CHECK COMBINATION OF #1 SEAL AND #2 SEAL LEAKOFF FLOWS - LESS THAN 8 GPM.	3. PERFORM THE FOLLOWING: 3.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 3.2 SECURE THE AFFECTED RCP.

IMMEDIATE ACTION

A/ER	RNO
	3.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 3.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 3.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

SUPPLEMENTARY ACTION

1. IF the high flow alarm is caused by excessive leakage of the #2 Seal AND the #2 Seal is considered to be hung open AND NOT damaged, THEN continue pump operation. A short period (up to 24 hours) of pump operation may be required before the #2 Seal seats and operates normally.

2. IF the high flow alarm is caused by damage to the #2 Seal AND there is NOT abnormal vibration AND the reactor coolant drain tank is capable of handling the increased leak rate, THEN continue pump operation. The seal should be replaced as soon as possible.

3. IF the affected pump removed from service, THEN refer to FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW.

4. Refer to Technical Specifications, LCO 3.5.5 condition A for LCO requirements.

5. DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

References: A-177100, Sh. 185; A-181541; D-175039, Sh. 1; U-258242; PLS Document; Technical Specifications; Westinghouse Tech Bulletin ESBU-TB-93-01-R1

LOCATION DB5

SETPOINT: 0.75 GPM

ORIGIN: RCP 1B #2 Seal leakoff Flow Switch (N1E21FSH159-N)

B5	1B RCP #2 SEAL LKOF FLOW HI

PROBABLE CAUSE

1. #2 Seal Hanging open.
2. Damage to #2 Seal.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

NOTE: DO NOT RESTART THE AFFECTED RCP UNTIL THE CAUSE OF THE SEAL MALFUNCTION HAS BEEN DETERMINED AND CORRECTED.

A/ER	RNO
1 IF ANNUNCIATOR DC2, RCP #1 SEAL LKOF FLOW HI IS IN ALARM, THEN REFER TO ANNUNCIATOR DC2 RESPONSE PROCEDURE FOR GUIDANCE.	
2. ENSURE PROPER SETTING OF SEAL WTR INJECTION HIK 186.	

NOTE:
- **The following step determines total flow through the # 1 seal (sum of the #1 and #2 leakoffs) to ensure degradation of #1 seal is not being masked by a simultaneous failure of #2 seal.**
- **Trending of RCDT level increase over time while only an approximation, is the only available method of determining #2 seal leakoff flow for the calculation below.**

3. CHECK COMBINATION OF #1 SEAL AND #2 SEAL LEAKOFF FLOWS - LESS THAN 8 GPM.	3. PERFORM THE FOLLOWING: 3.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 3.2 SECURE THE AFFECTED RCP.

LOCATION DB5

IMMEDIATE ACTION

A/ER	RNO
	3.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 3.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 3.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

SUPPLEMENTARY ACTION

1. IF the high flow alarm is caused by excessive leakage of the #2 Seal AND the #2 Seal is considered to be hung open AND NOT damaged, THEN continue pump operation. A short period (up to 24 hours) of pump operation may be required before the #2 seal seats and operates normally.

2. IF the high flow alarm is caused by damage to the #2 Seal AND there is NOT abnormal vibration AND the reactor coolant drain tank is capable of handling the increased leak rate, THEN continue pump operation. The seal should be replaced as soon as possible.

3. IF the affected pump removed from service, THEN refer to FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW.

4. Refer to Technical Specifications, LCO 3.5.5 condition A, for LCO requirements.

5. DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

References: A-177100, Sh. 190; A-181541; D-175039, Sh. 1; U-258242; PLS Document; Technical Specifications; Westinghouse Tech Bulletin ESBU-TB-93-01-R1

LOCATION DC1

SETPOINT: 0.95 GPM

C1	RCP #1 SEAL LKOF FLOW LO

ORIGIN:
1. RCP 1A #1 Seal Leakoff Flow Transmitter (N1E21FT156B-N)
2. RCP 1B #1 Seal Leakoff Flow Transmitter (N1E21FT155B-N)
3. RCP 1C #1 Seal Leakoff Flow Transmitter (N1E21FT154B-N)

PROBABLE CAUSE

NOTE: This annunciator has REFLASH capability.

1. A differential pressure of less than 200 PSID across the #1 Seal.
2. Excessive leakage of the #2 Seal.
3. Damage to the #1 Seal.
4. Volume Control Tank pressure higher than normal.
5. Improper setting of SEAL WTR INJECTION HIK 186.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

CAUTION: DO NOT RAISE SEAL INJECTION FLOW IF ANNUNCIATOR DC4 IS IN ALARM.

NOTE: DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

A/ER	RNO
1. ENSURE PROPER SETTING OF SEAL WTR INJECTION HIK 186.	
2. IF THE REACTOR IS SHUTDOWN, THEN PROCEED TO 8.	

IMMEDIATE ACTION

NOTE: REFER TO RCS & RHR GROUP DISPLAY ON PLANT COMPUTER FOR RCP SEAL WATER TEMPERATURES.

A/ER	RNO
3. CHECK RCP LOWER SEAL WATER BEARING AND SEAL WATER OUTLET TEMPERATURES -STABLE.	3. PERFORM THE FOLLOWING: 3.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 3.2 SECURE THE AFFECTED RCP. {CMT 0003908} 3.3 <u>WHEN</u> RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, <u>THEN</u> CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 3.4 <u>IF</u> 1A <u>OR</u> 1B RCP IS SECURED, <u>THEN</u> CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 3.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
4. CHECK FOR FAILURE OF #2 SEAL.	

IMMEDIATE ACTION

NOTE:
- THE FOLLOWING STEP DETERMINES TOTAL FLOW THROUGH THE # 1 SEAL TO ENSURE DEGRADATION OF #1 SEAL IS NOT BEING MASKED BY A SIMULTANEOUS FAILURE OF #2 SEAL.
- IF THE RCP #2 SEAL LEAKOFF FLOW HIGH ANNUNCIATOR IS IN ALARM, THEN ADD TOGETHER INDICATED #1 SEAL LEAKOFF AND ESTIMATED #2 SEAL LEAKOFF TO OBTAIN TOTAL #1 SEAL FLOW, OTHERWISE USE INDICATED #1 SEAL LEAKOFF AS TOTAL #1 SEAL FLOW THE STEP BELOW.
- IF THE RCP #2 SEAL LEAKOFF FLOW HIGH ANNUNCIATOR IS IN ALARM, THEN TRENDING OF RCDT LEVEL INCREASE OVER TIME WHILE ONLY AN APPROXIMATION, IS THE ONLY AVAILABLE METHOD OF ESTIMATING #2 SEAL LEAKOFF FLOW FOR THE STEP BELOW.

A/ER	RNO
4.1 CHECK TOTAL #1 SEAL FLOW- LESS THAN 8 GPM.	4.1 PERFORM THE FOLLOWING: 4.1.1 MANUALLY TRIP THE REACTOR. {CMT 0003908} 4.1.2 SECURE THE AFFECTED RCP {CMT 0003908} 4.1.3 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 4.1.4 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 4.1.5 GO TO FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

IMMEDIATE ACTION

A/ER	RNO
5. CHECK #1 SEAL LEAKOFF FLOW - LESS THAN 0.8 GPM	5. THEN PERFORM THE FOLLOWING: 5.1 CONTINUE MONITORING PUMP PARAMETERS FOR FURTHER DEGRADATION. 5.2 CONTACT WESTINGHOUSE FOR FURTHER GUIDANCE ON CONTINUED PUMP OPERATION. 5.3 PROCEED TO SUPPLEMENTARY ACTIONS SECTION.
6. CHECK #2 SEAL LEAKOFF FLOW HI ANNUNCIATOR FOR AFFECTED PUMP (DA5, DB5 OR DC5) - <u>NOT</u> IN ALARM	6. THEN PERFORM THE FOLLOWING: 6.1 CONTINUE MONITORING PUMP PARAMETERS FOR FURTHER DEGRADATION. 6.2 CONTACT WESTINGHOUSE FOR FURTHER GUIDANCE ON CONTINUED PUMP OPERATION. 6.3 PROCEED TO SUPPLEMENTARY ACTIONS SECTION.
7. PERFORM A CONTROLLED SHUTDOWN AS FOLLOWS: 7.1 PERFORM A CONTROLLED SHUTDOWN TO HAVE THE RCP SECURED AND SEAL LEAKOFF ISOLATED WITHIN EIGHT HOURS. 7.2 MAINTAIN > 9 GPM SEAL INJECTION FLOW TO THE AFFECTED RCP WHILE THE PUMP IS RUNNING. 7.3 <u>WHEN</u> THE REACTOR IS SHUTDOWN, <u>THEN</u> SECURE THE AFFECTED RCP.	

IMMEDIATE ACTION

A/ER	RNO
7.4 WHEN RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, THEN CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 7.5 IF 1A OR 1B RCP IS SECURED, THEN CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP). 7.6 PROCEED TO SUPPLEMENTARY ACTION SECTION.	
8. PERFORM THE FOLLOWING: 8.1 IF THE LOW FLOW ALARM IS CAUSED BY LOW DIFFERENTIAL PRESSURE, THEN STOP THE PUMP AND INCREASE REACTOR COOLANT SYSTEM PRESSURE TO GREATER THAN 325 PSIG.	

LOCATION DC1

IMMEDIATE ACTION

A/ER	RNO
8.2 CHECK RCP LOWER SEAL WATER BEARING AND SEAL WATER OUTLET TEMPERATURES -STABLE.	8.2 PERFORM THE FOLLOWING. 8.2.1 SECURE THE AFFECTED RCP 8.2.2 <u>WHEN</u> RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, <u>THEN</u> CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 8.2.3 <u>IF</u> 1A <u>OR</u> 1B RCP IS SECURED, <u>THEN</u> CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP).
8.3 VERIFY #1 SEAL LEAKOFF FLOW WITHIN THE NORMAL OPERATING RANGE PROVIDED IN FIGURE 1 OF FNP-1-SOP-1.1.	8.3 PERFORM THE FOLLOWING. 8.3.1 SECURE THE AFFECTED RCP 8.3.2 <u>WHEN</u> RCP HAS COME TO A COMPLETE STOP AS INDICATED BY MINIMUM RCS FLOW IN THE AFFECTED LOOP, <u>THEN</u> CLOSE THE 1A(B OR C) RCP SEAL LEAKOFF Q1E21HV8141A(B OR C) FOR THE AFFECTED RCP. 8.3.3 <u>IF</u> 1A <u>OR</u> 1B RCP IS SECURED, <u>THEN</u> CLOSE THE PRESSURIZER SPRAY VALVE FOR THE AFFECTED RCP (PK 444C FOR 1A RCP, PK 444D FOR 1B RCP).

LOCATION DC1

IMMEDIATE ACTION

A/ER	RNO
8.4 IF THE LOW FLOW ALARM IS CAUSED BY EXCESSIVE LEAKAGE OF THE #2 SEAL AND THE #2 SEAL IS CONSIDERED TO BE HUNG OPEN AND NOT DAMAGED, THEN CONTINUE PUMP OPERATION. A SHORT PERIOD (UP TO 24 HOURS) OF PUMP OPERATION MAY BE REQUIRED BEFORE THE #2 SEAL SEATS AND OPERATES NORMALLY.	

SUPPLEMENTARY ACTION

1. IF the affected pump removed from service, THEN refer to FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW.

2. IF the low flow alarm is caused by high VCT pressure, THEN adjust VCT pressure to the minimum required pressure.

3. DO NOT restart the affected RCP until the cause of the seal malfunction has been determined and corrected.

4. Refer to Technical Specifications, LCO 3.5.5 condition A, for LCO requirements.

References: A-177100, Sh. 191; D-175039, Sh. 1; U-175986; U-176002; U-176032; U-258242; PLS Document; Technical Specifications; Westinghouse Tech Bulletin ESBU-TB-93-01-R1

80. 025G2.1.32 001/1/1/RHR TS/C/A 3.8/NEW/FA011005/S/CVR

Unit 1 is in Mode 6 with the refueling cavity at 153' 6". 1A RHR pump is tagged out. 1B RHR pump is running and OPERABLE when the 1B RHR pump trips.
Which one of the following provides protection to the core in this condition?

A. The additional coolant inventory provides backup cooling due to decay heat removal by natural convection.

B. The additional coolant inventory provides more total boron available to keep the core subcritical in the event of boiling.

C. The OPERABLE train of RHR does not need backup cooling due to the lower decay heat load during fuel removal from the core.

D. The additional coolant inventory provides a sufficient volume of water to mitigate an inadvertent dilution accident if the mixing from the operable train is lost.

Reference TS 3.9.4 & bases

A. CORRECT.**The additional coolant inventory provides backup cooling due to decay heat removal by natural convection.**
per TS BASIS 3.9.4

B. INCORRECT. There is more total boron above the core, but boron is not removed from the inventory during boiling - it is left behind. The core inventory will have an adequate boron concentration at low water levels or at high, with boiling or without.

C. INCORRECT. For a period of time in mode 6, prior to moving any fuel, a full core of used fuel produces max decay heat allowed during mode 6.

D. INCORRECT. Even though one of the purposes of RHR circulation is to provide mixing to prevent boron stratification, the coolant in the core could become much lower in boron concentration than the refueling cavity during an inadvertent dilution with a loss of RHR due to the lack of forced circulation between the core and the cavity.

025G2.1.32 Loss of RHR System / 4 Ability to explain and apply all system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)
(2) Facility operating limitations in the technical specifications and their bases.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Residual Heat Removal System (OPS52101K01).
• 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation - High Water Level
• 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation — High Water Level

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

---NOTE---
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration. AND	Immediately
	A.2 Suspend loading irradiated fuel assemblies in the core. AND	Immediately
	A.3 Initiate action to satisfy RHR loop requirements. AND	Immediately
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close equipment hatch and secure with four bolts.	4 hours
	AND		
	A.5	Close one door in each air lock.	4 hours
	AND		
	A.6.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR		
	A.6.2	Verify each penetration is capable of being closed by an OPERABLE Containment Purge and exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	12 hours

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation — High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

The RHR and Coolant Circulation — High Water Level specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to not be in operation for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level."

BASES

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with a lower boron concentration than the required boron concentration specified in the COLR. Therefore, actions that could result in the addition of water to the RCS with a boron concentration less than the required boron concentration specified in the COLR must be suspended immediately.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If no RHR is in operation, the following actions must be taken:

a) the equipment hatch must be closed and secured with four bolts;
b) one door in each air lock must be closed; and
c) each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

(continued)

BASES

ACTIONS

A.4, A.5, A.6.1, and A.6.2 (continued)

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. FSAR, Section 5.5.7.

81. 026A2.08 001/2/1/CONTAINMENT SPRAY/C/A 3.7/BANK-2004/FA011005/S/GTO

Given the following conditions on Unit 2:

- The plant was at 100% power when the 2A S/G Main Steam line ruptured inside containment.
- All systems actuated as per design.
- Containment pressure spiked to 33 psig and is now continuing to decrease slowly.
- The crew has entered ESP-1.1, SI Termination.

Which one of the following provides the correct procedure which will align the Containment Spray (CS) system and the plant parameters which allow for securing the CS system?

A. ESP-1.1, SI Termination. Containment pressure is 12 psig and the RWST level is 10.5 feet and continuing to decrease.

B✓ ESP-1.3, Transfer to Cold Leg Recirc. Containment pressure is 15 psig and CS has been aligned for recirculation flow for 10 hours.

C. ESP-1.3, Transfer to Cold Leg Recirc. Containment pressure is 12 psig and CS has been aligned for recirculation flow for 7.5 hours.

D. ESP-1.1, SI Termination. Containment pressure is 15 psig, RWST level is <5.5 feet and it has been 8 hours since the initiation of the event.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

The requirement to secure CS is found in EEP-1, ESP-1.3 AND ESP-1.1 (WHEN, THEN statements). It is defined as "WHEN CS recirculation flow has been aligned for at least 8 hours and ctmt pressure is less than 16 psig THEN stop both CS pumps."

A - Incorrect, ESP-1.1 does provide guidance, but ESP-1.3 would be in progress at 4.5 ft in the RWST to commence placing the CS on recirc per the foldout page. In ESP-1.3, at 4.5 ft, IF transfer to recirc is not imminent, THEN the CS pumps are secured. This guidance is not in ESP-1.1. It takes about 1.5 hours to lower the RWST to 4.5 ft with both spray pumps running with full flow: 2 x 2600 gpm/(60min/hr)= ~1.5hrs. This is when the CS is initially placed on recirc. The requirement of 8 hours on recirc could not have been met.

B - Correct, **ESP-1.3, TRANSFER TO COLD LEG RECIRC. CS has been aligned for recirculation flow for 10 hours and containment pressure is 15 psig.**
ESP-1.3, does provide guidance; Containment pressure is <16# and the time on recirc is > 8 hours.

C - Incorrect, ESP-1.3, does provide guidance. The 7.5 hours of operation applys to HHSI/LHSI transferring from Cold Leg recirc to Simultaneous hot/cold leg recirc, but is not long enough to meet the 8 hour minimum for operation of CS on recirc prior to securing CS.

D - Incorrect, ESP-1.1 does provide guidance, but 8 hours from the initiation of the event is not long enough to have 8 hours of CS recirc time elapsed.

026 Containment Spray
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
SRO –A2.08 – Safe securing of containment spray (when it can be done).

2. Evaluate abnormal plant or equipment conditions associated with the Containment Spray and Cooling System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS50102C02).

Bank from 2004 NRC exam HLT 28A (NOT ON HLT 29 AUDIT)

Step	Action/Expected Response	Response NOT Obtained
___ 9	**WHEN RWST level less than 4.5 ft, THEN align containment spray for cold leg recirculation.**	
	9.1 Reset PHASE B CTMT ISO. [] MLB-3 1-1 not lit [] MLB-3 6-1 not lit	
	9.2 Open containment spray pump containment sump suction isolation valves. CTMT SUMP TO 1A(1B) CS PUMP [] Q1E13MOV8826A [] Q1E13MOV8827A [] Q1E13MOV8826B [] Q1E13MOV8827B	
	9.3 Close containment spray pump RWST suction isolation valves. RWST TO 1A(1B) CS PUMP [] Q1E13MOV8817A [] Q1E13MOV8817B	
	9.4 WHEN containment spray recirculation flow has been established for at least 8 hours, AND containment pressure is less than 16 psig, THEN stop both CS PUMPs.	
	9.5 Consult TSC staff to evaluate RWST makeup requirements.	
___10	**Go to procedure and step in effect.**	

-END-

82. 026AA2.02 001/1/1/CCW/CA 3.6/MOD/FA011005/S/

Unit 1 is at 100% steady-state power with the following lineup:

- 1C CCW pump is running through the 1C CCW Hx, supplying the miscellaneous header.
- 1B CCW pump is out of service for maintenance.

The following alarms come in:

- AA4- CCW SRG TK LVL A TRN HI-LO
- AB4- CCW SRG TK LVL B TRN HI-LO
- AA5- CCW SRG TK LVL A TRN LO-LO

CCW surge tank level stabilizes at 20 inches. Which one of the following describes the possible location of the leak and the correct operator response IAW AOP-9.0, Loss of Component Cooling Water?

A. Letdown heat exchanger. Initiate makeup to the CCW surge tank; evaluate tripping the reactor and RCPs due to imminent loss of CCW.

B✓ One of the sample coolers. Initiate makeup to the CCW surge tank; do not cross-connect CCW trains or transfer the miscellaneous header to the other train.

C. Seal water return heat exchanger. Trip the reactor and RCPs; transfer the miscellaneous header to the opposite train to allow immediate restarting of the RCPs.

D. RCP motor oil cooler. Initiate makeup to the CCW surge tank; transfer the miscellaneous header to the opposite train to allow continued operation while trying to isolate the leak.

References: AOP-9.0

A. INCORRECT. This leak would cause a rising ST level and cross connecting the trains with the leak still in would not be correct.

B. CORRECT. **One of the Sample coolers. Initiate makeup to the CCW surge tank; do not cross-connect CCW trains or transfer the miscellaneous header to the other train.**
The sample coolers is the only choice that will isolate automatically - HV2229 is closed on lo lo level. The proper actions iaw AOP-9 is to fill the CCW ST and continue operating.

C. INCORRECT. This would give these indications with the exception of isolating the leak automatically. If the leak did not stop, these would be the correct actions.

D. INCORRECT. This would not auto isolate. This would be the actions if the leak were not on the misc header.

026AA2.02 Loss of Component Cooling Water / 8 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: (CFR: 43.5 / 45.13) The cause of possible CCW loss

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. Evaluate abnormal plant or equipment conditions associated with the Component Cooling Water System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102G02).

AOP-9.0-52520I06 #17

original ques ?

AOP-9.0-52520I06 017/HLT/LOCT//C/A (LEVEL 2/3) PROC/APE026AA2.03///LOCT/

Evaluate the following plant conditions that apply to the question that follows:

- The plant is at 100% steady-state power with all systems operating properly.
- C CCW pump is running through the C CCW Hx, supplying the miscellaneous header.
- B CCW pump is out of service for maintenance.
- A leak in the miscellaneous header causes level in the on-service train surge tank to drop to the Lo-Lo level alarm; all automatic actions occur.
- CCW surge tank level stops decreasing and stabilizes at 15 inches.

Which of the following is a correct operator response?

A. Trip the reactor and RCPs; transfer the miscellaneous header to the opposite train to allow immediate restarting of the RCPs.

B. Cross-connect A train and B train to allow continued plant operation while trying to isolate the leak.

C. Initiate makeup to the CCW surge tank; do not cross-connect CCW trains or transfer the miscellaneous header to the other train.

D. Transfer the miscellaneous header to the opposite train to allow continued operation while trying to isolate the leak.

References: AOP-9.0

LOCATION AA4

A4
CCW SRG TK LVL A TRN HI-LO

SETPOINT: 1. HI: 50 ± 0.3 inches
2. LO: 33 ± 0.3 inches

ORIGIN: 1. Level Switch (N1P17LSH3027A-A).
2. Level Switch (N1P17LSL3027A-A).

PROBABLE CAUSE

1. HI - In Leakage of Reactor Coolant, Service Water, or through a Makeup Water Valve
   - Letdown heat exchanger (if letdown on service)
   - RCP thermal barriers
   - RHR heat exchanger (if on service)
   - Reactor makeup system (if normally closed, valves leaking by)
   - Demineralized water system (if normally closed, valves leaking by)
   - SW system (if SW discharge pressure higher than CCW discharge pressure)
   - RCDT heat exchanger (if at least one RCDT pump running

2. LO - Rupture or Leakage of A Train CCW components or piping:
   - Spent fuel pool heat exchanger
   - Charging pump oil coolers
   - RHR heat exchanger (if normally closed, MOV 3185 A or B are open)
   - RCP oil coolers
   - Excess letdown heat exchanger (if excess letdown secured)
   - Sample coolers (if sampling not in progress)
   - Seal water heat exchanger
   - Evaporator packages
   - Hydrogen recombiner
   - Waste gas compressors
   - Floor drain tank via CCW relief valves
   - SW system if CCW discharge pressure is higher than SW discharge pressure
   - Primary and secondary sample coolers (if sampling in progress)
   - GFFD sampling assembly

LOCATION AA4

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

CHECK A TRAIN CCW SURGE TANK LEVEL INDICATION AND DETERMINE WHETHER LEVEL IS HI OR LO.

HI LEVEL - DETERMINE SOURCE OF IN LEAKAGE AND ISOLATE IF POSSIBLE.

LO LEVEL -
1. DETERMINE SOURCE OF OUT LEAKAGE AND ISOLATE IF POSSIBLE.
2. ATTEMPT TO FILL CCW SURGE TANK PER FNP-1-SOP-23.0, COMPONENT COOLING WATER SYSTEM, TO MAINTAIN LEVEL ABOVE THE LO LEVEL ALARM POINT.

SUPPLEMENTARY ACTION

1. Refer to FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.
2. Refer to Technical Specification 3.7.7 for LCO requirements with a loss of the on service train of component cooling water.
3. Notify appropriate personnel to locate and correct the cause of the HI-LO level.
4. IF CCW Surge Tank level is high, THEN perform the following:
   - 4.1 Check Radiation Monitors R-17A and R-17B for increasing count rates.
   - 4.2 Verify the following make up valves are closed.
     MKUP TO CCW FROM DW STOR TK Q1P17MOV3030A
     MKUP TO CCW FROM DW STOR TK Q1P17MOV3030B
     MKUP TO CCW FROM RMW Q1P17MOV3031A
     MKUP TO CCW FROM RMW Q1P17MOV3031B
   - 4.3 IF desired, THEN close CCW SRG TK DEMIN INLET ISO N1P11V045.
   - 4.4 Commence lowering the CCW Surge Tank level per FNP-1-SOP-23.0, COMPONENT COOLING WATER SYSTEM.
   - 4.5 IF CCW Surge Tank level raise is due to RCS leakage, THEN refer to FNP-1-AOP-1.0, RCS LEAKAGE.
   - 4.6 IF CCW Surge Tank level raise is due to Letdown Heat Exchanger leakage, THEN isolate letdown per FNP-1-SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION AND refer to FNP-1-AOP-16.0, LOSS OF LETDOWN.

References: A-177100, Sh. 54; B-175810, Sh. 101; B-175968; D-175002, Sh. 1; D-177183; Technical Specification 3.7.7; NEL 98-0327

LOCATION AA5

SETPOINT: 20 ± .15"

ORIGIN: Level Switch (Q1P17LSLL3027CA-A)

A5
CCW SRG TK LVL A TRN LO-LO

PROBABLE CAUSE

Rupture or leakage of an A Train CCW component or pipe.

AUTOMATIC ACTION

1. Closes CCW Valves (Q1P17HV3096A&B) to isolate CCW to/from Evaporator Packages and H_2 Recombiners. (Q1P17LSLL3027CD-A)
2. Trips closed Q1P17HV2229, CCW to sample cooler (Q1P17LSLL3027CD-A).

IMMEDIATE ACTION

1. ENSURE THAT THE AUTOMATIC ACTIONS HAVE OCCURRED.
2. <u>IF</u> CCW FLOW HAS BEEN LOST TO THE SECONDARY HEAT EXCHANGER HEADER, <u>THEN</u> PERFORM THE ACTIONS REQUIRED BY FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.

SUPPLEMENTARY ACTION

1. Refer to FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.
2. Refer to Technical Specification 3.7.7 for LCO requirements with a loss of the on service train of component cooling water
3. Notify appropriate personnel to locate and correct the cause of the LO-LO level.

PROMPT OPERATOR ACTION REQUIRED

References: A-177100, Sh. 55; D-175002, Sh. 1 & 2; B-175968, Sh. 6; D-177183; D-277185; D-177092; D-177670; D-177853; B-175810, Sh. 9, 22, 23 & 101; Technical Specification 3.7.7; PCN B91-1-7431

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-9.0

LOSS OF COMPONENT COOLING WATER

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	ALL
Reference Use	
Information Use	

SAFETY RELATED

Approved:

TODD YOUNGBLOOD

Operations Manager

Date Issued: 11/12/2002

A. Purpose

This procedure provides actions for response to a loss of an operating component cooling water train.

This procedure is applicable at all times.

B. Symptoms or Entry Conditions

1. This procedure is entered when a loss of component cooling water is indicated by any of the following:

   a. Trip of any operating CCW PUMP

   b. Actuation of any of the following annunciators:

      [] CCW SRG TK LVL A TRN HI-LO annunciator AA4(for low conditions)
      [] CCW SRG TK LVL B TRN HI-LO annunciator AB4(for low conditions)
      [] CCW SRG TK LVL A TRN LO-LO annunciator AA5 (20 in)
      [] CCW SRG TK LVL B TRN LO-LO annunciator AB5 (20 in)

   c. Loss of SW supply to an operating CCW train

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: To prevent pump damage the 86 lockout relay for a faulted CCW PUMP must not be reset until the cause of the fault has been determined. Placing the affected CCW PUMP handswitch to STOP will reset this relay.

****

NOTE:
- The standby CCW PUMP will automatically start if the pump in the train it is aligned to trips due to electrical overload.
- The term "on service train" refers to the train which is aligned to supply the miscellaneous header.

___ 1 **Verify CCW pump started in affected train.**

yes

1 IF CCW cooling lost to running charging pump, THEN perform the following:

1.1 Verify CCW pump started in the non affected train.

1.2 IF CCW pump running in non affected train, THEN start charging pump in non affected train.

1.3 IF charging pump started in non affected train, THEN stop charging pump in affected train.

___ 2 **Check CCW surge tank level stable.**

No

2 Perform the following while continuing with step 3.0.

2.1 Maintain a level of 13 inches on CCW surge tank using SOP-23.0 sec 4.12 or 4.13.

2.2 Dispatch personnel to locate and isolate the source of leakage.

Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Step 3.0 is a continuing action step.
- IF RCP motor bearing temperatures exceed 195°F, THEN the ON SERVICE train is affected.
- Adequate CCW flow means sufficient cooling is available to maintain acceptable temperatures.(i.e. charging pumps,RHR cooling, SFP cooling, RCP's etc.)
- Indications of pump cavitation are: Abnormal CCW flow oscillations or cavitation noise reported at the pump.

Step	Action/Expected Response	Response NOT Obtained
3	**Check CCW system adequate for continued plant support.** • Check CCW flow adequate in affected train. • Check RCP motor bearing temperatures less than 195°F. • Check CCW pump not cavitating. Stop any cavitating CCW pump. • CCW Surge tank level being maintained at or above 13 inches.	3 Perform the following: 3.1 IF the ON SERVICE train is affected THEN perform the following: 3.1.1 IF the reactor is critical THEN trip the reactor and perform, FNP-1-EEP-0.0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure. 3.1.2 Verify all Reactor Coolant pumps stopped. 3.1.3 IF in Mode 3 or 4, THEN perform FNP-1-AOP-4.0 while continuing with procedure.

NOTE: Indications of CCW pump cavitation will be abnormal CCW flow oscillations or cavitation noise reported at the pump.

Step	Action/Expected Response	Response NOT Obtained
		3.2 IF evidence of CCW pump cavitation exists, THEN stop affected CCW pump.

Step 3 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

3.3 IF CCW cooling lost to running charging pump, THEN perform the following:

3.3.1 Verify CCW pump started in the non affected train.

3.3.2 IF CCW pump running in non affected train, THEN start charging pump in non affected train.

3.3.3 IF charging pump started in non affected train, THEN stop charging pump in affected train.

****

CAUTION: IF seal injection is in service, THEN at least one charging pump must be maintained to support RCP's.

****

NOTE: IF necessary to swap train alignment for the 1B charging pump for continued support of plant operations, THEN it is desirable to have separate operators performing the electrical and mechanical alignments simultaneously.

3.4 IF required to establish an operable charging pump in the non affected train, THEN shift the B charging pump to the non affected train per FNP-1-SOP 2.1B or FNP-1-SOP-2.1C.

3.5 IF CCW is NOT available to support charging operation, THEN align fire water using ATTACHMENT 1.

Step 3 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained
		3.6 IF CCW cooling to a running RHR pump is inadequate, THEN perform the following. 3.6.1 Stop the affected RHR pump. 3.6.2 IF in Mode 5 or 6 perform FNP-1-AOP-12.0, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION, in conjunction with this procedure.

NOTE: The ON SERVICE train includes the miscellaneous header and associated train header.

Step	Action/Expected Response	Response NOT Obtained
___ 4	**Verify SW supplied to the affected train.** SW flow from 1A,1B, or 1C CCW heat exchanger [] Q1P16FI3009AA 1A CCW HX DISC [] Q1P16FI3009BA 1B CCW HX DISC [] Q1P16FI3009CA 1C CCW HX DISC	4 IF Service Water is not available, THEN peform the following in conjunction with FNP-1-AOP-10, LOSS OF SERVICE WATER. 4.1 Check the following conditions met. [] Check if affected train is the ON SERVICE train. [] Unaffected train has adequate SW available. [] No CCW leak exists 4.2 IF conditions of Step 4.1 met, THEN swap ON SERVICE trains per FNP-1-SOP-23.0B or FNP-1-SOP-23.0C.

NOTE: IF previous actions have restored the ON SERVICE train, THEN the ON SERVICE train should no longer be considered affected.

Step	Action/Expected Response	Response NOT Obtained
___ 5	**Check ON SERVICE train affected.**	5 Go to step 10.0

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 6	**Check the ON SERVICE train intact.**	6 Perform the following:
		6.1 IF the miscellaneous header is intact, THEN shift the miscellaneous header to the non-affected train using ATTACHMENTS 2 or 3 depending upon train affected.
		6.1.1 WHEN ATTACHMENT 2 or 3 complete, THEN go to step 8.0.
		6.2 IF the miscellaneous header is not intact THEN proceed to step 8.0
___ 7	**Check 1A AND 1C CCW pumps AVAILABLE.**	7 Shift the Miscellaneous header to the non-affected train using Attachments 2 or 3 AND proceed to step 8.0.
	7.1 Shift the Miscellaneous header to the non-affected train using FNP-1-SOP-23.0B OR FNP-1-SOP-23.0C.	
8	**IF both seal injection and CCW are lost to the RCP's THEN isolate the RCP seal cooling**	
	8.1 Verify CCW return from RCP thermal barrier valves CLOSED. [] Q1P17HV3045 Closed [] Q1P17HV3184 Closed	8.1 Locally isolate CCW return [] Q1P17V107(121 PPR)
	8.2 Isolate RCP seal return valves. [] Q1E21MOV8112 Closed [] Q1E21MOV8100 Closed	8.2 Locally isolate seal water return lines(139 ft. rad side filter room) [] SEAL WATER RTN FILTER INLET Q1E21V189A---CLOSED [] SEAL WATER RTM FILTER BYPASS Q1E21V190---CLOSED

Step 8 continued on next page.

____Page Completed

Step | **Action/Expected Response** | **Response NOT Obtained**

8.3 Isolate seal injection.

Close seal water injection filter inlet valves(139 ft. rad side filter room)

SEAL WATER INJ FILTER A INLET

[] Q1E21V127A
[] Q1E21V127C

SEAL WATER INJ FILTER B INLET

[] Q1E21V127B
[] Q1E21V127D

NOTE:
- IF it is believed the miscellaneous header will be restored, THEN step 9 actions may be performed as necessary to reduce loads as needed to isolate affected equipment.
- This is a continuing action step.

9 **Perform the following:**

9.1 Secure letdown

9.1.1 Have Chemistry secure the Zinc Addition System per FBP-1-CCP-335.

9.1.2 Manually adjust LP LTDN PRESS PK 145 TO 50%.

9.1.3 Close Letdown Orifice isolation valves

LTDN ORIF ISO 45 GPM

[] Q1E21HV8149A Closed

LTDN ORIF ISO 60 GPM

[] Q1E21HV8149B Closed
[] Q1E21HV8149C Closed

Step 9 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
9.2	Manually secure charging flow. [] CLOSE FK-122.	

NOTE: Aligning charging pump suction to the RWST will result in borating the RCS.

Step	Action/Expected Response	Response NOT Obtained
9.3	Align charging pump suction to the RWST. RWST [] Q1E21LCV115B OPEN [] Q1E21LCV115D OPEN VCT [] Q1E21LCV115C CLOSED [] Q1E21LCV115E CLOSED	
9.4	Isolate RCP seal return RCP seal water return line. [] Q1E21MOV8112 CLOSED [] Q1E21MOV8100 CLOSED	9.4 Locally isolate seal water return line.(139' filter room) [] SEAL WATER RTN FILTER INLET Q1E21V189A---CLOSED [] SEAL WATER RTM FILTER BYPASS Q1E21V190---CLOSED
9.5	Verify excess letdown secured EXC LTDN ISO [] Q1E21HV8153 CLOSED [] Q1E21HV8154 CLOSED	
9.6	Verify RCDT in not on recirculation.	
9.7	Verify waste gas system shutdown	
9.8	Inform Chemistry to secure any sampling.	

Step 9 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 3.0 is a continuing action step and the RNO column has guidance for Charging and RHR pumps which may be applicable.

Step	Action/Expected Response	Response NOT Obtained
9.9	IF CCW leak is in the miscellaneous header AND cannot be isolated, THEN isolate the Miscellaneous header as follows:	
9.9.1	IF the reactor is critical THEN trip the reactor and perform, FNP-1-EEP-0.0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.	
9.9.2	Verify all Reactor Coolant pumps stopped.	
9.9.3	IF in Mode 3 or 4, THEN perform FNP-1-AOP-4.0 while continuing with procedure.	
9.9.4	Close CCW TO SECONDARY HXS, Q1P17MOV3047(Q1P17V030)	9.9.4 Manually isolate the Miscellaneous header by closing the following valves. (100' CCW HX Room) [] 1B CCW HX Outlet iso, Q1P17V008B. [] CCW SUPP HDR XCON, Q1P17V009B [] CCW SUPP HDR XCON, Q1P17V009C

___ Page Completed

Step	Action/Expected Response		Response NOT Obtained
___10	**Check both RHR pumps stopped**	10	<u>IF</u> CCW cooling to a running RHR pump is inadequate <u>THEN</u> perform the following:
		10.1	Stop the affected RHR pump.
		10.2	<u>IF</u> in modes 5 or 6 <u>THEN</u> perform FNP-1-AOP-12.0, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION, in conjunction with this procedure.
___11	**Evaluate event classification and notification requirements using FNP-0-EIP-8, NON-EMERGENCY NOTIFICATIONS <u>AND</u> FNP-0-EIP-9, EMERGENCY CLASSIFICATION <u>AND</u> ACTIONS.**		
___12	**Check SFP cooling aligned to an operating CCW train.**	12	Align SFP cooling to non affected train using FNP-1-SOP-23.0, COMPONENT COOLING WATER SYSTEM.
___13	**Check on service CCW train operating.**	13	Determine actions required to restore an ON SERVICE Train.
___14	**Go to procedure and step in effect.**		

-END-

83. 028AA2.02 001/1/2/PZR LVL/C/A 3.8/NEW/FA011005/S/CVR/GTO

A Unit 2 startup is in progress with plant conditions as follows:

- Reactor Power is 54%.
- A RCS temperature LOOP is in test and the corresponding Tavg 412D, Tavg 1A RCS LOOP, output is failed high.
- The B RCS Loop Tcold RTD fails high at the same time.

Which one of the following is the correct pressurizer level, if the operator takes no initial action, and the procedure the operator will use for the event?

A✓ Pressurizer level will rise to approximately 50% and stabilize. Manual control of FCV-122, Chg Flow Reg, will be required to control pressurizer level at program per SOP-2.1, Chemical and Volume Control System Plant Startup and Operation.

B. Pressurizer level will stabilize at approximately 36%. Manual control will not be required and UOP-3.1, Power Operation, will be maintained as the controlling procedure.

C. Pressurizer level will drop to approximately 21% and stabilize. Manual control of FCV-122, Chg Flow Reg, will be required to control pressurizer level at program per SOP-2.1.

D. Pressurizer level will drop to to 15% causing an automatic isolation of Letdown. Restoration of letdown per AOP-16.0, Loss of Letdown, and manual control of FCV-122, Chg Flow Reg, will be required to control pressurizer level at program per SOP-2.1.

A. CORRECT. **Pressurizer level will rise to approximately 50% and stabilize. Manual control of FCV-122, Chg Flow Reg, will be required to control pressurizer level at program per SOP-2.1, Chemical and Volume Control System Plant Startup and Operation.**

When B Tc fails high, B Tavg fails high also (B delta T fails low). this causes both A & B loop Tavg to be failed high. That causes Median Tavg to fail high also which causes Prz Program level to fail high. Program level is limited high to 50.2%.

B. INCORRECT. This would be correct if the failed B loop Tavg failed low instead of high. If that were the case, the remaining operable C loop Tavg would be the median signal.

C. INCORRECT. This would be correct if 2 out of 3 Tavg failed low instead of high.

D. INCORRECT. This would be correct if 2 out of 3 Tavg failed low instead of high AND pzr level program was not limited low to 21.4%.

028AA2.02 Pressurizer Level Malfunction / 2
Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions:(CFR: 43.5 / 45.13)
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

AA2.02 – PZR level as a function of power level or T-ave including interpretation of malfunction

4. Predict and explain the following instrument/equipment response expected when performing TAVG, .T, and PIMP System evolutions including the fail condition, alarms, trip setpoints (OPS52201J08):

- Thot RTD fails high
- Thot RTD fails low
- Tcold RTD fails high
- Tcold RTD fails low
- PT-446 failures
- PT-447 failures
- High TAVG Alarm
- TAVG Deviation Alarm
- Rod Insertion Limit Computer
- .T Deviation Alarm
- Steam Dump control
- Pressurizer Level Control

an excessive cooldown of the RCS. Excessive cooldown could result in a positive reactivity addition accident. The interlock signal is generated by T_{AVG} less than 554°F in coincidence with a reactor trip (P-4).

Low-Low T_{AVG} Interlock

Another excessive cooldown protection signal is the low-low T_{AVG} interlock (P-12). The setpoint is 543°F with T_{AVG} decreasing. The interlock signal automatically blocks the steam dump system below the setpoint (543°F). The interlock may be manually bypassed to allow continued cooldown of the plant below 543°F. The low-low T_{AVG} signal (P-12), in conjunction with high steam line flow, automatically isolates the main steam lines.

High Temperature Alarm

Any loop's T_{AVG} channel that is above 4°F above full load T_{AVG} will actuate a high T_{AVG} alarm on the MCB.

T_{AVG} Control Circuits

Figure 6

The three T_{AVG} protection signals from the loop instruments are input to a median signal selector (MSS) after being electronically isolated to prevent a failure in the control systems from adversely affecting the protection functions. The function of the MSS is to automatically reject both the highest and the lowest T_{AVG} signals, therefore using the median T_{AVG} for all control purposes. Use of the median selected T_{AVG} ensures that, in the event of the failure of a single RTD, the control functions supplied by T_{AVG} will see little or no effect, thus ensuring that plant transients will be kept to a minimum.

Pressurizer Level Program

The median T_{AVG} signal is used by the pressurizer level control system to generate a reference level. This reference level is compared to actual pressurizer level in order to control the charging flow rate and to maintain the reference and actual levels equal. Refer to the Pressurizer Pressure and Level Control lesson for more detail.

T_{AVG} Control

The T_{AVG} signal used for control is the result of a process in which the T_{AVG} protection signal for each loop is input to an MSS. The MSS rejects both the highest and the lowest T_{AVG} signals using the median selected T_{AVG} signal for all control functions.

Rod Control System

The median T_{AVG} signal is processed and filtered prior to being compared to T_{ref} in the rod control system. Refer to the Full Length Rod Control lesson for a more detailed explanation of the system operation.

Steam Dump Control System

The median T_{AVG} signal is compared to T_{ref} for load rejection control and compared to an identical setpoint for plant trip control. Refer to the Steam Dump Control lesson for more detail.

Pressurizer Level Program

The median T_{AVG} signal, in conjunction with an adjustable no-load T_{AVG} setpoint, is used to generate the level program for the pressurizer. The level program is 21.4 percent to 54.9 percent from no-load to full load T_{AVG}. Refer to the Pressurizer Pressure and Level Control lesson for more detail.

Rod Insertion Limits

The minimum required shutdown margin given in the Technical Specifications protects the core from damage if a postulated steam line break occurs. The shutdown margin criteria is set to maintain departure from nucleate boiling ratio (DNBR) at greater than 1.3.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and T_{AVG}. The most restrictive condition occurs at end-of-life (EOL), with T_{AVG} at no-load operating temperature.

In the analysis of the steam line break accident, a minimum shutdown margin of 1.77 percent $\Delta K/K$ is required to control the positive reactivity addition. The reactor will still go critical, but power will be turned by the safety injection boron prior to reaching a DNBR of 1.3 (refer to Figure 11). As a further conservative consideration in obtaining a minimum shutdown

FARLEY JANUARY 2005 EXAM

EXAM 50-348 AND 50-364/2005-301
JANUARY 10 - 14, 2005
JANUARY 18, 2005 (written)

DRAFT SRO WRITTEN EXAM

OpsSsp007 Tavg Protection Figure 5
OpsFth224 Tavg and ?T Control Figure 6
OpsPrs013 Pressurizer Program Level Figure 8
OpsPrs053 Pressurizer Level Program Figure 6
(Total of 4 Pages Omitted)

INTENTIONALLY OMITTED

PER SISP REVIEW

84. 028G2.4.48 001/2/2/H2 RECOMB/C/A 3.8/BANK/FA011005/S/CVR/GTO

Given the following plant conditions:

- A large break LOCA has occurred on Unit 2 thirty (30) minutes ago.
- Hydrogen concentration inside containment is 4.5%.

Which ONE of the following is the correct actions which should be taken within the next 30 minutes to reduce hydrogen concentration?

A. Place only ONE Post LOCA containment hydrogen recombiner in service at a power setting of 100 kilowatts.

B. Place BOTH Post LOCA containment hydrogen recombiners in service at a power setting of 50 kilowatts.

C. Evaluate placing the post LOCA containment pressurization and venting system in service.

D. Evaluate placing the post LOCA containment air mixing system in service.

A - Incorrect, Recombiner operation should commence within 1 hour of the event, however, the max power setting is initially set at approximately 68 kilowatts and recombiners should not be put in service if hydrogen concentration is above 4%.

B - Incorrect, Recombiner operation should commence within 1 hour of the event, however, the recombiners should not be put in service if hydrogen concentration is above 4%.

C - Correct, **Evaluate placing the post LOCA containment pressurization and venting system in service.**
Per EEP-1, the post accident containment venting system should be evaluated to be placed in service when hydrogen concentration is above 4%.

D - Incorrect, The post LOCA containment air mixing system is used to prevent the formation of hydrogen pockets and not to reduce the overall concentration.

028G2.4.48

028 Hydrogen Recombiner and Purge Control

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.
(CFR: 43.5 / 45.12)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. Evaluate abnormal plant or equipment conditions associated with the Post LOCA Atmospheric Control System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102D02).

EEP-1-52530B04 01
Source: Farley NRC Exam 1998
2001 nrc exam

Given the following plant conditions:

- A large break LOCA has occurred on Unit 2 thirty (30) minutes ago.
- Hydrogen concentration inside containment is 4.5%.

Which ONE of the following actions should be taken within the next 30 minutes to reduce hydrogen concentration?

A. Place only ONE Post LOCA containment hydrogen recombiner is service at a power setting of 100 kilowatts.

B. Place BOTH Post LOCA containment hydrogen recombiners in service at a power setting of 50 kilowatts.

C. (CORRECT) Evaluate placing the post LOCA containment pressurization and venting system in service.

D. Evaluate placing the post LOCA containment air mixing system in service.

Step	Action/Expected Response	Response NOT Obtained

NOTE: The purpose of the following step is to establish an available PORV flowpath for mitigation of overpressure conditions, without relying on the PRZR code safety valves. A failed open PORV must not be unisolated. A leaking PORV which is isolated with power available to the isolation valve should remain isolated until needed to reduce RCS pressure or mitigate an RCS overpressure condition. Any leaking PORV should be re-isolated when not in use.

Step	Action/Expected Response	Response NOT Obtained
5.3	Check at least one PRZR PORV ISO - OPEN.	5.3 Open any PRZR PORV ISO not required to isolate an open or leaking PORV.

NOTE: IF the event has been diagnosed as a secondary system break, THEN the performance of the following actions are NOT required. Otherwise the following one hour actions should commence immediately.

Step	Action/Expected Response	Response NOT Obtained
6	**Perform the following within one hour of start of event.**	
6.1	Close recirculation valve disconnects using ATTACHMENT 1.	
6.2	Establish 1A and 1B post LOCA containment hydrogen analyzers - IN SERVICE USING ATTACHMENT 2, POST LOCA CONTAINMENT HYDROGEN ANALYZER OPERATION.	6.2 IF at least one analyzer in service, THEN proceed to step 6.3 IF NOT, direct Chemistry to sample containment atmosphere for hydrogen using FNP-0-CCP-1300, CHEMISTRY AND ENVIRONMENTAL ACTIVITIES DURING A RADIOLOGICAL ACCIDENT.
6.3	Plot hydrogen concentration on FIGURE 1.	

Step 6 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Fire or explosion may occur if post LOCA hydrogen recombiners are placed in service when containment hydrogen concentration is greater than 4%.

****

****

CAUTION: To prevent diesel generator overloading, at least 0.075 MW (75 kW) of diesel generator capacity must be available prior to starting a hydrogen recombiner.

****

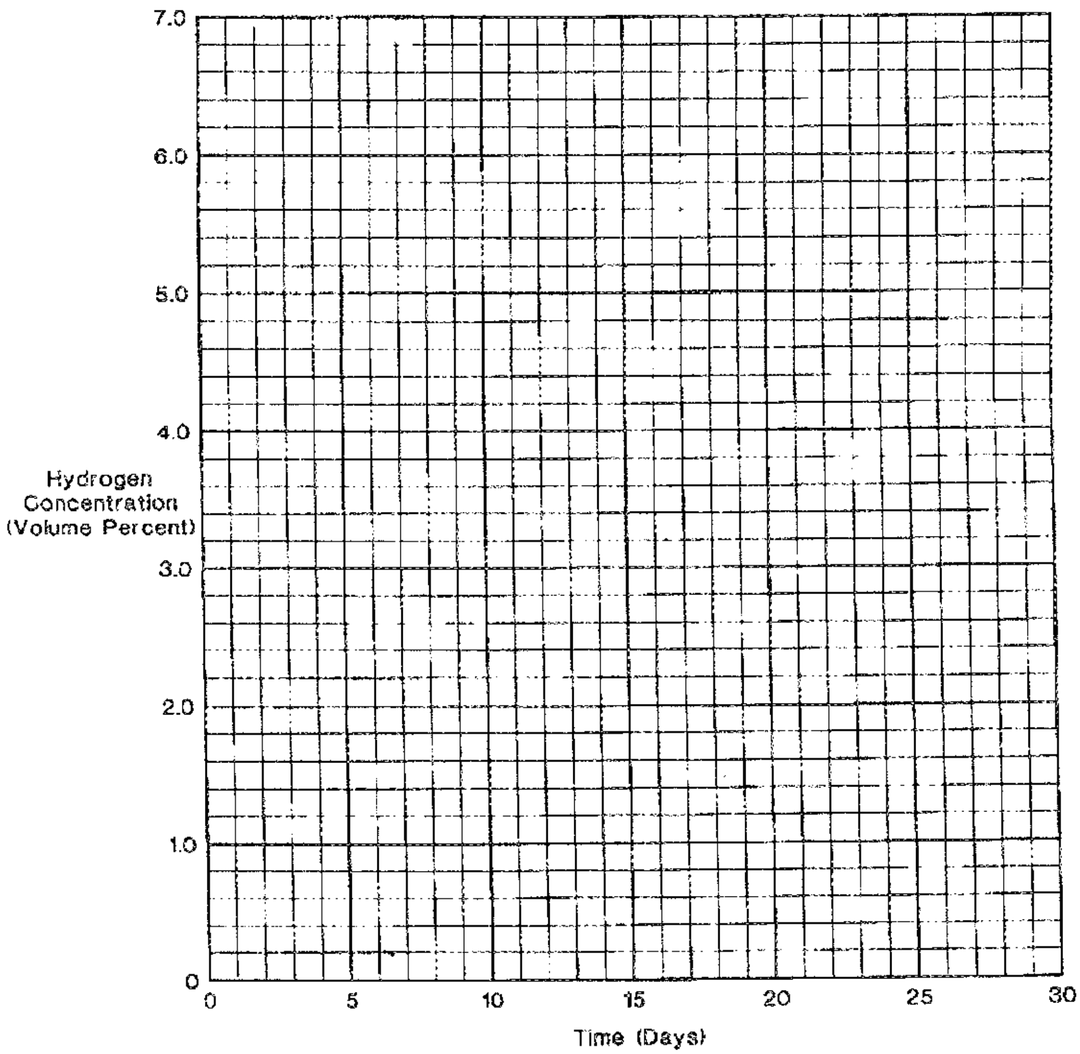
Step	Action/Expected Response	Response NOT Obtained
6.4	Check containment hydrogen concentration - LESS THAN 3.5%.	6.4 Evaluate other methods of hydrogen control such as post LOCA containment pressurization and vent system.
6.5	Check containment hydrogen concentration - LESS THAN 0.5%.	6.5 Establish both post LOCA containment hydrogen recombiners in service using ATTACHMENT 3, POST LOCA HYDROGEN RECOMBINER OPERATION.

____Page Completed

FIGURE 1

CONTAINMENT HYDROGEN CONCENTRATION PLOT

Fig 1
Att. 1
Att.2
Att. 3



ATTACHMENT 1

NOTE: The Master "Z" key may be obtained for use in the following step instead of checking out the individual keys.

___ 1 Close the following disconnects.

'B' Train Disconnects				
Disconnect TPNS No.	Description	Position	Key	Location
Q1R18B033-B	Disconnect FV-T2 MOV 8884-B	ON	Z-91	139' hallway across from chemistry sample room
Q1R18B034-B	Disconnect FV-J2 MOV 8132B-B	ON	Z-89	
Q1R18B035-B	Disconnect FV-S2 MOV 8808B-B	ON	Z-86	
Q1R18B036-B	Disconnect FV-B2 MOV 8889-B	ON	Z-88	
Q1R18B041-B	Disc for MOV 8130B-B	ON	Z-408	
Q1R18B042-B	Disc for MOV 8131B-B	ON	Z-412	
Q1R18B043-B	Disc for MOV 8133B-B	ON	Z-416	

ATTACHMENT 1

NOTE: The Master "Z" key may be obtained for use in the following step instead of checking out the individual keys.

____ 2 Close the following disconnects.

'A' Train Disconnects				
Disconnect TPNS No.	Description	Position	Key	Location
Q1R18B030-A	Disconnect FU-J2 MOV 8132A-A	ON	Z-203	139' hallway across from MCC 1A
Q1R18B029-A	Disconnect FU-R2 MOV 8886-A	ON	Z-90	
Q1R18B038-A	Disc for MOV 8130A-A	ON	Z-405	
Q1R18B039-A	Disc for MOV 8131A-A	ON	Z-410	
Q1R18B040-A	Disc for MOV 8133A-A	ON	Z-415	
Q1R18B031-A	Disconnect FU-Z3 MOV 8808C-A	ON	Z-85	
Q1R18B032-A	Disconnect FU-Z2 MOV 8808A-A	ON	Z-84	

ATTACHMENT 1

___ 3 Verify recirculation valves MCB indication - POWER AVAILABLE.

CHG PUMP
SUCTION HDR ISO
[] Q1E21MOV8130A
[] Q1E21MOV8130B
[] Q1E21MOV8131A
[] Q1E21MOV8131B

CHG PUMP
DISCH HDR ISO
[] Q1E21MOV8132A
[] Q1E21MOV8132B
[] Q1E21MOV8133A
[] Q1E21MOV8133B

1A(1B,1C) ACCUM
DISCH ISO
[] Q1E21MOV8808A
[] Q1E21MOV8808B
[] Q1E21MOV8808C

CHG PUMP RECIRC
TO RCS HOT LEGS
[] Q1E21MOV8884
[] Q1E21MOV8886

RHR TO RCS
HOT LEGS ISO
[] Q1E11MOV8889

___ 4 Notify control room of recirculation valve disconnect status.

-END-

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 2

POST LOCA CONTAINMENT HYDROGEN ANALYZER OPERATION

NOTE: Key CAT8 and eight BOP keys are required to perform this attachment.

1 Verify 1A post LOCA containment hydrogen analyzer - IN SERVICE.

1.1 Verify the following containment sample valves - OPEN. (BOP key operated switches)

1.1 Proceed to step 2.

CTMT POST ACCIDENT SAMPLE ISO
[] Q1E23MOV3739A

CTMT POST ACCIDENT SAMPLE RTN
[] Q1E23MOV3745A
[] Q1E23MOV3835A

1.2 Verify only one of the following containment sample valves - OPEN. (BOP key operated switches)

1.2 Proceed to step 2.

CTMT POST ACCIDENT SAMPLE PT 1
[] Q1E23MOV3528A

CTMT POST ACCIDENT SAMPLE PT 2
[] Q1E23MOV3528B

Step 1 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 2

Step	Action/Expected Response	Response NOT Obtained
1.3	Verify control room remote panel aligned.	1.3 Proceed to step 2.
1.3.1	Verify REMOTE CONTROL SELECTOR switch - DEPRESSED.	
1.3.2	Verify FUNCTION SELECTOR switch - IN SAMPLE POSITION.	
1.3.3	Verify OFF-STANDBY-ANALYZE switch - IN ANALYZE POSITION.	

****

CAUTION: Post accident dose rates may not allow access to the electrical penetration room.

****

Step	Action/Expected Response	Response NOT Obtained
1.4	Check H2 ANAL 10% RANGE indicating light - LIT.	1.4 Place RANGE SELECT switch in 10% position. (139 ft, AUX BLDG electrical penetration room at local control panel)
1.5	Turn on chart recorder power switch. (right side of chart recorder)	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 2

2 Verify 1B post LOCA containment hydrogen analyzer - IN SERVICE.

2.1 Verify the following containment sample valves - OPEN. (BOP key operated switches)
2.1 Proceed to step 3.

CTMT POST ACCIDENT SAMPLE ISO
[] Q1E23MOV3739B

CTMT POST ACCIDENT SAMPLE RTN
[] Q1E23MOV3745B
[] Q1E23MOV3835B

2.2 Verify only one of the following containment sample valves - OPEN. (BOP key operated switches)
2.2 Proceed to step 3.

CTMT POST ACCIDENT SAMPLE PT 3
[] Q1E23MOV3528C

CTMT POST ACCIDENT SAMPLE PT 4
[] Q1E23MOV3528D

2.3 Verify control room remote panel aligned.
2.3 Proceed to step 3.

2.3.1 Verify REMOTE CONTROL SELECTOR switch - DEPRESSED.

2.3.2 Verify FUNCTION SELECTOR switch - IN SAMPLE POSITION.

2.3.3 Verify OFF-STANDBY-ANALYZE switch - IN ANALYZE POSITION.

Step 2 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 2

****

CAUTION: Post accident dose rates may not allow access to the electrical penetration room.

****

Step	Action/Expected Response	Response NOT Obtained
2.4	Check H2 ANAL 10% RANGE indicating light - LIT.	2.4 Place RANGE SELECT switch in 10% position. (139 ft, AUX BLDG electrical penetration room at local control panel)
2.5	Turn on chart recorder power switch. (right side of chart recorder)	
___ 3	Notify control room of post LOCA containment hydrogen analyzer status.	

-END-

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 3

POST LOCA HYDROGEN RECOMBINER OPERATION

___ 1 Check 1A post LOCA containment hydrogen recombiner output temperature - 1150-1400°F.

TEMPERATURE READOUT
[] Channel 1
[] Channel 2
[] Channel 3

1 Align 1A post LOCA containment hydrogen recombiner for operation.

1.1 Verify POWER AVAILABLE indicating light lit.

1.2 Adjust POWER ADJUST setting to 000.

1.3 Turn on POWER OUT SWITCH.

1.4 Verify red pilot light lit.

NOTE: The 12 psig value should be used for containment pressures greater than 12 psig.

1.5 Determine initial power out setting using ATTACHMENT 3, FIGURE 1 .

1.6 Slowly adjust POWER ADJUST clockwise until POWER OUT meter indicates initial power out setting.

1.7 Control POWER ADJUST to maintain recombiner output temperature 1150-1400°F.

TEMPERATURE READOUT
[] Channel 1
[] Channel 2
[] Channel 3

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 3

___ 2 Check 1B post LOCA containment hydrogen recombiner output temperature - 1150-1400°F.

TEMPERATURE READOUT
- [] Channel 1
- [] Channel 2
- [] Channel 3

2 Align 1B post LOCA containment hydrogen recombiner for operation.

2.1 Verify POWER AVAILABLE indicating light lit.

2.2 Adjust POWER ADJUST setting to 000.

2.3 Turn on POWER OUT SWITCH.

2.4 Verify red pilot light lit.

NOTE: The 12 psig value should be used for containment pressures greater than 12 psig.

2.5 Determine initial power out setting using ATTACHMENT 3, FIGURE 1 .

2.6 Slowly adjust POWER ADJUST clockwise until POWER OUT meter indicates initial power out setting.

2.7 Control POWER ADJUST to maintain recombiner output temperature 1150-1400°F.

TEMPERATURE READOUT
- [] Channel 1
- [] Channel 2
- [] Channel 3

___ 3 Notify control room of post LOCA containment hydrogen recombiner status.

-END-

FARLEY JANUARY 2005 EXAM

EXAM 50-348 AND 50-364/2005-301
JANUARY 10 - 14, 2005
JANUARY 18, 2005 (written)

DRAFT SRO WRITTEN EXAM

Loss of Reactgor or Secondary Coolant
9/22/2004 07:52 - FNP-1-EEp-1 - Revision 25
Page 3 of 3
Recombiner Power Out Setting
Attachment 3, Figure 1

INTENTIONALLY OMITTED

PER SISP REVIEW

85. 029G2.4.21 001/1/1/ATWT/C/A 4.3/NEW/FA011005/S/CVR/GTO

Unit 1 was at 100% when both SGFPs tripped. The Reactor failed to trip automatically and manually and one Rod Control Motor Generator breaker did not trip.

Current plant conditions are as follows:

- Manual Control Rod insertion in progress.
- Reactor Power is 1% and falling.
- Intermediate Range Startup rate is negative.
- A Safety Injection has occurred.
- Containment pressure is 16 psig and rising.

During the performance of FRP-S.1, Response to Nuclear Power Generation/ ATWT, at the step to Verify AFW pumps running, the following is reported by the Unit Operator:

- Narrow Range Steam Generator water levels are: A-33%, B-35%, C-30%
- Total AFW flow of 200 gpm is the maximum obtainable.

Which one of the following is the correct procedural response?

A. Exit FRP-S.1 immediately. Enter FRP-H.1, Response to Loss of Secondary Heat Sink.

B✓ Complete FRP-S.1, then enter FRP-H.1.

C. Exit FRP-S.1 immediately. Enter EEP-0, then enter FRP-H.1 when directed by EEP-0.

D. Complete FRP-S.1, then enter EEP-0, then enter FRP-H.1 when directed by EEP-0.

A. Incorrect. Even though the red path on Subcriticality is no longer in, SOP-0.8 step 4.4 states that once a FRP is initiated due to a red or orange path the procedure must be continued to the point that the procedure directs exit.

B. Correct. **Complete FRP-S.1, then enter FRP-H.1.**
The FRP-S.1 will be completed at step 14 and the red path (which is red due to adverse numbers requiring at least one SG >48% NR LVL or AFW flow >395 gpm) on heat sink will be addressed by FRP-H.1.

C. Incorrect. Even though the red path on Subcriticality is clear, SOP-0.8 states that once a FRP is initiated due to a red or orange path the procedure must be completed to the point that the procedure directs exit. During the performance of FRP-S.1, the note prior to step 6 directs doing steps 1-15 of EEP-0 only as manpower and time permit **in conjunction with** FRP-S.1.

D. Incorrect. This would be true if Adverse numbers were not in effect, or if >395 gpm AFW flow was obtainable. Since there is a red path on Heat Sink, it must be addressed, but not until FRP-S.1 directs exit.

029G2.4.21 ATWS / 1
2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.
(CFR: 43.5 / 45.12)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

9. Using appropriate ERPs, recognize the symptoms and perform the procedural guidance to mitigate the consequences of the conditions to include (OPS52504A09):
a. Reactor trip (w/wo stuck rod)
b. Safety injection (steam break, SGTR, LOCA, or Spurious)
c. Anticipated transient without trip (ATWT)
d. Reactor trip recovery
e. SI termination
f. Loss of secondary heat sink
g. Loss of forced flow

3.17 Use of Adverse Containment Values

Due to the affect of environmental conditions on instrumentation, the ERP's are written such that some operator decisions are based on conditions within containment. The following guidelines are used to determine the continued applicability of adverse values:

- If adverse values were entered due to containment pressure > 4 psig, then adverse values remain in effect until pressure is < 4 psig and the applicable instruments have been channel checked.
- If adverse values were entered due to containment radiation > 10^5 R/hr, then adverse values remain in effect until the integrated radiation dose is verified to be less than 10^6 Rad.

4.0 Critical Safety Function Status Trees (CSFSTs)

4.1 General

The ERP network is designed to protect the health and safety of the public by maintaining the fission product barriers intact. In the initial stage of ERP performance, if AC power is available, the user ensures that the automatic plant systems are functioning properly to protect these barriers. Afterwards, the CSFSTs are monitored to detect challenges to the barriers due to worsening plant conditions or equipment failure and to direct the user to an appropriate procedure.

4.2 Applicability

The user should begin monitoring the CSFSTs when directed by EEP-0 or upon transition from EEP-0. The CSFSTs are not monitored initially because the ERPs are already directing the initial action required to protect the barriers. If the user enters ECP-0.0, the CSFSTs should be monitored for information only. The Function Restoration Procedures assume that at least one train of safeguards busses is available. If all AC power has been lost, ECP-0.0 will provide the appropriate actions to protect the barriers.

Although some procedures are provided with a foldout page containing a CSF red path summary, the user must continue to monitor the CSFSTs when performing these procedures as well.

4.3 Proper Use

The CSFSTs follow a logic tree format. Each CSFST has a single entry point at the left side of the page. When manually monitoring the CSFSTs, the user must enter at this point and then proceed to the right until reaching an endpoint. The endpoint will either indicate that the particular CSF is satisfied or direct the user to an appropriate procedure. The Safety Parameter Display System (SPDS) provides real time monitoring of the CSFSTs. The user should perform CSF-0 to determine if SPDS is functioning properly. If so, it should be used. If not, manual monitoring is required.

4.4 Priority

The CSFSTs shall be continuously monitored in the following order of priority:

S - Subcriticality
C - Core Cooling
H - Heat Sink
P - Integrity
Z - Containment
I - Inventory

If a red path is identified, the user will, unless specifically directed otherwise, immediately suspend the procedure in effect and transition to the specified FRP. CSFST monitoring will continue so that if a higher priority red path occurs, it will be identified.

If an orange path is identified, the user will monitor the remaining CSFSTs to ensure that no red path exists. Unless specifically directed otherwise, he will then suspend the procedure in effect and transition to the specified FRP. CSFST monitoring will continue so that if a higher priority orange path or any red path occurs, it will be identified.

If a yellow path is identified, the user is not required to transition to the specified FRP. This indicates an off normal condition that the user should be aware of, but which does not yet challenge a CSF. Implementation of a yellow path FRP is based upon operator judgement when it is determined that adequate time exists to implement it. Optimal recovery procedures (EEPs, ESPs, and ECPs) have priority over yellow path FRPs. While performing a yellow path FRP, continuous actions or foldout page items of the optimal recovery procedure in effect are still applicable and should be monitored. Concurrent procedure usage should not cause difficulties since yellow path FRPs are only performed when adequate time exists.

Once an FRP has been entered due to a red or orange path, the FRP must be performed to completion unless it is preempted by a higher priority FRP. It is expected that the FRP will correct the red or orange condition before all of the operator actions are performed but the user must continue until the FRP directs a transition. In general, the performance of the critical safety functions is based on the current plant parameters. IF a red or orange path condition comes in and clears, THEN the associated FRP does not need to be performed. IF conditions degrade, THEN the status of the safety function will become a continuous red or orange condition at which time the operator would be directed to the appropriate critical safety function.

5.0 User Feedback {CMT 0008175}

5.1 Purpose

The ERP network has been designed to provide the user with a tool for managing the plant during emergency conditions. The primary goal is to make the ERPs as technically accurate as is practical. To accomplish this, it is necessary to receive input from the ERP users concerning possible problems or potential improvements.

5.2 Method

As the users study and perform the ERPs in the plant and in training, they should be alert to any possible problems such as typographical errors, labeling errors or confusing language.

When a user identifies a possible problem with or a potential improvement for an ERP, he should fill out an ERP USER FEEDBACK FORM (Figure 1) or utilize other approved methods for procedure change suggestions as described in FNP-0-AP-1.0, DEVELOPMENT, REVIEW AND APPROVAL OF PLANT PROCEDURES. IF utilized, THEN the completed form should be routed to the ERP Writer. It is important that the form be filled out as completely as possible except for the disposition section. This will help the ERP Writer understand what has been requested and enable him to contact the particular user for more information or to advise him of the disposition of his input.

The ERP Writer and ERP Coordinator will review the form to determine if an ERP change is required and if so, determine the priority of the change. Once the appropriate disposition has been determined, the disposition section will be completed and the form will be signed by the ERP Coordinator. A copy of the completed form will be sent to the originator.

FARLEY NUCLEAR PLANT

FUNCTION RESTORATION PROCEDURE

FNP-1-FRP-S.1

RESPONSE TO NUCLEAR POWER GENERATION/ATWT

S
A
F
E
T
Y

R
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L
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T
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D

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	**Remainder of Procedure**
Reference Use	**Steps 1-2**
Information Use	

Approved:

TODD YOUNGBLOOD
Operations Manager

Date Issued: 3-6-02

Table of Contents

A. **Purpose**

This procedure provides actions to add negative reactivity to a core which is observed to be critical when expected to be shutdown.

B. **Symptoms or Entry Conditions**

I. This procedure is entered when reactor trip is not verified and manual trip is not effective; from the following:

a. FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION, step 1

b. A Foldout Page

II. This procedure is entered from the Subcriticality Critical Safety Function Status Tree on either a Red or Orange condition.

FNP-1-FRP-S.1	RESPONSE TO NUCLEAR POWER GENERATION/ATWT	Revision 21

Step	Action/Expected Response	Response NOT Obtained

NOTE: Steps 1 and 2 are IMMEDIATE ACTION steps.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Verify reactor - TRIPPED.** RX TRIP ACTUATION [] AB to TRIP	1 Perform the following. 1.1 Trip CRDM MG set supply breakers. 1A(1B) MG SET SUPP BKR [] N1C11EC05A [] N1C11EC05B 1.2 IF reactor still NOT tripped, THEN perform one of the following. • Insert control rods in manual control. OR • Verify rods insert in AUTO at greater than 48 steps per minute.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: The DEH Valve Test Display may be used for alternate indication of turbine stop valve and governor valve position.

Step	Action/Expected Response	Response NOT Obtained
2	**Check turbine - TRIPPED.** [] TSLB2 14-1 lit [] TSLB2 14-2 lit [] TSLB2 14-3 lit [] TSLB2 14-4 lit	2 Perform the following. 2.1 Place MAIN TURB EMERG TRIP switch to TRIP for at least 5 seconds. 2.2 <u>IF</u> turbine <u>NOT</u> tripped, <u>THEN</u> close GVs. 2.2.1 Reduce GV position demand signal to zero from DEH panel. [] TURBINE MANUAL depressed [] CV CLOSE depressed [] FAST ACTION depressed

Step 2 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

2.3 IF steam flow to main turbine NOT secured, THEN perform the following.

2.3.1 Close all main steam line isolation and bypass valves.

1A(1B,1C) SG MSIV - TRIP
- [] Q1N11HV3369A
- [] Q1N11HV3369B
- [] Q1N11HV3369C
- [] Q1N11HV3370A
- [] Q1N11HV3370B
- [] Q1N11HV3370C

1A(1B,1C) SG MSIV - BYPASS
- [] Q1N11HV3368A
- [] Q1N11HV3368B
- [] Q1N11HV3368C
- [] Q1N11HV3976A
- [] Q1N11HV3976B
- [] Q1N11HV3976C

2.3.2 IF any MSIV fails to close THEN place the associated test switch in the TEST position.

SG	1A	1B	1C
1A(1B,1C) SG MSIV - TEST Q1N11HV	[]3369A/70A	[]3369B/70B	[]3369C/70C

___ Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Verify AFW pumps - RUNNING.**	
	3.1 MDAFWPs - RUNNING [] 1A amps > 0 [] 1B amps > 0	
	3.2 TDAFWP - RUNNING IF NECESSARY • TDAFWP STM SUPP FROM 1B(1C) SG [] MLB-4 1-3 lit [] MLB-4 2-3 lit [] MLB-4 3-3 lit • TDAFWP SPEED [] SI 3411A > 3900 rpm • TDAFWP SPEED CONT [] SIC 3405 at 100%	

NOTE:
- 2500 gallons of emergency boration is required for each control rod not fully inserted, up to a maximum of 17,309 gallons.
- Emergency boration should continue until an adequate shutdown margin is established.

Step	Action/Expected Response	Response NOT Obtained
4	**Initiate Emergency Boration of the RCS.**	
	4.1 Verify at least one CHG PUMP - RUNNING.	

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.2	Start a boric acid transfer pump. BATP [] 1A [] 1B	4.2 Perform the following. 4.2.1 Align charging pump suction to RWST. RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed 4.2.2 Proceed to step 4.4.
4.3	Align normal emergency boration. EMERG BORATE TO CHG PUMP SUCT [] Q1E21MOV8104 open	4.3 Perform the following. • Align charging pump suction to RWST. RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed <u>OR</u> • Align manual emergency boration flow path. BORIC ACID TO BLENDER [] Q1E21FCV113A open MAN EMERG BORATION [] Q1E21V185 open (100 ft, AUX BLDG rad-side chemical mixing tank area)

Step 4 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.4	Establish adequate letdown.	
4.4.1	Verify 45 GPM letdown orifice - IN SERVICE. LTDN ORIF ISO 45 GPM [] Q1E21HV8149A open	
4.4.2	Verify one 60 GPM letdown orifice - IN SERVICE. LTDN ORIF ISO 60 GPM [] Q1E21HV8149B open [] Q1E21HV8149C open	
4.5	Check pressurizer pressure LESS THAN 2275 psig	4.5 Verify PRZR PORVs and PRZR PORV ISOs - OPEN. <u>IF</u> <u>NOT</u>, <u>THEN</u> open PRZR PORVs and PORV ISOs as necessary until pressurizer pressure less than 2135 psig.
4.6	Establish adequate charging flow. • <u>IF</u> boration is from boric acid storage tank, <u>THEN</u> verify charging flow - GREATER THAN 40 gpm. <u>OR</u> • <u>IF</u> boration is from the RWST, <u>THEN</u> verify charging flow - GREATER THAN 92 gpm.	

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.7	Verify emergency boration flow adequate.	
4.7.1	IF normal emergency boration flow path aligned, THEN check emergency boration flow greater than 30 gpm. BORIC ACID EMERG BORATE [] FI 110	
4.7.2	IF manual emergency boration flow path aligned, THEN check boric acid flow greater than 30 gpm. MAKEUP FLOW TO CHG/VCT [] BA FI 113	
4.7.3	IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 gpm.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 5	**Verify containment ventilation isolation.**	
5.1	Verify containment purge dampers - CLOSED. [] 3197 [] 3198D [] 3198C [] 3196 [] 3198A [] 3198B	5.1 Manually close dampers.
5.2	Verify containment mini purge dampers - CLOSED. CTMT PURGE DMPRS MINI-2866C & 2867C FULL-3198A & 3198D [] 2866C [] 2867C CTMT PURGE DMPRS MINI-2866D & 2867D FULL-3196 & 3197 BOTH-3198B & 3198C [] 2866D [] 2867D	5.2 Manually close dampers.
5.3	Stop MINI PURGE SUPP/EXH FAN.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: IF an SI signal exists or occurs, THEN steps 1 through 15 of FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION, should be performed (if not already performed) in conjunction with this procedure as manpower and time permit.

****

Step	Action/Expected Response	Response NOT Obtained
___ 6	**Check the following trips.**	
6.1	Verify all reactor trip and reactor trip bypass breakers - OPEN.	6.1 Locally open the reactor trip and reactor trip bypass breakers.
6.2	Check all turbine stop valves - CLOSED. [] TSLB2 14-1 lit [] TSLB2 14-2 lit [] TSLB2 14-3 lit [] TSLB2 14-4 lit	6.2 Locally place turbine overspeed lever to TRIP for at least 5 seconds. (189 ft, TURB BLDG)
___ 7	**Monitor CST level.**	
7.1	Check CST level greater than 5.3 ft. CST LVL [] LI 4132A [] LI 4132B	7.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.
7.2	Align makeup to the CST from water treatment plant OR demin water system using FNP-1-SOP-5.0, DEMINERALIZED MAKEUP WATER SYSTEM, as necessary.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
8	**Check SG levels.**	
8.1	Check at least one SG narrow range level - GREATER THAN 31%{48%}.	8.1 Perform the following. 8.1.1 Verify total AFW flow greater than 700 gpm. AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229 8.1.2 Verify AFW valves open. MDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3227AA [] HIC 3227BA [] HIC 3227CA TDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3228AA [] HIC 3228BA [] HIC 3228CA AFW TO 1A(1B,1C) SG STOP VLV [] Q1N23MOV3350A [] Q1N23MOV3350B [] Q1N23MOV3350C MDAFWP TO 1A(1B,1C) SG ISO (BOP) [] Q1N23MOV3764A [] Q1N23MOV3764D [] Q1N23MOV3764F [] Q1N23MOV3764E [] Q1N23MOV3764B [] Q1N23MOV3764C

Step 8 continued on next page.

____Page Completed

Step **Action/Expected Response** **Response NOT Obtained**

8.2 WHEN SG narrow range level greater than 31%(48%), THEN maintain SG narrow range level 31%-65%(48%-65%).

8.2.1 Control MDAFWP flow.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3227A in MOD	[]3227B in MOD	[]3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

Step 8 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

8.2.2 Control TDAFWP flow.

TDAFWP FCV 3228
[] RESET reset

TDAFWP
SPEED CONT
[] SIC 3405 adjusted

SG	1A	1B	1C
TDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3228A in MOD	[]3228B in MOD	[]3228C in MOD
TDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3228AA adjusted	[]3228BA adjusted	[]3228CA adjusted

___ 9 **Verify dilution paths isolated.**

9.1 Isolate RMW to boric acid blender.

RMW
TO BLENDER
[] Q1E21FCV114B closed

> 9.1 Locally isolate RMW to CVCS.
>
> RMW TO CVCS ISO
> [] Q1E21V233 closed
> (100' elev. AUX BLDG above hydro test pump)

9.2 Notify Chemistry to secure the zinc addition system (ZAS).

9.3 Verify RMW supp iso to CHG pump suction locked closed (key # Z-58).

[] Q1E21V212 locked closed
(100' elev. AUX BLDG BIT area at suction of CHG pumps)

___ Page Completed

Step	Action/Expected Response	Response NOT Obtained
___10	**Check for uncontrolled cooldown reactivity insertion.** 10.1 Check RCS cold leg temperature - FALLING IN AN UNCONTROLLED MANNER. RCS COLD LEG TEMP [] TR 410 <u>OR</u> 10.2 Check any SG pressure - FALLING IN AN UNCONTROLLED MANNER OR LESS THAN 50 psig.	10 Perform the following. a) Stop any controlled cooldown. b) Proceed to step 13.
___11	**Check Main steam line isolation and bypass valves - CLOSED.**	11 Perform the following. 11.1 Close the main steam line isolation and bypass valves. 1A(1B,1C) SG MSIV - TRIP [] Q1N11HV3369A [] Q1N11HV3369B [] Q1N11HV3369C [] Q1N11HV3370A [] Q1N11HV3370B [] Q1N11HV3370C 1A(1B,1C) SG MSIV - BYPASS [] Q1N11HV3368A [] Q1N11HV3368B [] Q1N11HV3368C [] Q1N11HV3976A [] Q1N11HV3976B [] Q1N11HV3976C 11.2 <u>IF</u> any MSIV fails to close <u>THEN</u> place the associated test switch to the TEST position.

SG	1A	1B	1C
1A(1B,1C) SG MSIV - TEST Q1N11HV	[]3369A/70A	[]3369B/70B	[]3369C/70C

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___12	**Check SGs not faulted.**	
12.1	Check no SG pressure - FALLING IN AN UNCONTROLLED MANNER OR LESS THAN 50 psig.	12.1 Isolate faulted SG(s) using ATTACHMENT 1.
___13	**Check core exit T/Cs - LESS THAN 1200°F.**	13 IF fifth hottest core exit T/C greater than 1200°F AND rising, THEN go to FNP-1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

****

CAUTION: Emergency boration should be continued until adequate shutdown margin is established.

****

Step	Action/Expected Response	Response NOT Obtained
___14	**Check if reactor still critical.**	14 Go to procedure and step in effect.

- Check power range indication - GREATER THAN OR EQUAL TO 5%.

  PR1(2,3,4)
  PERCENT FULL POWER
  [] NI 41B
  [] NI 42B
  [] NI 43B
  [] NI 44B

  OR

- Check any intermediate range startup rate - POSITIVE.

  IR1(2)
  S/U RATE
  [] NI 35D
  [] NI 36D

___Page Completed

86

032G2.4.46 001/1/2/SR NI/C/A 3.6/MOD/FA011005/S/

Following a reactor trip, MCB annunciator FB2, NI-35 LOSS OF COMPENSATING VOLTAGE, is observed to be lit.

At the current time the following indications exist:

- Intermediate range channel NI-35 is indicates 4 X 10^{-10} amps.
- Intermediate range channel NI-36 is indicates 6 X 10^{-11} amps and continues to decrease.
- Source range channels NI31 and NI-32 are de-energized.

Which one of the following statements is correct concerning the source range nuclear instruments and the action that needs to be taken IAW ESP-0.1, Reactor Trip Response?

A. Neither NI-31 or NI-32 can be energized, within one hour a shutdown margin is required to be performed.

B. NI-31 should have automatically energized, and NI-32 will remain de-energized; place NI-31 SR BLOCK/RESET switch to RESET to re-energize NI-31.

C. Both NI-31 and NI-32 should have automatically energized and cannot be energized; within one hour a shutdown margin is required to be performed.

D. Neither NI-31 or NI-32 should have automatically energized; place both SR BLOCK/RESET switch to RESET to re-energize the source range detectors.

A. Incorrect - They can both be energized.

C. Incorrect- Both SR NIs should not have re-energized due to the loss of compensating voltage. They can be energized by going to the REST position. ESP-0.1, step 11 says to Verify SR detectors energized and if not, go to RNO column to w/i one hour verify adequate shutdown margin.

B. Incorrect - Either they both will be energized or they will not. The candidate may believe that the SR in the same channel as the IR will energize based on that IR reading. If the rest switch is paced in RESET, then both SRs will be energized. The candidate may believe each switch goes to the respective SR.

D. Correct - **Neither NI-31 or NI-32 should have automatically energized; place one SR BLOCK/RESET switch to RESET to re-energize the source range detectors.** This will re-energize the SR detectors. The intent of the step in ESP-0.1 is to re-energize the SR detectors so nuclear instumentation can monitor the Rx. Since there is a failure, FB2 has guidance to use the RESET switches to when power is less than 1×10-10 amps on the operating intermediate range channel.

A loss of compensating voltage on IR N-35 has caused this IR channel to remain above P-6 (10E-10 amps) following a reactor trip. In order for SR NI's to auto energize, 2 out of 2 IR channels must be below 10E-10 amps (P-6). Also note that 3 out of 4 PR NI's must be < 10% power (P-10). Since no information is provided regarding PR NI's, they should be assumed to operate normally (below P-10). Since these conditions are not met, neither SR NI will auto energize. The SR NI's can be manually energized by taking either SR block/reset handswitch to 'RESET'. Reference figure 28, OPS-52201D, Excore lesson text.

Source Range Block/Reset Switches

Refer to Figure 28. These three-position switches (BLOCK/NORM/RESET, spring return to NORM) allow the operator to block the source range trips for the train designated. Placing either switch momentarily to the BLOCK position will block that train's trips for both source range channels, provided power is above the P-6 setpoint. Placing the switch to RESET will reset that train's trips if below the P-10 set point. Furthermore, P-6 will automatically reset the trips. Both trains blocked will de-energize the source range high volts. Unblocking either train will re-energize source range high volts.

LOCATION FB2

SUPPLEMENTARY ACTION

Step 5. IF either intermediate range channel fails high, THEN the source range high voltage must be manually energized by placing both source range block-reset switches in the RESET position when power is less than $1\times10_{-10}$ amps on the operating intermediate range channel.

032G2.4.46 Loss of Source Range NI / 7
2.4.46 Ability to verify that the alarms are consistent with the plant conditions.
(CFR: 43.5 / 45.3 / 45.12)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

8. Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Excore Nuclear Instrumentation System including (OPS52201D11):
·Power Range Channels
·Intermediate Range Channels
·Source Range Channels
·Gamma-Metrics Neutron Flux Monitor
·Miscellaneous Control and Indication Drawer
·Audio Count Rate Drawer
·Scaler-Timer Drawer
·Comparator and Rate Drawer

9. Evaluate abnormal plant or equipment conditions associated with the Excore Nuclear Instrumentation System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52201D12).

EXCORE-52201D07 #9

QUESTIONS REPORT

Original

for Plant Systems Questions 6-11-2004

EXCORE-52201D07 009/HLT/LOCT//C/A (LEVEL 2/3) SYS////LOCT/

Following a reactor trip, MCB annunciator FB2, NI-35 LOSS OF COMPENSATING VOLTAGE, is observed to be lit.

At the current time, intermediate range channel N-35 is indicating approximately 4E-10 amps and stable. Intermediate range channel N-36 is indicating approximately 6E-11 amps and continues to decrease. Source range channels N-31 and N-32 are deenergized.

Which one of the following statements is correct concerning the source range nuclear instruments?

A. Both NI-31 and NI-32 should have automatically energized.

B. NI-32 should have automatically energized, and NI-31 will remain deenergized.

C. Both NI-31 and NI-32 will remain deenergized unless operator action is taken.

D. Neither NI-31 or NI-32 can be energized.

A loss of compensating voltage on IR N-35 has caused this IR channel to remain above P-6 (10E-10 amps) following a reactor trip. In order for SR NI's to auto energize, 2 out of 2 IR channels must be below 10E-10 amps (P-6). Also note that 3 out of 4 PR NI's must be < 10% power (P-10). Since no information is provided regarding PR NI's, they should be assumed to operate normally (below P-10). Since these conditions are not met, neither SR NI will auto energize. The SR NI's must be manually energized by taking either SR block/reset handswitch to 'RESET'. Reference figure 28, OPS-52201D, Excore lesson text.

Step	Action/Expected Response	Response NOT Obtained
___11	**Monitor nuclear instrumentation.**	
11.1	WHEN intermediate range indication less than 10^{-10} amps OR BYP & PERMISSIVE P-6 light off, THEN perform the following.	
11.1.1	Verify source range detectors - ENERGIZED.	11.1.1 IF no source range detector energized, THEN within one hour verify adequate shutdown margin using FNP-1-STP-29.1, SHUTDOWN MARGIN CALCULATION (TAVG 547°F), or FNP-1-STP-29.2, SHUTDOWN MARGIN CALCULATION (TAVG <547°F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING).
11.1.2	Plot both source range channels on NR-45 recorder.	
12	**Align secondary components.**	
12.1	Stop both heater drain pumps. HDP [] 1A [] 1B	
12.2	IF started, THEN stop all but one condensate pump. CNDS PUMP [] 1A [] 1B [] 1C	12.2 IF no condensate pumps are started, THEN ensure handswitches are positioned to OFF. CNDS PUMP [] 1A [] 1B [] 1C

Step 12 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___39	**Monitor nuclear instrumentation.**	
39.1	WHEN intermediate range indication less than than 10^{-10} amps OR BYP & PERMISSIVE SOURCE RANGE PERMISSIVE P-6 light off, THEN verify source range detectors energized.	
39.2	WHEN source range detectors energized, THEN plot both source range channels on NR-45 recorder.	
___40	**Consult TSC staff for appropriate cooldown procedure.**	
40.1	Go to FNP-1-ESP-3.1, POST-SGTR COOLDOWN USING BACKFILL.	
	OR	
40.2	Go to FNP-1-ESP-3.2, POST-SGTR COOLDOWN USING BLOWDOWN.	
	OR	
40.3	Go to FNP-1-ESP-3.3, POST-SGTR COOLDOWN USING STEAM DUMP.	

-END-

LOCATION FB2

SETPOINT: 50% of Compensating Voltage Setting

ORIGIN: Intermediate Range Channel N-35

B2
NI 35 LOSS OF COMPENSATING VOLTAGE

PROBABLE CAUSE

1. Loss of 118 VAC Instrument Power to Compensating Voltage Power Supply.
2. Loss of 118 VAC Control Power to Loss of Compensating Voltage Bistable Relay Driver.
3. Compensating Voltage Power Supply malfunction.

AUTOMATIC ACTION

NOTE: This trip will normally be manually blocked after Permissive P-10 is satisfied.

IF the malfunction causes an Intermediate Range trip setpoint to be reached on one out of two detectors, THEN a reactor trip will occur.

IMMEDIATE ACTION

IF BOTH INTERMEDIATE RANGE CHANNELS FAIL ABOVE P-6 BUT BELOW P-10, THEN SUSPEND ALL OPERATIONS INVOLVING POSITIVE REACTIVITY ADDITION AND REFER TO TECHNICAL SPECIFICATION 3.3.1 CONDITION G.

SUPPLEMENTARY ACTION

IF the malfunction does not result in a reactor trip, THEN perform the following:

1. IF an intermediate range channel fails below P-6, THEN restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 setpoint.
2. IF an intermediate range channel fails above P-6 but below P-10, THEN refer to Technical Specifications 3.3.1 Condition F.

LOCATION FB2

SUPPLEMENTARY ACTION

3. IF an intermediate range channel fails ≥ P-10, THEN POWER OPERATION may continue.
4. Carefully monitor the remaining nuclear instrument channels.
5. IF either intermediate range channel fails high, THEN the source range high voltage must be manually energized by placing both source range block-reset switches in the RESET position when power is less than 1×10^{-10} amps on the operating intermediate range channel.
6. IF both intermediate range channels fail AND power is ≥ P-10, THEN power operation may continue.
7. Refer to Technical Specifications section 3.3.1 for limiting conditions for operation.
8. Notify appropriate plant personnel to investigate and correct the cause of the alarm.

References: A-177100, Sh. 277; U-260268; U266647 PLS Document; Technical Specifications

Main Control Board Controls

Source Range Block/Reset Switches

Refer to Figure 28. These three-position switches (BLOCK/NORM/RESET, spring return to NORM) allow the operator to block the source range trips for the train designated. Placing either switch momentarily to the BLOCK position will block that train's trips for both source range channels, provided power is above the P-6 setpoint. Placing the switch to RESET will reset that train's trips if below the P-10 set point. Furthermore, P-6 will automatically reset the trips. Both trains blocked will de-energize the source range high volts. Unblocking either train will re-energize source range high volts.

Intermediate Range Block Switches

Refer to Figure 28. These two-position switches (BLOCK/NORM, spring return to NORM) allow the operator to block the intermediate range reactor trip and rod stop for the train designated, when greater than P-10. Power less than P-10 will reset the trip and rod stop.

Power Range Block Switches

Refer to Figure 28. These two-position switches (BLOCK/NORM, spring return to NORM) allow the operator to block the power range high flux low setpoint reactor trip for the train designated when greater than P-10. When power drops below the P-10 setpoint, the trip block will automatically reset.

NR45 Selector Switches

These two switches control the input to the NR45 recorder. The switches enable the operator to select any two of the NI channels and/or the ΔI indications to be displayed on the two pens of the NR45 recorder.

ACTIONS FOR INDICATED POWER LEVEL EXCEEDING 102%

Indicated reactor power level exceeding 102% on either unit requires that a Condition Report be submitted.

If either Unit exceeds 102% reactor power, a Condition Report should be generated prior to the end of the shift during which the power excursion occurred. Operations Management will

FARLEY JANUARY 2005 EXAM

EXAM 50-348 AND 50-364/2005-301
JANUARY 10 - 14, 2005
JANUARY 18, 2005 (written)

DRAFT SRO WRITTEN EXAM

OpsNis033
Nuclear Instrumentation Trip Signals, Permissives, and Blocks
Figure 28

INTENTIONALLY OMITTED

PER SISP REVIEW

87. 054AA2.04 001/1/1/AFW/C/A 4.3/NEW/FA011005/S/CVR

Unit 1 has tripped due to a loss of all Main Feed.

- FRP-H.1, Response to Loss of Secondary Heat Sink, is in progress.
- Efforts to establish AFW flow or to start up a SGFP initially failed.
- The TDAFWP tripped when it Auto started.
- The UO adjusted all three TDAFWP FCVs potentiometers to the '0' DEMAND position.
- TDAFWP FCV 3228 RESET light is LIT.
- The UO ran the TDAFWP speed control to SLOW on the potentiometer.

While performing the step for Steam Generator pressure reduction, the SSS-plant reports that he has successfully reset the TDAFW pump trip throttle valve.
The Unit Operator takes the handswitch for TDAFWP STM SUPP FROM 1C SG HV3235B to START.

Which one of the following will be the correct position of the TDAFWP FCVs after the pump start, the pump speed and the earliest that the crew can exit FRP-H.1?

A✓ OPEN; approximately 2000 rpm; As soon as total AFW flow to the SGs is verified to be greater than 395 gpm.

B. OPEN; approximately 3980 rpm; As soon as total AFW flow to the SGs is verified to be greater than 395 gpm **AND** at least one SG level is greater than 31% NR level.

C. CLOSED; approximately 2000 rpm; As soon as total AFW flow to the SGs is verified to be greater than 395 gpm **AND** at least one SG level is greater than 31% NR level.

D. CLOSED; approximately 3980 rpm; As soon as total AFW flow to the SGs is verified to be greater than 395 gpm.

SOP-0.8 FRP-H.1

NOTE: Step 5.3.3.3 is a continuing action until step 12 is reached.

5.3.3.3 WHEN any available AFW pump started, THEN return to step 5.4.

5.3.3.4 Proceed to step 6.

A. Correct. **OPEN; approximately 2000 rpm; As soon as total AFW flow to the SGs is verified to be greater than 395 gpm.**

The FCVs go full open on an auto open signal regardless of POT setting. The TDAFWP will go to max speed unless the pot is run back, which in this case it is. If the pot is at a min speed, then the TDAFWP will go to a min speed of approx 2000 rpm.

H.1 directs the operator in a continuing action step (see note prior to step 5.3.3.3) to continue efforts to establish AFW flow while continuing in the procedure, when you get AFW flow > 395 gpm, then go to procedure and step in effect.

B. Incorrect. OPEN and speed is Incorrect. Procedure does not say Either AFW Flow >395 gpm **OR** NR SG Lvl > 31% {48} is an adequate heat sink. If AFW flow is established, the procedure does not require a specific SG NR level.

C. Incorrect. FCVs incorrect, speed correct, Procedure does not say Either AFW Flow >395 gpm **OR** NR SG Lvl > 31% {48} is an adequate heat sink. If AFW flow is established, the procedure does not require a specific SG NR level.

D. Incorrect. FCVs incorrect, speed incorrect,correct reasoning.

054AA2.04 Loss of Main Feedwater / 4 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):(CFR: 43.5 / 45.13)
AA2.04 – proper operation of AFW pumps and regulating valves

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. Evaluate abnormal plant or equipment conditions associated with the Auxiliary Feedwater System (AFW) and determine the integrated plant actions needed to mitigate the consequence of abnormality (OPS-52102H02).

Step	Action/Expected Response	Response NOT Obtained

5.3.3 IF any AFW pump started, THEN proceed to step 5.4. IF NOT perform the following.

5.3.3.1 Restore the following components to REMOTE.

1A MDAFWP
[] Q1N23P001A (A-HSDP)

1B MDAFWP
[] Q1N23P001B (C-HSDP)

TDAFWP STM SUPP FROM 1B(1C) SG
[] Q1N12HV3235A/26 (F-HSDP)
[] Q1N12HV3235B (F-HSDP)

5.3.3.2 IF necessary, THEN dispatch an operator locally to TDAFWP to assist with restoration of TDAWFP.

NOTE: Step 5.3.3.3 is a continuing action until step 12 is reached.

5.3.3.3 WHEN any available AFW pump started, THEN return to step 5.4.

5.3.3.4 Proceed to step 6.

Step 5 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

5.4 Verify at least one flow path to at least one SG - ALIGNED.

SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3227A in MOD	[]3227B in MOD	[]3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted
TDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3228A in MOD	[]3228B in MOD	[]3228C in MOD
TDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3228AA adjusted	[]3228BA adjusted	[]3228CA adjusted
AFW TO 1A(1B,1C) SG STOP VLV Q1N23MOV	[]3350A open	[]3350B open	[]3350C open
MDAFWP TO 1A(1B,1C) SG ISO Q1N23MOV (BOP)	[]3764A open []3764E open	[]3764D open []3764B open	[]3764C open []3764F open

5.4 Direct personnel to perform the following.

- Direct personnel to establish flow path to affected SGs from hot shutdown panel. (121 ft. AUX BLDG HSDP rooms)

<u>OR</u>

- Direct personnel to establish flow to affected SGs. (127 ft. AUX BLDG main steam valve room)

Step 5 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 5.5 is a continuing action until step 12 is reached.

Step	Action/Expected Response	Response NOT Obtained
5.5	Check total AFW flow to SGs greater than 395 gpm.	5.5 Perform the following. 5.5.1 Continue efforts to establish AFW flow. 5.5.2 Proceed to Step 6.
5.6	Go to procedure and step in effect.	
___ 6	**Stop all RCPs.** RCP [] 1A [] 1B [] 1C	
7	**Try to establish main feedwater flow to intact SGs with one SGFP.**	
7.1	Verify at least one CNDS PUMP - STARTED.	7.1 Perform the following. a) Continue efforts to start a CNDS PUMP. b) Proceed to step 11.
7.1.1	Have back up cooling aligned to condensate pumps	
7.2	Verify an EH pump running.	7.2 Proceed to step 9. OBSERVE CAUTION PRIOR TO STEP 9.
7.3	Check all intact SG pressures - GREATER THAN 540 psig.	7.3 Proceed to Step 9. OBSERVE CAUTION PRIOR TO STEP 9.
7.4	Check condenser - AVAILABLE. BYP & PERMISSIVE COND AVAIL [] C-9 status light lit	7.4 Proceed to Step 9. OBSERVE CAUTION PRIOR TO STEP 9.

Step 7 continued on next page.

___Page Completed

88. 060AA2.05 001/1/2/GAS RELEASE/C/A 4.2/NEW/FA011005/S/CVR

A Waste Gas Decay Tank containing 30 psig of radioactive gas is leaking into the Unit 1 Aux Building due to a through wall pipe leak. The following alarms are in:

- R-14, PLANT VENT GAS.
- R-22, PLANT VENT STACK.
- R-35A & B, CONTROL ROOM VENTS.

Which one of the following does **NOT** automatically occur **AND** must be performed per LD4, R-35A HI ALARM, and ND4, R-35B HI ALARM, to protect the control room team?

A. Secure one Auxiliary Building Main Exhaust Fan.

B✓ Start one of the Control Room Pressurization units.

C. Close the Control Room Utility Exhaust Fan Suction Damper.

D. Shift the Technical Support Center HVAC into the recirc mode.

A. Incorrect. This is not an auto action nor is it required to be done by ARP-3.1. It would slow the release from the Aux. Building down some, but would also concentrate the airborne activity in the Aux. Building. Protecting the Control Room personnel by isolating the Control Room is necessary, slowing the release rate from the Aux. Building is not.

B. Correct. **Start one of the Control Room Pressurization units.**
This is directed by ARP-3.1 & 3.2, LD4 & ND4 and does not automatically occur. This helps insure that the airborne activity does not get into the Control Room.

C. Incorrect. This is an auto action per ARP-3.1 & 3.2, LD4 & ND4 and are directed to be verified, but do not have to be manually performed unless they malfunction and do not occur. This should have occurred, and it is not stated that it did not occur automatically as required. Knowledge of auto actions and procedure requirements is tested by this question.

D. Incorrect. This is an auto action per ARP-3.1 & 3.2, LD4 & ND4 and are directed to be verified, but do not have to be manually performed unless they malfunction and do not occur. This should have occurred, and it is not stated that it did not occur automatically as required. Knowledge of auto actions and procedure requirements is tested by this question.

060AA2.05 Accidental Gaseous Radwaste Rel. / 9 - Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: (CFR: 43.5 / 45.13) AA2.05 – that the automatic safety actions have occurred as a result of a high ARM system signal.

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Evaluate abnormal plant or equipment conditions associated with the Waste Gas System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52106B02).

Evaluate abnormal plant or equipment conditions associated with the Control Room Ventilation System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52107C02).

LOCATION LD4

SETPOINT: Variable, as per RCP-252

D4	
	R-35A HI ALARM

ORIGIN: Radiation Monitor Cabinet channel R-35A

PROBABLE CAUSE

1. High radiation level in the intake of the Computer Room AHU.
2. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

AUTOMATIC ACTION

1. Closes Computer Room AHU dampers HV 3622, 3624, and 3626.
2. Closes Control Room Utility Exhaust Fan suction damper HV 3628.
3. Shifts Technical Support Center HVAC into its recirculation mode of operation.

OPERATOR ACTION

1. Verify Computer Room AHU dampers closed.
2. Verify Technical Support Center HVAC aligned in recirculation mode PER FNP-0-SOP-56.1, TECHNICAL SUPPORT CENTER HVAC SYSTEM, step 4.3.3.
3. Determine the validity of the high activity indication as follows:
   a. Verify that the instrument is aligned for normal operation and is functioning properly.
   b. Sample or survey the affected system or area as required.
4. Verify both control room doors closed.
5. Manually start one of the redundant Control Room Pressurization Units and one of the redundant Control Room Recirculation Units per FNP-0-SOP-56.0, CONTROL ROOM HVAC SYSTEM.
4. Stop the Control Room Utility Exhaust Fan.
5. Determine the source or cause of the high activity and correct or isolate as required.

LOCATION LD4

OPERATOR ACTION (continued)

6. WHEN the alarm is cleared, THEN:
   a. Return Computer Room HVAC to normal per FNP-0-SOP-56.0, CONTROL ROOM HVAC SYSTEM, section 4.1.
   b. Open S-CRV-HV-3628 (QSV49V003A) and start the Control Room Utility Exhaust Fan.
   c. Verify TSC HVAC in normal per FNP-0-SOP-56.1, TECHNICAL SUPPORT CENTER HVAC SYSTEM, section 4.1.

References: MDFD #89-2053; D-181672; U-258972; U-258973; D-205012; D-205014, Sh. 2; D-177373, Sh. 2; D-177394, Sh. 1 & 2; D-207270, Sh. 1-3

LOCATION FH1

H1
RMS HI-RAD

SETPOINT: 1. Variable, as per FNP-1-RCP-252

ORIGIN: Any of the below listed Area, Process or Gaseous and Particulate Monitors: R01, R02, R03, R04, R05, R06, R07, R08, R10, R11, R12, R13, R14, R15, R16, R17A R17B, R18, R19, R20A, R20B, R21, R22, R23A, R23B, R26A, or R26B.

PROBABLE CAUSE

1. High Radiation Level in the System, Area or at the Component monitored.
2. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

AUTOMATIC ACTION

1. The following actions will occur if a High Radiation Alarm is actuated on the associated Radiation Monitor.
   a) R14: (Plant Vent Gas) closes Waste Gas Release Valve 1-GWD-HV-014.
   b) R16: (Boron Recycle System) diverts 1-CVC-RCV-016 Recycle Evaporator discharge from Reactor Makeup Water System to the Recycle Evaporator Demineralizer.
   c) R17A or B: (Component Cooling Water) closes Q1P17RCV3028 CCW SRG TANK VENT.
   d) R18: (Liquid Waste Processing) closes Liquid Waste Release Valve 1-LWP-RCV-018.
   e) R19: (Steam Generator Blowdown) isolates Steam Generator Blowdown Sample Lines.
   f) R23A :(Steam Generator Blowdown Processing) closes 1-BD-FCV-1152 S/G Blowdown Heat Exchanger Discharge Valve.
   g) R23B :(Steam Generator Blowdown Processing) closes 1-BD-RCV-023B Dilution Discharge Valve.

2. ARDA will automatically start for the conditions listed in the following Note.

NOTE: **ARDA will automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on either unit and use the latest 15 minute average monitor value to perform the calculations:**

Plant Vent Stack Monitors R29 (SPING)	
Noble Gas	**4.44e-4 μc/ml**
Iodine	**1.20e-6 μc/ml**
Particulate	**4.00e-5 μc/ml**
Steam Jet air Ejector R15C	**27 mr/hr**
TDAFW Exhaust R60D	**38 mr/hr**
Steam Generator A R60A	**38 mr/hr**
Steam Generator B R60B	**38 mr/hr**
Steam Generator C R60C	**38 mr/hr**

ARDA will also automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on either unit. The ARDA system will use the plant Vent stack SPING latest 15 minute average monitor value to perform the calculations when these monitors activate the system:

Plant Vent Stack Monitors	
Gas monitor R 14	**13,000**
Gas monitor R 21	**1800**
Particulate monitor R 22	**156**

IMMEDIATE ACTION

1. CHECK INDICATIONS ON RADIATION MONITORING SYSTEM CONSOLE AND DETERMINE WHICH RADIATION MONITOR CHANNEL INDICATES HIGH ACTIVITY.
2. INSURE THAT ANY AUTOMATIC ACTIONS, ASSOCIATED WITH THE ALARMED CHANNEL, HAVE OCCURRED.

SUPPLEMENTARY ACTION

1. Perform the following general actions as appropriate.

   1.1. Determine the source or cause of the high activity and correct or isolate as required.

   1.2 Determine the validity of the high activity indication as follows:

      1.2.1 Verify that the instrument is aligned for normal operation and is functioning properly.

      1.2.2 IF a known problem exists such that the detector is saturated, THEN momentarily pull the affected detector's fuses (located on the front of the drawer) to clear the condition.

      1.2.3 Sample or survey the affected system or area as required. **{CMT 0008755}.**

   1.3 Do not allow personnel to enter the affected area without the approval of the Health Physics Department.

   1.4 IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.3, 3.4.15 and TRM TR 13.3.4.

   1.5 IF high activity indication of RCS leakage is present AND accompanied by either decreasing pressurizer level, OR decreasing VCT level, THEN go to FNP-1-AOP-1.0, RCS LEAKAGE.

   1.6 IF high activity indication of Steam Generator Tube Leakage is present, THEN go to FNP-1-AOP-2.0, STEAM GENERATOR TUBE LEAKAGE.

   1.7. IF ARDA activated and not required, THEN have counting room stop the automated dose assessment per FNP-0-EIP-9.1, AUTOMATED DOSE ASSESSMENT METHOD.

   1.8 Refer to FNP-0-ACP-16, PRIMARY-TO-SECONDARY STEAM GENERATOR TUBE LEAK PROGRAM, for additional guidance on tube leakage or problems with radiation monitors R-70A, B & C.

   1.9 WHEN radiation levels have decreased below alarm setpoint, THEN reset the appropriate HI radiation alarm on the RAD monitor drawer.

NOTE: For ease of reference the radiation monitor number corresponds to the associated section 2 step number following the decimal in (e.g. the associated actions for R-13 are found in step 2.13).

2. In addition to the general actions perform supplementary steps indicated in the "ACTIONS" column of the following table. (Some of the radiation monitors included in the table do not input into this alarm but are included for reference).

RADIATION MONITOR REFERENCE TABLE

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-1A	Control Room (Unit I Panel)	Area	G-M (W)		Perform Step 2.1
R-1B	Technical Support Center (Unit II Panel) R-1B	Area	G-M (W)		No input to this alarm
R-2	Containment (155' elev)	Area	G-M (W)		Perform Steps 2.2
R-3	Radiochemistry Lab (AB 139')	Area	G-M (W)		Perform Step 2.3
R-4	#3 Charging Pump (AB 100')	Area	G-M (W)		Perform Step 2.4
R-5*	Spent Fuel Pool Room (AB 155')	Area	G-M (W)		Perform Steps 2.5
R-6	Sampling Room (AB 139')	Area	G-M (W)		Perform Step 2.6
R-7	In-core NIS Area (CTMT 129', near Seal Table)	Area	G-M (W)		Perform Steps 2.7
R-8	Drumming Station (AB 155')	Area	G-M (W)		Perform Step 2.8
R-9	SG Sample Panel (Unit II Panel) (AB 139')	Area	G-M (W)		No input to this alarm
R-10	Penetration Room Filtration Discharge (AB 155')	APD	Scint. (Victoreen)		Perform Step 2.10
R-11*	Containment Atmosphere (AB 121')	APD	Scint. (Victoreen)		Perform Step 2.11

*Technical Specification related

LOCATION FH1

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-12*	Containment Atmosphere (AB 121')	Gas	G-M (W)		Perform Step 2.12
R-13	Waste Gas Compressor Suction (AB 100' WGC Valve Room)	Gas	G-M (W)		Perform Step 2.13
R-14 ODCM	Plant Vent Stack (AB Roof)	Gas	G-M (W)	Closes HCV-14	Perform Step 2.14
R-15A ODCM	Condenser Air Ejector Discharge Header (TB 155')	Gas	G-M		Perform Step 2.15A
R-15B*	Condenser Air Ejector (Intermediate Range) (TB 189')	Gas	G-M (Eberline)		Perform Step 2.15B
R-15C*	Condenser Air Ejector (High Range) (TB 189')	Gas	Ion Chamber (Eberline)		Perform Step 2.15B
R-16	Recycle Evaporator Distillate Line (AB 139')	Liquid	Scint. (W)	Diverts RCV-16 from RMW system to the Recycle Evaporator Demineralizer	Perform Step 2.16
R-17A	Component Cooling Water (CCW Hx Room)	Liquid	Scint. (W)	Closes CCW surge tank vent (RCV-3028)	Perform Step 2.17
R-17B	Component Cooling Water (CCW Hx Room)	Liquid	Scint.	Closes CCW surge tank vent (RCV-3028)	Perform Step 2.17
R-18 ODCM	Waste Monitor Tank Pump Discharge (AB 121' at the Batching Funnel)	Liquid	Scint. (W)	Closes RCV-18	Perform Step 2.18
R-19	Steam Generator Blowdown/Sample (AB 139')	Liquid	Scint. (W)	Isolates sample lines 3328, 3329, 3330	Perform Step 2.19
R-20A	Service Water from Containment Coolers A and B (AB 121' BTRS Chiller Room)	Liquid	Scint. (W)		Perform Step 2.20

*Technical Specification related

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-20B	Service Water from Containment Coolers C and D (AB 121')	Liquid	Scint. (W)		Perform Step 2.20
R-21	Plant Vent Stack (AB 155')	APD	Scint. (Victoreen)		Perform Step 2.21
R-22 ODCM	Plant Vent Stack (AB 155')	Gas	G-M (W)		Perform Step 2.22
R-23A	SG Blowdown Surge Tank Inlet (AB 130')	Liquid	Scint. (W)	Closes FCV-1152	Perform Step 2.23
R-23B ODCM	SG Blowdown Surge Tank Discharge (AB 130')	Liquid	Scint. (W)	Closes RCV-23B	Perform Step 2.23
R-24A*	Containment Purge (AB 155')	Gas	Scint.	Closes containment purge supply & exhaust dampers 2866C & 2867C and 3198A & D	No input to this alarm
R-24B*	Containment Purge (AB 155')	Gas	Scint. (Victoreen)	Closes valves: 2866D &2867D, 3196, 3197, 3198B & C	No input to this alarm
R-25A/ R-25B*	Spent Fuel Pool Ventilation (AB 184')	Gas	Scint. (Victoreen)	Trip fuel bldg supply and exhaust fans; closes SFP HVAC supply and exhaust dampers; starts associated trains of penetration room filtration.	No input to this alarm
R-26A	Recycle Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)		Perform Step 2.26
R-26B	Waste Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)		Perform Step 2.26

*Technical Specification related

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-27A*	Containment (High Range)	Area	Ion Chamber (Victoreen)		No input to this alarm
R-27B*	Containment (High Range)	Area	Ion Chamber (Victoreen)	Alert alarm starts @ 10R/hr recorder	No input to this alarm
R-28	Condenser Air Ejector Off-Line Sample (TB 155')	None	None	Provide off line sample capability	No input to this alarm
R-29A	Plant Vent Stack	None	None	Provide off line sample capability	No input to this alarm
R-29B* ODCM	Plant Vent Stack	SPING-4	Scint./G-M (Eberline)		No input to this alarm
R-30A	Radwaste Area Vents El 100'	APD	Scint. (Victoreen)		No input to this alarm
R-30B	Radwaste Area Vents El 100'	Gas	Scint. (Victoreen)		No input to this alarm
R-31	Radwaste Area Vents El 121'	APD	Scint. (Victoreen)		No input to this alarm
R-32	Radwaste Area Vents El 139'	APD	Scint. (Victoreen)		No input to this alarm
R-33	Radwaste Area Vents El 155'	APD	Scint. (Victoreen)		No input to this alarm
R-34	Access Control Area (Unit 1 only)	APD	Scint. (Victoreen)		No input to this alarm

*Technical Specification related

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-35A* / R-35B*	Control Room Vents (Control Room Ventilation Room)	Gas	Scint. (Victoreen)	Isolates control room supply and return from computer room AHU, closes utility exhaust dampers, and shifts TSC ventilation to recirc (filtration mode)	No input to this alarm
R-60A, B, & C*	Main Steam Relief Valve Exhaust Plume (Roof above MSVR)	Gas	Ion chamber (Victoreen)		No input to this alarm
R-60D*	Turbine-Driven Auxiliary Feedwater Pump Exhaust (Roof above MSVR)	Gas	Ion chamber (Victoreen)		No input to this alarm
R-66A through Bldg F	Low Level Rad Waste Building	Area	Ion chamber (Eberline)	Alert (10 mr/hr) - LLRW vent secures High (100 mr/hr) - Actuates magenta flashing lights throughout interior and on roof of LLRW Bldg	No input to this alarm
R-67	Containment Atmosphere	None	None		No input to this alarm
R-68	Vent Stack Sampling	None	None		No input to this alarm
R-69	Containment Purge Sampling	None	None		No input to this alarm
R-70A, B, C	Main Steam Valve Room On Each Steam Line Nitrogen 16 (N-16) Monitor	Process	NaI Scint (Merlin- Gerin)		No input to this alarm

*Technical Specification related

2.1 IF **R-01A** alarms and a high radiation condition exists, THEN perform the following (1999 RMS SSSA Observation RMS-OPS-01):
   - 2.1A.1 Validate the R-01A alarm.
   - 2.1A.2 IF the R-01A alarm is valid, THEN obtain HP input on control room habitability. **{CMT 0008755}**
   - 2.1A.3 IF HP determines the control room is uninhabitable, THEN implement FNP-1-AOP-28.0, CONTROL ROOM INACCESSIBILITY.

2.2 IF **R-2** in alarm THEN perform the following:
   - 2.2.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
   - 2.2.2 Have all personnel evacuate the affected area.
   - 2.2.3 IF either R-27A OR R-27B is inoperable, THEN within 72 hours initiate the preplanned alternate method of monitoring CTMT radiation by having Health Physics perform FNP-1-STP-818, RE27A & B CONTINGENCY MONITORING. (Reference Tech. Spec. 3.3.3)

2.3 IF **R-3** in alarm THEN perform the following:
   - 2.3.1 Announce the affected area on the public address system.
   - 2.3.2 Have all personnel evacuate the affected area.

2.4 IF **R-4** in alarm THEN perform the following:
   - 2.4.1 Announce the affected area on the public address system.
   - 2.4.2 Have all personnel evacuate the affected area.

2.5 IF **R-5** alarms and a refueling accident is possible, THEN refer to FNP-1-AOP-30, REFUELING ACCIDENT.

2.6 IF **R-6** in alarm THEN perform the following:
   - 2.6.1 Announce the affected area on the public address system.
   - 2.6.2 Have all personnel evacuate the affected area.

2.7 IF **R-7** in alarm THEN perform the following:
   2.7.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
   2.7.2 Have all personnel evacuate the affected area.
   2.7.3 IF either R-27A OR R-27B is inoperable, THEN within 72 hours initiate the preplanned alternate method of monitoring CTMT radiation by having Health Physics perform FNP-1-STP-818, RE27A & B CONTINGENCY MONITORING. (Reference Tech. Spec. 3.3.3)

2.8 IF **R-8** in alarm THEN perform the following:
   2.8.1 Announce the affected area on the public address system.
   2.8.2 Have all personnel evacuate the affected area.

2.9 Step deleted

2.10 IF **R-10** alarms and high activity in the penetrations rooms is possible, THEN consider placing penetration room filtration in service using FNP-1-SOP-60 PENETERATION ROOM FILTRATION SYSTEM.

2.11 IF **R-11** alarms, THEN perform the following:
   2.11.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
   2.11.2 IF high activity in containment is possible, THEN consider securing containment purge / minipurge (refer to FNP-1-SOP-12.2 CONTAINMENT PURGE AND PREACCESS FILTRATION SYSTEM.
   2.11.3 IF RCS leakage is possible then perform actions of FNP-1-AOP-1.0, RCS LEAKAGE

2.12 IF **R-12** alarms, THEN perform the following:
   2.12.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
   2.12.2 IF high activity in containment is possible, THEN consider securing containment purge / minipurge (refer to FNP-1-SOP-12.2 CONTAINMENT PURGE AND PREACCESS FILTRATION SYSTEM.
   2.12.3 IF RCS leakage is possible then perform actions of FNP-1-AOP-1.0, RCS LEAKAGE

2.13 IF **R-13** alarms, THEN refer to FNP-1-SOP-51, WASTE GAS SYSTEM for potential problems with the waste gas system.

2.14 IF **R-14** alarms, THEN perform the following:

2.14.1 IF high effluent activity is possible, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.14.2 Refer to FNP-1-SOP-51, WASTE GAS SYSTEM for potential problems with the waste gas system.

2.15A IF **R-15** alarms AND remains above the alarm setpoint (not a momentary spike), THEN perform the following:

2.15A.1 IF high effluent activity is possible, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.15A.2 Notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate.

2.15A.3 Notify the Operations Shift Superintendent

2.15B IF **R-15B** OR **R-15C**, alarms AND remains above the alarm setpoint (not a momentary spike), THEN notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate.

2.16 IF **R-16** alarms, THEN refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

2.17 IF **R-17A** OR **R-17B** alarms, THEN monitor CCW pump operation while the CCW surge tank vents are closed.

NOTE:	**IF CCW surge tank vents are closed for reasons other than an actual high radiation alarm, THEN with Shift Supervisor concurrence, the CCW surge tank vents should be cycled once every shift (eight hours) and documented in AutoLog.**

2.18 IF **R-18** alarms with high liquid effluent activity possible, THEN verify any liquid waste release is secured and refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

2.19 IF **R-19** alarms AND remains above the alarm setpoint (not a momentary spike), THEN notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate. Refer to FNP-1-SOP-45.0, RADIATION MONITORING SYSTEM for guidance in sampling steam generators with R-19 in alarm.

2.20 **R-20A** AND **R-20B** would not normally be expected to indicate high radioactivity since SW pressure is higher than the opposite side of the 'coolers' upstream of R-20A and R-20B. Request Counting Room to sample the SW effluent. IF high activity is confirmed, THEN investigate for a possible cross system connection to the SW system. **{CMT 0005153}**

2.21 IF **R-21** alarms, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.22 IF **R-22** alarms, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.23 IF **R-23A** OR **R-23B** alarms AND remains above the alarm setpoint (not a momentary spike), THEN perform the following:

2.23.1 Notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate

2.23.2 Contact the RAD man to verify blowdown secured.

2.24 Step deleted

2.25 Step deleted

2.26 IF **R-26A** OR **R-26B** alarms, THEN refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

References: A-177100, Sh. 306; U-260841; D-181751; D-181752; D-181753; FSAR, Section 11.4

89. 064G2.2.22 001/2/1/T.S. 3.8.1/C/A 4.1/NEW/FA011005/S/CVR

Both Units are at 100% power with the following conditions:

- All Big emergency diesel generators are INOPERABLE pending resolution of an industry-wide concern with the Colt-Pielstick diesel fuel injectors.
- All required surveillances are current.

Concerning both units, which one of the following has correct times that will clear the LCO(s) in effect at this time and not require either unit to ramp to Mode 3?

References Provided

A. Restore 1-2A DG, 1B DG and 2B DG to OPERABLE status in 7 days.

B. Restore 1B and 2B DG to OPERABLE status in 12 hours, then restore 1-2A DG to OPERABLE status in 48 hours.

C✓ Restore 1-2A DG to OPERABLE status in 5 hours, then restore 1B and 2B DG to OPERABLE status in 6 days.

D. Restore 1B DG to OPERABLE status in 1 hour, then restore 2B DG in 13 hours and 1-2A DG to OPERABLE status in 24 hours.

T. S. 3.8.1 PROVIDED thru 3.8.1- 99 - Do not include surveillance requirements

A. Incorrect. This will meet the requirements for condition B but does not meet condition E. After 8 hours Condition F will need to be entered and Mode 3 in the next 6 hours.

B. Incorrect. This meets the requiements of condition E, the last condition if 1C DG was Inop. This is not the TS they are in and so it does not apply. Condition E is required to be met w/i 8 hours which is to have either 1-2A DG or 1B and 2B DG OPERABLE in 8 hours.

C. Correct. **Restore 1-2A DG to OPERABLE status in 5 hours, then restore 1B and 2B DG to OPERABLE status in 6 days.**
This meets Condition E requirements of getting either 1-2A DG or 1B and 2B DG OPERABLE w/i 8 hours, then Condition B comes in to play and you now have 10 days to get the other set on both units OPERABLE.

D. Incorrect. Getting 1B DG in 1 hour is correct for Unit 1 but 2B DG has to be back in the next 8 hours since 1-2A DG does not come back until 24 hours to prevent Unit 2 from being required to ramp to mode 3.

064G2.2.22 Emergency Diesel Generator
2.2.22 Knowledge of limiting conditions for operations and safety limits.
(CFR: 43.2 / 45.2)

(2) Facility operating limitations in the technical specifications and their bases.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Diesel Generator and Auxiliaries System (OPS52102I01).
• 3.8.1 AC Sources – Operating
• 3.8.2 AC Sources – Shutdown
• 3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources — Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generator (DG) sets capable of supplying the onsite Class 1E power distribution subsystem(s); and

c. Automatic load sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	2 hours AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status.	72 hours AND 13 days from discovery of failure to meet LCO
B. One DG set inoperable.	-----------------NOTE----------------- LCO 3.0.4 is not applicable when only one of the three DGs is inoperable. --		
	B.1	Perform SR 3.8.1.1 for the required offsite circuit(s).	2 hours AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG set inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG set is not inoperable due to common cause failure.	24 hours
	OR		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.2	Perform SR 3.8.1.6 for OPERABLE DG set.	24 hours
	AND		
	B.4	Restore DG set to OPERABLE status.	10 days one diesel set AND 13 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. <u>AND</u> One DG set inoperable.	----------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems — Operating," when Condition D is entered with no AC power source to any train. ------------------------------------	
	D.1 Restore required offsite circuit to OPERABLE status.	24 hours
	<u>OR</u>	
	D.2 Restore DG set to OPERABLE status.	24 hours
E. Two DG sets inoperable.	E.1 Restore one DG set to OPERABLE status.	2 hours if all three DGs are inoperable <u>OR</u> 8 hours if DG 1-2A and DG 1(2)B are inoperable <u>OR</u> 24 hours if DG 1C and DG 1(2)B are inoperable
F. Required Action and associated Completion Time of Condition C or E not met.	F.1 Be in MODE 3.	6 hours

2B, 1C, 1/2A

1-2A + 2B

1C + 2B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One automatic load sequencer inoperable.	G.1 Restore automatic load sequencer to OPERABLE status.	12 hours
H. Required Action and associated Completion Time of Condition A, B, D, or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 5.	6 hours 36 hours
I. Three or more required AC sources inoperable.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-------------------------NOTES------------------------- 1. Performance of SR 3.8.1.6 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.6 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

SURVEILLANCE REQUREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.6. ---------- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2700 kW and ≤ 2850 kW for the 2850 kW DG and ≥ 3875 kW and ≤ 4075 kW for the 4075 kW DGs.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 900 gal of fuel oil for the 4075 kW DGs and 700 gal of fuel oil for the 2850 kW DG.	31 days
SR 3.8.1.5	Verify the fuel oil transfer system operates to transfer fuel oil from storage tank to the day tank.	31 days
SR 3.8.1.6	-----NOTE----- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves in ≤ 12 seconds, voltage ≥ 3952 V and frequency ≥ 60 Hz.	184 days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.7	-------------------------------NOTE------------------------------- This Surveillance shall not be performed in MODE 1 or 2. -- Verify manual transfer of AC power sources from the normal offsite circuit to the alternate required offsite circuit.	18 months
SR 3.8.1.8	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the speed is ≤ 75% of the difference between nominal speed and the overspeed trip setpoint; and b. Following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V.	18 months

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	-----NOTES----- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 12 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 3740 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-----NOTE----- All DG starts may be preceded by prelube period. -----	
	Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and:	18 months
	a. In ≤ 12 seconds after auto-start and during tests, achieves voltage ≥ 3952 V;	
	b. In ≤ 12 seconds after auto-start and during tests, achieves frequency ≥ 60 Hz;	
	c. Operates for ≥ 5 minutes and maintains a steady state generator voltage and frequency of ≥ 3740 V and ≤ 4580 V and ≥ 58.8 Hz and ≤ 61.2 Hz;	
	-----NOTE----- SR 3.8.1.10.d and e shall not be performed in MODE 1 or 2. -----	
	d. Permanently connected loads remain energized from the offsite power system; and	
	e. Emergency loads are energized from the offsite power system.	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.11	Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus and/or an actual or simulated ESF actuation signal except: a. Engine overspeed; b. Generator differential current; and c. Low lube oil pressure.	18 months
SR 3.8.1.12	----------NOTE---------- Momentary transients below the minimum load specified do not invalidate this test. ------------------------------ Verify each DG operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 4353 for the 4075 kW DGs and ≥ 3100 kW for the 2850 kW DG; and b. For the remaining hours of the test loaded ≥ 4075 kW for the 4075 kW DGs and ≥ 2850 kW for the 2850 kW DG.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-----NOTES----- 1. This Surveillance shall be performed within 10 minutes of shutting down the DG after the DG has operated $\geq$ 2 hours loaded $\geq$ 4075 kW for the 4075 kW DGs and $\geq$ 2850 kW for the 2850 kW DG. Momentary transients below the minimum load specified do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. ----- Verify each DG starts and achieves, in $\leq$ 12 seconds, voltage $\geq$ 3952 V and frequency $\geq$ 60 Hz.	18 months
SR 3.8.1.14	-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by returning DG to ready-to-load operation.	18 months
SR 3.8.1.16	Verify interval between each sequenced load block is within ± 10% of design interval or 0.5 seconds, whichever is greater, for each emergency load sequencer.	18 months
SR 3.8.1.17	------------------------------NOTES------------------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --	
	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 12 seconds,	18 months (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.17 (continued)	2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage $\geq$ 3740 V and $\leq$ 4580 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	
SR 3.8.1.18	---------------------------NOTE--------------------------- Testing of the shared Emergency Diesel Generator (EDG) set (EDG 1-2A or EDG 1C) on either unit may be used to satisfy this surveillance requirement for these EDGs for both units. -- Verify each DG does not trip and voltage is maintained $\leq$ 4990 V and $\geq$ 3330 V during and following a load rejection of $\geq$ 1200 kW and $\leq$ 2400 kW.	5 years
SR 3.8.1.19	---------------------------NOTE--------------------------- All DG starts may be preceded by an engine prelube period. -- Verify when started simultaneously from standby condition, each DG achieves, in $\leq$ 12 seconds, voltage $\geq$ 3952 V and frequency $\geq$ 60 Hz.	10 years

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG set. DG set A consists of the 1-2A and 1C DGs. DG set B consists of the 1B DG (Unit 1) and the 2B DG (Unit 2).

Offsite power is supplied to the 230 kV and 500 kV switchyard(s) from the transmission network by six transmission lines. From the 230 kV switchyard, two electrically and physically separated circuits provide AC power, through startup auxiliary transformers, to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the FSAR, Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

In addition to providing a pre-determined sequence of loading the DGs, the train A and train B automatic load sequencers also function to actuate the required ESF loads on the offsite circuits. When offsite power is available, the automatic load sequencers function to simultaneously start the required ESF loads upon receipt of an SI actuation signal.

The onsite standby power source is provided from 4 DGs (1-2A, 1B, 2B, and 1C). The DGs are of two different sizes. The 1B, 2B, and

(continued)

BASES

BACKGROUND (continued)

1-2A DGs are rated at 4075 kW and the 1C DG is rated at 2850 kW. DG 1-2A and 1-C are assigned to the redundant load group train A. The train A load group is supplied from 4160V emergency Buses, F, H, and K. The 4160V H bus does not supply any design basis required loads by itself but is required to support the operation of DG 1C to supply the emergency Buses F and K which in turn supply design basis required loads. DGs 1B and 2B are assigned to the redundant load group train B. The train B load group is supplied from 4160V emergency Buses G, J, and L. The 4160V bus J does not supply any design basis required loads and is only required for the response to a station blackout which is not a design basis accident.

DGs 1B and 2B are dedicated to train B of Unit 1 and Unit 2, respectively, and each DG comprises a required DG set for its associated unit. DGs 1-2A and 1C are dedicated to train A but are shared between both units and together comprise a required DG set for both units. However, there are no design basis events in which DG 1-2A or 1C are required to supply power to the safety loads of both units simultaneously. In all events, DG 1-2A and 1C are assigned to only one of the two units depending on the event.

The 4.16 kV emergency busses required to supply equipment essential for safe shutdown of the plant at F, G, H, J, K, and L for each unit. These are supplied by two startup transformers on each unit connected to the offsite source during normal and emergency operating conditions. In the event one startup transformer on a unit fails, three of the emergency busses on that unit will be de-energized with their loss annunciated in the Main Control Room. The respective busses Diesel Generators will start and LOSP loads will be sequenced on to those busses. In the event Diesels fail, manual action will be required to re-energize the affected busses from the other startup transformer for that unit.

A DG starts automatically on a safety injection (SI) signal (i.e., low pressurizer pressure or high containment pressure signals) or on an ESF bus degraded voltage or undervoltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, a sequencer strips nonpermanent loads from the ESF

(continued)

BASES

BACKGROUND (continued)

bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 2850 kW for DG 1C and 4075 kW for DGs 1-2A, 1B, and 2B. DG 1C has a 2000 hour rating of 3100 kW and overload permissible up to 3250 kW for 300 hours per year. DGs 1-2A, 1B, and 2B have a 2000 hour rating of 4353 kW and overload permissible up to 4474 kW for 2 hours in any 24 hour period with a maximum of 300 hours cumulative per year. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during Accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits (i.e., consistent with the requirements of GDC 17) consisting of two physically independent transmission lines from the offsite transmission network to the switchyard and two independent circuits between the switchyard and the onsite Class 1E Electrical Power System along with separate and independent DG sets for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

In addition, one automatic load sequencer per train must be OPERABLE (B1F, B2F, B1G, and B2G).

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

Two physically independent circuits between the transmission network and the onsite system may consist of any combination that includes two of the six transmission lines normally supplying the 230 and 500 kV switchyards and both independent circuits from the 230 kV switchyard to the Class 1E buses via Startup Auxiliary Transformers 1A (2A) and 1B (2B). The two of six combination of transmission lines may be shared between Unit 1 and 2. If either of the transmission lines are 500 kV, one 500/230 kV Autotransformer connecting the 500 and 230 kV switchyards is available. If both of the transmission lines are 500 kV, both 500/230 kV Autotransformers

(continued)

BASES

LCO (continued)

connecting the 500 and 230 kV switchyards are available. Any combination of 500 and 230 kV circuit breakers required to complete the independent circuits is permissible.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within 12 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. For DG 1C this capability requires the support of the 4160 V H bus to enable DG 1C to supply the 4160 V buses F and K. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. All ESF buses, with two power sources available, have their supply breakers interlocked such that the buses can receive power from only one source at a time.

APPLICABILITY

The AC sources and sequencers are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

(continued)

BASES

APPLICABILITY (continued)

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. The redundant required features referred to in this Required Action include the motor driven auxiliary feedwater pump as well as the turbine driven auxiliary feedwater pump. One motor driven auxiliary feedwater pump does not provide 100% of the auxiliary feedwater flow assumed in the safety analyses. Therefore, in order to ensure the auxiliary feedwater safety function, the turbine driven auxiliary feedwater pump must be considered a redundant required feature addressed by this Required Action.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying it loads; and

b. A required feature on the other train is inoperable.

(continued)

BASES

ACTIONS

A.2 (continued)

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous

(continued)

BASES

ACTIONS

A.3 (continued)

occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 10 days. This could lead to a total of 13 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 10 days (for a total of 23 days) allowed prior to complete restoration of the LCO. The 13 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 13 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

The Condition B Required Actions are modified by a Note that is applicable when only one of the three individual DGs is inoperable. The Note specifies that the provisions of LCO 3.0.4 do not apply. The allowance provided by this note, to enter the MODE of applicability with a single inoperable DG, takes into account the capacity and capability of the remaining AC sources and the fact that operation is ultimately limited by the Condition B Completion Time for the inoperable DG set.

To ensure a highly reliable power source remains with an inoperable DG set, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

(continued)

BASES

ACTIONS (continued)

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG set is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. The redundant required features referred to in this Required Action include the motor driven auxiliary feedwater pump as well as the turbine driven auxiliary feedwater pump. One motor driven auxiliary feedwater pump does not provide 100% of the auxiliary feedwater flow assumed in the safety analyses. Therefore, in order to ensure the auxiliary feedwater safety function, the turbine driven auxiliary feedwater pump must be considered a redundant required feature addressed by this Required Action. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG set.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG set exists; and

b. A required feature on the other train (Train A or Train B) is inoperable.

If at any time during the existence of this Condition (one DG set inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG set inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG set, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

(continued)

BASES

ACTIONS

B.2 (continued)

In this Condition, the remaining OPERABLE DG set and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG set does not exist on the OPERABLE DG set, SR 3.8.1.6 does not have to be performed. If the cause of inoperability exists on other DG(s), the other DG set would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG set cannot be confirmed not to exist on the remaining DG set, performance of SR 3.8.1.6 suffices to provide assurance of continued OPERABILITY of that DG set.

In the event the inoperable DG set is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DG set is not affected by the same problem as the inoperable DG set.

B.4

Operation may continue in Condition B for a period that should not exceed 10 days.

(continued)

BASES

ACTIONS

B.4 (continued)

In Condition B, the remaining OPERABLE DG set and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 10 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 13 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 16 days) allowed prior to complete restoration of the LCO. The 13 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 10 day and 13 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. The redundant required features referred to in this Required Action include the motor driven auxiliary feedwater pump as well as the turbine driven auxiliary feedwater pump. One motor driven auxiliary feedwater pump does not provide 100% of the auxiliary feedwater flow assumed in the safety analyses. Therefore, in order to ensure the auxiliary feedwater safety function, the turbine driven auxiliary feedwater pump must be considered a redundant required feature addressed by this Required Action.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems — Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

Operation may continue in Condition D for a period that should not exceed 24 hours.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

E.1

With all or part of Train A DG set and Train B DG set inoperable, the capacity of the remaining standby AC sources is reduced depending on which combination of individual DGs is affected. Thus, with an assumed loss of offsite electrical power, standby AC sources may be insufficient to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

With all or part of each train of DG sets inoperable, operation may continue for a given unit for different periods of time depending on the combination of individual DGs that are inoperable. The length of time allowed increases with decreasing severity in the combinations of inoperable DGs. One set must be restored to operable status in 2 hours if DGs 1-2A, 1C, and 1B on Unit 1 or DGs 1-2A, 1C, and 2B on Unit 2 are inoperable. Operability of one set must be restored in 8 hours if DGs 1-2A and 1B on Unit 1 or DGs 1-2A and 2B on Unit 2 are inoperable. Operability of one set must be restored in 24 hours if DGs 1C and 1B on Unit 1 or DGs 1C and 2B on Unit 2 are inoperable.

(continued)

BASES

ACTIONS (continued)

F.1

Condition F provides the default Required Actions for the Conditions which address two inoperable offsite circuits or two inoperable DG sets. If the inoperable AC Sources cannot be restored to OPERABLE status within the applicable Completion Time, Required Action F.1 specifies that the unit be placed in MODE 3 within 6 hours. Once shut down, the unit is in a more stable condition and the time allowed to remain in MODE 3 is ultimately limited by the Required Actions and Completion Times applicable to a single inoperable AC Source based on the time that an AC Source initially became inoperable. In addition, the Required Actions applicable to one inoperable DG set or offsite circuit would remain applicable until both inoperable DG sets or offsite circuits are restored to OPERABLE status or the unit is placed in a MODE in which the LCO does not apply (MODE 5). The allowed Completion Times are reasonable to reach the required unit conditions from full power in an orderly manner and without challenging plant systems.

G.1

The sequencer(s) B1F, B2F, B1G, and B2G are an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. Therefore, loss of an ESF bus sequencer affects every major ESF system in the train. The 12 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.

H.1 and H.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. This condition exists when any combination of sources from the categories in LCO 3.8.1 totaling three or more are not OPERABLE. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.108 (Ref. 9), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3740 V is 90% of the nominal 4160 V output voltage. This value, which is specified in NEMA MG1 (Ref. 12), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 4580 V limits bus voltage to 110% of the nominal 4160 V. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1 (continued)

appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.6

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.6 testing, the DGs are started from standby conditions. Standby conditions for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer. During a modified start, a DG will not respond to a ESF or LOSP signal automatically. Therefore, the DG is considered inoperable with respect to response to ESF or LOSP signals during the brief duration of modified starts. If necessary, Operator action is required to place the speed control in automatic and reset the excitation system. This will immediately allow the DG to achieve normal voltage and frequency.

The DG shall be verified to accelerate to at least a synchronous speed of 900 rpm for the 2850 kW generator and 514 rpm for the 4075 kW generators.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.6 (continued)

SR 3.8.1.6 requires that, at a 184 day Frequency, the DG starts from standby conditions and achieves required voltage and frequency within 12 seconds. The permissive for closing the generator output breaker requires frequency to be greater than 57 Hz and voltage greater than 3952 V. The 12 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter 15 (Ref. 5).

The 12 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 12 second start requirement of SR 3.8.1.6 applies.

Since SR 3.8.1.6 requires a 12 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The normal 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.108 (Ref. 9). The 184 day Frequency for SR 3.8.1.6 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads in a range comparable to the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.108 (Ref. 9).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Note 3 indicates that this Surveillance should be conducted on only one DG per unit at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 3 is intended to be applied on a per unit basis and is not intended to preclude testing DGs on different units at the same time. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above a level which ensures sufficient time for manual transfer of fuel oil from the DG storage tank if the automatic transfer fails. The level is expressed as an equivalent volume in gallons, and ensures adequate fuel oil for a minimum of 3 hours of DG operation at the continuous rating.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for fuel transfer systems are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.5 (continued)

The design of fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate.

SR 3.8.1.6

See SR 3.8.1.2.

SR 3.8.1.7

Transfer of the unit power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

SR 3.8.1.8

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and while maintaining a specified margin to the overspeed trip. The single load for each DG is approximately 1000 kW. This Surveillance may be accomplished by:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by Regulatory Guide 1.9 (Ref. 3), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint.

The voltage tolerance specified in this SR is derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence interval. The voltage specified is consistent with the design range of the equipment powered by the DG. SR 3.8.1.8.b is the steady state voltage value to which the system must recover following load rejection. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

SR 3.8.1.9

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of 12 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.10

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (12 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.10.d and SR 3.8.1.10.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power. Emergency loads are started simultaneously by logic in the load sequencers sensing the availability of offsite power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for the first Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for the second Note (which only applies to SR 3.8.1.10.d and e) is that during operation with the reactor critical, performance of SR 3.8.1.10.d and e could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

SR 3.8.1.11

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal and/or an ESF actuation test signal, i.e., are bypassed during accident conditions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.1.12

This surveillance requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent to the 2000 hour load rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR. The steady-state generator voltage and frequency shall be maintained between 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by a Note. The Note states that momentary transients due to changing bus loads do not invalidate this test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 12 seconds. The 12 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least 2 hours at full load conditions prior to performance of this Surveillance is consistent with the manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.14

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.15

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open.

This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8).

SR 3.8.1.16

Under accident conditions, loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The 10% (or 0.5 seconds, whichever is greater) load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

SR 3.8.1.17

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.9, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.18

This Surveillance demonstrates the DG capability to reject a load of 1200-2400 kW without overspeed tripping or exceeding the predetermined voltage limits. The DG load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a 1200-2400 kW load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated. The DG output breaker(s) must remain closed such that the DG is connected to at least one ESF bus. All fuses and breakers on the energized ESF bus(es) must be verified not to trip.

This surveillance is modified by a note which states that testing of the shared Emergency Diesel Generator (EDG) set (EDG 1-2A or EDG 1C) on either unit may be used to satisfy this surveillance requirement

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

for these EDGs for both units. The surveillance requirement consists of sufficient testing to demonstrate that each DG, the DG output breaker, and bus fuses and breakers can successfully withstand a 1200-2400 kW load rejection on each unit. This does not require, however, that each shared DG be aligned to each unit and a load rejection be performed in a redundant fashion. This surveillance is intended to assure the correct performance of the DG voltage regulators and governors.

The 5 year Frequency is adequate and has been shown to be acceptable by operating experience.

SR 3.8.1.19

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9). This surveillance would also be applicable after any modifications which could affect DG interdependence.

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 1, 1971.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.

(continued)

BASES

REFERENCES (continued)

6. Regulatory Guide 1.93, Rev. 0, December 1974.

7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

8. 10 CFR 50, Appendix A, GDC 18.

9. Regulatory Guide 1.108, Rev. 1, August 1977.

10. ASME, Boiler and Pressure Vessel Code, Section XI.

11. IEEE Standard 308-1971.

12. NEMA MG1-1967.

90. 068A2.04 001/2/1/R-18/MEM 3.3/MOD/FA011005/S/GTO

A radioactive liquid release is in progress from the #2 Waste Monitor Tank (WMT) to the river in accordance with a liquid waste permit and SOP-50.1, Liquid Waste Processing System Liquid Waste Release from Waste Monitor Tank.

- Annunciator FH1, RMS HI RAD, has just alarmed.
- R-18, LIQ WASTE DISCH, is pegged high on the Radiation Monitoring system console and the High Alarm light is illuminated.
- The Radside SO reports that RCV-18 did not close.

Which ONE of the following describes the actions required in accordance with SOP-50.1 and the sample requirements necessary to release a WMT while RCV-18 is INOPERABLE per the ODCM?

A. Immediately have the Rad side SO secure #2 WMT pump, and inform the OSS. No liquid release is allowed until R-18 is returned to service.

B✓ Close the manual discharge valve to the environment, and inform the SSS. At least two independent samples must be analyzed for each Batch release prior to discharging the Waste Monitor Tank.

C. Manually close RCV-18, WMT Disch to Environment, and inform the Shift Radio chemist. A grab sample must be taken and analyzed after the release is initiated and once per hour while the release is in progress.

D. Fail air to RCV-18, WMT Disch to Environment, and notify Chemistry. At least one independent sample must be analyzed for each Batch release prior to discharging the Waste Monitor Tank and a grab sample taken hourly while the release is in progress.

ODCM page 2-1, 2-3, 2-4, 2-7, 2-9

A. Incorrect – a release is permitted as long as the ODCM is adhered to. Securing the pump will stop the release and the OSS is not the correct person per procedure to notify.

B. Correct –**Close the manual discharge valve to the environment, and inform the SSS. At least two independent samples must be analyzed for each Batch release prior to discharging the Waste Monitor Tank.**
per ODCM and Page 90 of 052106D
SOP-50.1 states that if the discharge is in progress and R-18 becomes inoperable the discharge is to be immediately stopped and the Shift Support Supervisor notified. The manual discharge valve will stop the release.

C. Incorrect - No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples. Also the SRC is the wrong person to notify.

D. Incorrect – No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples. Also if RCV 18 did not close on the auto close signal, it may not shut when air is failed. Also the wrong person is notified.

068A2.04 Liquid Radwaste
Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A2.04-Failure of automatic isolation

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Liquid and Solid Waste System (OPS52106A01).
·Technical Specification 5.5.1, Offsite Dose Calculation Manual (ODCM)
·Technical Specification 5.5.4, Radioactive Effluent Controls Program
·Technical Specification 5.6.2, Annual Radiological Environmental Operating Report
·Technical Specification 5.6.3, Radioactive Effluent Release Report
·TR 13.12.2, Liquid Holdup Tanks

modified from RMS-40305A12 #1 and LIQ SD WAST-40303A11 #7
Modified from the 2000 and 2001 NRC exams.

orig l

RMS-40305A12 001/HLT//C/A (LEVEL 2/3) SYS/068A2.04////

A radioactive liquid release is in progress from the #2 Waste Monitor Tank (WMT) to the river in accordance with a liquid waste permit and SOP-50.1, "LIQUID WASTE PROCESSING SYSTEM LIQUID WASTE RELEASE FROM WASTE MONITOR TANK."

- Annunciator FH2, "RMS CH FAILURE," has just alarmed.
- R-18, "LIQ WASTE DISCH," is pegged low on the Radiation Monitoring system console and the Low Alarm light is illuminated.

Which ONE of the following describes the actions required in accordance with SOP-50.1?

A✓ Immediately close RCV-18, WMT Disch to Environment, and inform the Shift Support Supervisor.

B. Check RCV-18, WMT Disch to Environment, closed automatically, notify Chemistry to implement sampling in accordance with the Offsite Dose Calculation Manual.

C. Immediately secure #2 WMT pump, inform the Operations Support Supervisor (OSS), then inform Chemistry to implement sampling in accordance with the Offsite Dose Calculation Manual.

D. Check RCV-18, WMT Disch to Environment, closed automatically, verify the last reading on R-18 was below the setpoint and inform the Shift Support Supervisor.

A - Correct, SOP-50.1 states that if the discharge is in progress and R-18 becomes inoperable the discharge is to be immediately stopped and the Shift Support Supervisor notified. Shutting RCV-18 immediately stops the discharge (R-18 automatic action for hi radiation).
B - Incorrect, This is the provision for discharging with R-18 unavailable. (NEED TO ENSURE THIS ANSWER IS CORRECT BY REVIEWING FNP-0-M-011, CHAPTER 2).
C - Incorrect, Securing the pump will stop the discharge although not immediately due to pump coast down and possible syphoning effects on the discharge line. Notifying Chemistry and Health Physics may be prudent but notifying the Shift Support Supervisor is required.
D - Incorrect, The discharge in progress must be securred.
Source: New
2001 nrc exam

original

QUESTIONS REPORT

for Plant Systems Questions 6-11-2004

LIQ SD WAST-40303A11 007/HLT/M (LEVEL 1) SYS/068K6.10////

Which of the following is correct concerning the sample requirements for an effluent release with R-18 inoperable?

A✓ At least two independent samples are analyzed for each Batch release prior to discharging the Waste Monitor Tank.

B. At least one independent sample is analyzed for each Batch release prior to discharging the Waste Monitor Tank <u>and</u> a grab sample taken hourly while the release is in progress.

C. No release can be made until R-18 is returned to service.

D. A grab sample taken and analyzed after the release is initiated <u>and</u> once per hour while the release is in progress.

ODCM page 2-1, 2-3, 2-4, 2-7, 2-9

*ABASIS	Correct – per ODCM and Page 90 of 052106D
*BBASIS	Incorrect – No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples
*CBASIS	Incorrect – release is permitted
*DBASIS	Incorrect - No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples

2000 NRC exam

2.6 The version of this procedure has been verified to be the current version and correct unit for the task. (OR 1-98-498)

3.0 Precautions and Limitations

3.1 Radiation monitor R-18 must be frequently observed during the release of radioactive liquid to assure that the count rate is not approaching R-18 set point as stated on the release permit.

3.2 IF R-18 becomes inoperable while discharging liquid waste to the river, THEN the discharge shall be stopped immediately and the Shift Support Supervisor notified.

3.3 Once a waste monitor tank has been placed on recirculation for sampling purposes prior to discharging to the environment, the tank shall remain in an isolated condition to prevent the introduction of any liquids which could alter the concentration of the tank's contained volume.

3.4 IF WMT pump maintenance has been performed since the last time step 2.4.4 and Table 1 of Appendix 1 or 2 were revised, THEN these tables are no longer valid and Appendix 3 must be performed.

4.0 Instructions

4.1 Waste Monitor Tank 1 Release to the Environment

4.1.1 Perform Appendix 1.

4.2 Waste Monitor Tank 2 Release to the Environment

4.2.1 Perform Appendix 2.

4.3 Radiation Monitor R-18 Check

4.3.1 Verify waste monitor tank 1 and 2 discharge valves 1-LWP-V-7446 (Q1G21V111) and 1-LWP-V-7448 (Q1G21V114) are closed and locked before performing this test.

4.3.2 IF not performed per Appendix 1 or 2, THEN verify meter responds in the top scale direction on channel R-18 by inserting check source. IF check source is NOT sufficient, THEN have Health Physics source check with a portable source.

4.3.3 IF required, THEN reset R-18 trip. IF R-18 high background prevents resetting trip, THEN increase the R-18 pot setting to a setting above background AND reset the trip.

2.14 IF **R-14** alarms, THEN perform the following:

2.14.1 IF high effluent activity is possible, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.14.2 Refer to FNP-1-SOP-51, WASTE GAS SYSTEM for potential problems with the waste gas system.

2.15A IF **R-15** alarms AND remains above the alarm setpoint (not a momentary spike), THEN perform the following:

2.15A.1 IF high effluent activity is possible, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMT 0008751, 0008755}.**

2.15A.2 Notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate.

2.15A.3 Notify the Operations Shift Superintendent

2.15B IF **R-15B** OR **R-15C**, alarms AND remains above the alarm setpoint (not a momentary spike), THEN notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate.

2.16 IF **R-16** alarms, THEN refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

2.17 IF **R-17A** OR **R-17B** alarms, THEN monitor CCW pump operation while the CCW surge tank vents are closed.

NOTE:	**IF CCW surge tank vents are closed for reasons other than an actual high radiation alarm, THEN with Shift Supervisor concurrence, the CCW surge tank vents should be cycled once every shift (eight hours) and documented in AutoLog.**

2.18 IF **R-18** alarms with high liquid effluent activity possible, THEN verify any liquid waste release is secured and refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

2.19 IF **R-19** alarms AND remains above the alarm setpoint (not a momentary spike), THEN notify the Counting Room to immediately sample the SG's per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate. Refer to FNP-1-SOP-45.0, RADIATION MONITORING SYSTEM for guidance in sampling steam generators with R-19 in alarm.

CHAPTER 2

LIQUID EFFLUENTS

2.1 LIMITS OF OPERATION

The following Liquid Effluent Controls implement requirements established by Technical Specifications Section 6.0 {5.0}. Terms printed in all capital letters are defined in Chapter 10.

2.1.1 Liquid Effluent Monitoring Instrumentation Control

In accordance with Technical Specification 6.8.3.e(i) {5.5.4.a}, the radioactive liquid effluent monitoring instrumentation channels shown in Table 2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits specified in Section 2.1.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Section 2.3.

2.1.1.1 Applicability

This limit applies at all times.

2.1.1.2 Actions

With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above control, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperable, or change the setpoint to a conservative value.

With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 2-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report {Radioactive Effluent Release Report} pursuant to Section 7.2 why this inoperablility was not corrected in a timely manner.

This control does not affect shutdown requirements or MODE changes.

2.1.1.3 Surveillance Requirements

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK {source check}, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST {CHANNEL OPERATIONAL TEST (COT)} operations at the frequencies shown in Table 2-2.

2.1.1.4 Basis

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in Section 2.3 to ensure that the alarm/trip will occur prior to exceeding the limits of Section 2.1.2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 2-1. Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	OPERABILITY Requirements[a]	
	Minimum Channels Operable	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE-18)	1	28
b. Steam Generator Blowdown Effluent Line (RE-23B)	1	29
2. Flowrate Measurement Devices		
a. Liquid Radwaste Effluent Line		
1) Waste Monitor Tank No. 1	1	30
2) Waste Monitor Tank No. 2	1	30
b. Discharge Canal Dilution Line (Service Water)	1	30
c. Steam Generator Blowdown Effluent Line	1	30

a. All requirements in this table apply to each unit.

Table 2-1 (contd). Notation for Table 2-1 – ACTION Statements

ACTION 28 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Section 2.1.2.3, and

b. At least two technically qualified members of the Facility Staff independently verify the discharge line valving and

(1) Verify the manual portion of the computer input for the release rate calculations performed on the computer, or

(2) Verify the entire release rate calculations if such calculations are performed manually.

Otherwise, suspend release of radioactive effluents via this Pathway.

ACTION 29 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a MINIMUM DETECTABLE CONCENTRATION no greater than 1 × 10^{-7} μCi/mL:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 μCi/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 μCi/gram DOSE EQUIVALENT I-131.

ACTION 30 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

Table 2-2. Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	Surveillance Requirements[d]			
	CHANNEL CHECK	{Source Check} SOURCE CHECK	CHANNEL CALIBRATION	{CHANNEL OPERATIONAL TEST} CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitors Providing Automatic Termination of Release				
a. Liquid Radwaste Effluent Line (RE-18)	D	P	R[b]	Q[a]
b. Steam Generator Blowdown Effluent Line (RE-23B)	D	M	R[b]	Q[a]
2. Flowrate Measurement Devices				
a. Liquid Radwaste Effluent Line 1) Waste Monitor Tank No. 1	D[c]	NA	R	NA
2) Waste Monitor Tank No. 2	D[c]	NA	R	NA
b. Discharge Canal Dilution Line (Service Water)	D[c]	NA	R	Q
c. Steam Generator Blowdown Effluent Line	D[c]	NA	R	NA

Table 2-2 (contd). Notation for Table 2-2

a. In addition to the basic functions of a CHANNEL FUNCTIONAL TEST {CHANNEL OPERATIONAL TEST} (Section 10.2):

(1) The CHANNEL FUNCTIONAL TEST {CHANNEL OPERATIONAL TEST} shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

(a) Instrument indicates measured levels above the alarm/trip setpoint;

(b) Loss of control power; or

(c) Instrument controls loss of instrument power.

(2) The CHANNEL FUNCTIONAL TEST {CHANNEL OPERATIONAL TEST} shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

(a) Instrument indicates a downscale failure; or

(b) Instrument controls not set in operate mode.

b. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology or using standards that have been obtained from suppliers that participate in measurements assurance activities with NIST. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

c. CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

d. All requirements in this table apply to each unit.

2.1.2 Liquid Effluent Concentration Control

In accordance with Technical Specifications 6.8.3.e(ii) {5.5.4.b} and 6.8.3.e(iii) {5.5.4.c}, the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 10-1) shall be limited at all times to ten times the concentrations specified in 10 CFR 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 1×10^{-4} μCi/mL total activity.

2.1.2.1 Applicability

This limit applies at all times.

2.1.2.2 Actions

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the limits stated in Section 2.1.2, immediately restore the concentration to within the stated limits.

This control does not affect shutdown requirements or MODE changes.

2.1.2.3 Surveillance Requirements

The radioactivity content of each batch of radioactive liquid waste shall be determined by sampling and analysis in accordance with Table 2-3. The results of radioactive analyses shall be used with the calculational methods in Section 2.3 to assure that the concentration at the point of release is maintained within the limits of Section 2.1.2.

2.1.2.4 Basis

This control is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than ten times the concentration levels specified in 10 CFR 20, Appendix B, Table 2, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.1301 to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection

(ICRP) Publication 2 (1959). The resulting concentration of 2×10^{-4} was then multiplied by the ratio of the effluent concentration limit for Xe-135, stated in Appendix B, Table 2, Column 1 of 10 CFR 20 (paragraphs 20.1001 to 20.2401), to the MPC for Xe-135, stated in Appendix B, Table II, Column 1 of 10 CFR 20 (paragraphs 20.1 to 20.601), to obtain the limiting concentration of 1×10^{-4} μCi/mL.

Table 2-3. Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling and Analysis Requirements[a,b]			
	Sampling FREQUENCY	Minimum Analysis FREQUENCY	Type of Activity Analysis	MINIMUM DETECTABLE CONCENTRATION (MDC) (μCi/mL)
A. Waste Tanks Producing BATCH RELEASES				
All	P Each BATCH	P Each BATCH	PRINCIPAL GAMMA EMITTERS I-131	5 E-7 1 E-6
	P One BATCH/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1 E-5
	P Each BATCH	M COMPOSITE	H-3 Gross Alpha	1 E-5 1 E-7
	P Each BATCH	Q COMPOSITE	Sr-89, Sr-90 Fe-55	5 E-8 1 E-6
B. CONTINUOUS RELEASES[c]				
Steam Generator Blowdown	D Grab Sample	W COMPOSITE	PRINCIPAL GAMMA EMITTERS I-131	5 E-7 1 E-6
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1 E-5
	D Grab Sample	M COMPOSITE	H-3 Gross Alpha	1 E-5 1 E-7
	D Grab Sample	Q COMPOSITE	Sr-89, Sr-90 Fe-55	5 E-8 1 E-6
Turbine Building Sump	P[d] Grab Sample	W COMPOSITE	PRINCIPAL GAMMA EMITTERS H-3	5 E-7 1 E-5

Table 2-3 (contd). Notation for Table 2-3

a. All requirements in this table apply to each unit. Deviation from the MDC requirements of this table shall be reported in accordance with Section 7.2.

b. Terms printed in all capital letters are defined in Chapter 10.

c. Sampling will be performed only if the effluent will be discharged to the environment.

d. Samples will be taken prior to or during each discharge.

a) R14: (Plant Vent Gas) closes Waste Gas Release Valve GWD-HV-014.

b) R16: (Boron Recycle System) diverts CVC-RCV-016 Recycle Evaporator discharge from Reactor Makeup Water System to the Recycle Evaporator Demineralizer.

c) R17A or B: (Component Cooling Water) closes RCV3028 CCW SRG TANK VENT.

d) R18: (Liquid Waste Processing) closes Liquid Waste Release Valve LWP-RCV-018.

e) R19: (Steam Generator Blowdown) isolates Steam Generator Blowdown Sample Lines.

f) R23A:(Steam Generator Blowdown Processing) closes BD-FCV-1152 S/G Blowdown Heat Exchanger Discharge Valve.

g) R23B:(Steam Generator Blowdown Processing) closes BD-RCV-023B Dilution Discharge Valve.

2. ARDA will automatically start for the conditions listed in Note.

IMMEDIATE ACTIONS

1. Check indications on radiation monitoring system console and determine which radiation monitor channel indicates high activity.
2. Insure that any automatic actions, associated with the alarmed channel, have occurred.

Consult a current copy of ARP-1.6 for subsequent actions.

RAD MONITOR CONTINGENCIES

Monitor	Contingency
R-5 (OOS)	Perform area survey 1/24 hours
R-11 (OOS)	Obtain gaseous grab from RE- 67 1/24 hours
R-12 (OOS)	Verify (CACCLMS) operable 1/24 hours

(CACCLMS) OOS		Verify R-12 operable

R-12 & (CACCLMS) OOS	Obtain samples from RE -- 67 1/24 hours

R-14 & R-22 (OOS)	Obtain a plant vent stack gas grab 1/8 hours from R-29A,RE-68 or R-29B

R-15A (OOS) Obtain a gas sample from RE-28 1/8 hours

R-15B & R-15-C (OOS)	Verify R-15A is operable 1/24 hours

R-18 (OOS) Prior to discharging a WMT, verify WMT releases using 2 independent samples and analysis per CCP-212

R-23B (OOS) Obtain a liquid sample of SGBD IAW CCP-643 or CCP-652 1/24 hours. If DEI is < 0.01 μCi/gm or 1/8 if STEAM GENERATOR DEI >0.01 μCi/gm

R-28 (OOS) No action required

R-29B If R-29B particulate and iodine samples are inoperable, verify R-29A or R-68 is operable and collect samples continuously. If R-29B noble gas monitor is inoperable, verify R-14 or R-22 in service 1/24 hours

R-29A Verify R-29B or R-68 is operable

R-60A (OOS) Ensure SJAE is in service and R-15A is operable

R-60B (OOS)

R-60C (OOS)

R-60D (OOS)

TABLE 1
RADIATION MONITORING SYSTEM

CHANNEL	LOCATION	MONITOR	DETECTOR	FUNCTION
R-1A	Control Room (Unit I Panel)	Area	G-M (W)	
R-1B	Technical Support Center (Unit II Panel) R-1B	Area	G-M (W)	
R-2	Containment (155' elev)	Area	G-M (W)	
R-3	Radiochemistry Lab (AB 139')	Area	G-M (W)	
R-4	#3 Charging Pump (AB 100')	Area	G-M (W)	
R-5*	Spent Fuel Pool Room (AB 155')	Area	G-M (W)	
R-6	Sampling Room (AB 139')	Area	G-M (W)	
R-7	In-core NIS Area (CTMT 129', near Seal Table)	Area	G-M (W)	
R-8	Drumming Station (AB 155')	Area	G-M (W)	
R-9	SG Sample Panel (Unit II Panel) (AB 139')	Area	G-M (W)	
R-10	Penetration Room Filtration Discharge (AB 155')	APD	Scint. (Victoreen)	
R-11*	Containment Atmosphere (AB 121')	APD	Scint. (Victoreen)	
R-12*	Containment Atmosphere (AB 121')	Gas	G-M (W)	
R-13	Waste Gas Compressor Suction (AB 100' WGC Valve Room)	Gas	G-M (W)	
R-14 ODCM	Plant Vent Stack (AB Roof)	Gas	G-M (W)	Closes HCV-14
R-15A ODCM	Condenser Air Ejector Discharge Header (TB 155')	Gas	G-M	
R-15B*	Condenser Air Ejector (Intermediate Range) (TB 189')	Gas	G-M (Eberline)	
R-15C*	Condenser Air Ejector (High Range) (TB 189')	Gas	Ion Chamber (Eberline)	

*Technical Specification related

TABLE 1, CONT.
RADIATION MONITORING SYSTEM

CHANNEL	LOCATION	MONITOR	DETECTOR	FUNCTION
R-16	Recycle Evaporator Distillate Line (AB 139')	Liquid	Scint. (W)	Diverts RCV-16 from RMW system to the Recycle Evaporator Demineralizer
R-17A	Component Cooling Water (CCW Hx Room)	Liquid	Scint. (W)	Closes CCW surge tank vent (RCV-3028)
R-17B	Component Cooling Water (CCW Hx Room)	Liquid	Scint.	Closes CCW surge tank vent (RCV-3028)
R-18 ODCM	Waste Monitor Tank Pump Discharge (AB 121' at the Batching Funnel)	Liquid	Scint. (W)	Closes RCV-18
R-19	Steam Generator Blowdown/Sample (AB 139')	Liquid	Scint. (W)	Isolates sample lines 3328, 3329, 3330
R-20A	Service Water from Containment Coolers A and B (AB 121' BTRS Chiller Room)	Liquid	Scint. (W)	
R-20B	Service Water from Containment Coolers C and D (AB 121')	Liquid	Scint. (W)	
R-21	Plant Vent Stack (AB 155')	APD	Scint. (Victoreen)	
R-22 ODCM	Plant Vent Stack (AB 155')	Gas	G-M (W)	
R-23A	SG Blowdown Surge Tank Inlet (AB 130')	Liquid	Scint. (W)	Closes FCV-1152
R-23B ODCM	SG Blowdown Surge Tank Discharge (AB 130')	Liquid	Scint. (W)	Closes RCV-23B
R-24A*	Containment Purge (AB 155')	Gas	Scint.	Closes containment purge supply & exhaust dampers 2866C & 2867C and 3198A & D

*Technical Specification related

TABLE I, CONT.
RADIATION MONITORING SYSTEM

CHANNEL	LOCATION	MONITOR	DETECTOR	FUNCTION
R-24B*	Containment Purge (AB 155')	Gas	Scint. (Victoreen)	Closes valves: 2866D &2867D, 3196, 3197, 3198B & C
R-25A/R-25B*	Spent Fuel Pool Ventilation (AB 184')	Gas	Scint. (Victoreen)	Trip fuel bldg supply and exhaust fans; closes SFP HVAC supply and exhaust dampers; starts associated trains of penetration room filtration.
R-26A	Recycle Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)	
R-26B	Waste Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)	
R-27A*	Containment (High Range)	Area	Ion Chamber (Victoreen)	
R-27B*	Containment (High Range)	Area	Ion Chamber (Victoreen)	Alert alarm starts recorder @ 10R/hr
R-28	Condenser Air Ejector Off-Line Sample (TB 155')	None	None	Provide off line sample capability
R-29A	Plant Vent Stack	None	None	Provide off line sample capability
R-29B* ODCM	Plant Vent Stack	SPING-4	Scint./G-M (Eberline)	
R-30A	Radwaste Area Vents El 100'	APD	Scint. (Victoreen)	
R-30B	Radwaste Area Vents El 100'	Gas	Scint. (Victoreen)	
R-31	Radwaste Area Vents El 121'	APD	Scint. (Victoreen)	
R-32	Radwaste Area Vents El 139'	APD	Scint. (Victoreen)	

*Technical Specification related

TABLE 1, CONT.
RADIATION MONITORING SYSTEM

CHANNEL	LOCATION	MONITOR	DETECTOR	FUNCTION
R-33	Radwaste Area Vents El 155'	APD	Scint. (Victoreen)	
R-34	Access Control Area (Unit 1 only)	APD	Scint. (Victoreen)	
R-35A*/ R-35B*	Control Room Vents (Control Room Ventilation Room)	Gas	Scint. (Victoreen)	Isolates control room supply and return from computer room AHU, closes utility exhaust dampers, and shifts TSC ventilation to recirc (filtration mode)
R-60A, B, & C*	Main Steam Relief Valve Exhaust Plume (Roof above MSVR)	Gas	Ion chamber (Victoreen)	
R-60D*	Turbine-Driven Auxiliary Feedwater Pump Exhaust (Roof above MSVR)	Gas	Ion chamber (Victoreen)	
R-66A through Bldg F	Low Level Rad Waste Building	Area	Ion chamber (Eberline)	Alert (10 mr/hr) - LLRW vent secures High (100 mr/hr) - Actuates magenta flashing lights throughout interior and on roof of LLRW Bldg
R-67	Containment Atmosphere	None	None	
R-68	Vent Stack Sampling	None	None	
R-70A, B, C	Main Steam Valve Room On Each Steam Line Nitrogen 16 (N-16) Monitor	Process	NaI Scint (Merlin-Gerin)	

*Technical Specification related

91. 073G2.2.25 001/2/1/R-11/12/C/A 3.7/NEW/FA011005/S/GTO

Which one of the following meets the MINIMUM reactor coolant leakage detection system(s) that must be in operation and OPERABLE in Mode 4 without entering an LCO and the reason?

A. The containment air cooler condensate level monitoring system (CACCLMS); since the likelihood of leakage and crack propagation in Mode 4 is much smaller than in Modes 1, 2 and 3.

B. R-12 **AND** the containment air cooler condensate level monitoring system; to be able to detect RCS pressure boundary leakage to minimize the potential for a gross failure to occur.

C. R-11 **AND** R-12; to be able to detect RCS pressure boundary leakage to minimize the potential for a gross failure to occur.

D. R-11; since the likelihood of leakage and crack propagation in Mode 4 is much smaller than in Modes 1, 2 and 3.

A. Incorrect - The CACCLMS in mode 4 by itself would put you in an LCO condition A The reason is NOT correct for this mode but is correct for mode 5/6.

B. Incorrect - R-11 has to be in operation or Condition A is entered. The reason is correct for R-11

C. Correct - **R-11 AND R-12; to be able to detect RCS pressure boundary leakage to minimize the potential for a gross failure to occur.**

IAW TS 3.4.15; R-11 and 12 or R-11 and CACCLMS is all that is required to be operable in Modes 1-4. The reason given is from Bases.

D. Incorrect - R-11 alone would cause condition B to be entered.
The reason is NOT correct for this mode but is correct for mode 5/6

073G2.2.25 Process Radiation Monitoring
Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.(CFR: 43.2)

(2) Facility operating limitations in the technical specifications and their bases.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Radiation Monitoring System (OPS52106D01).
• Technical Specification 3.4.15, RCS Leakage Detection Instrumentation

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment atmosphere particulate radioactivity monitor; and

b. One containment air cooler condensate level monitor or one containment atmosphere gaseous radioactivity monitor.

R-11 and R-12
OR
R-11 and CACCLMS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment atmosphere particulate radioactivity monitor inoperable. R-11	A.1.1	Analyze grab samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>		
	A.1.2	Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>		
	A.2	Restore the containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required containment atmosphere gaseous radioactivity monitor inoperable. R-12 AND Required containment air cooler condensate level monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere. OR B.1.2 Perform SR 3.4.13.1. AND B.2 Restore at least one required monitor to OPERABLE status.	Once per 24 hours Once per 24 hours 30 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. All required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, or flow rate, or in the operating frequency of a pump. The containment air cooler condensate level monitor is instrumented to alarm for abnormal increases in the level (flow rates). The sensitivity is acceptable for detecting increases in unidentified LEAKAGE. The condensate flow rate is measured by monitoring the water level in a vertical standpipe. As flow rate increases, the water level in the standpipe rises.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} µCi/cc radioactivity for particulate monitoring and of 10^{-6} µCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity

(continued)

BASES

BACKGROUND (continued)

levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow from the containment condensate air coolers. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment atmosphere particulate radioactivity monitor (R-11) in combination with a gaseous radioactivity monitor (R-12) or a containment air cooler condensate level monitor provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment particulate radioactivity monitor, the containment gaseous radioactivity monitor, and the containment air cooler condensate level monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

A.1.1, A.1.2, and A.2

R-11

With the required containment atmosphere particulate radioactivity monitor inoperable, no other form of sampling can provide the

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

equivalent information; however, the containment atmosphere gaseous radioactivity monitor or the containment air cooler condensate level monitor will provide indications of changes in leakage. Together with the atmosphere gaseous monitor or the condensate level monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours or grab samples of the containment atmosphere must be taken and analyzed once per 24 hours to provide information that is adequate to detect leakage.

Restoration of the required Particulate radioactivity monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance or containment grab sample analyses required by Required Action A.1.1 or A.1.2.

B.1.1, B.1.2, and B.2

With both the required gaseous containment atmosphere radioactivity monitoring instrumentation channel and the required containment air cooler condensate level monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the required containment monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

or SR 3.4.13.1 must be performed once per 24 hours and at least one required monitor must be restored to operable status in 30 days.

If the above required action and associated completion times cannot be met, the plant must be placed in mode 3 in 6 hours and in mode 5 in 36 hours.

If all required monitors are inoperable, immediately enter LCO 3.0.3.

The basis for the RCS Leakage Detection Instrumentation states that the systems must have the capability to detect significant reactor coolant pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. An early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage.

The containment air cooler condensate level monitor is instrumented to alarm for abnormal increases in the level (flow rates). The condensate flow rate is measured by monitoring the water level in a vertical standpipe. As flow rate increases, the water level in the standpipe rises.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage.

An increase in the humidity of the containment atmosphere indicates release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS leakage.

Surveillance requirements state to perform a channel check of the required containment atmosphere radioactivity monitor every 12 hours.

A COT of the required containment atmosphere radioactivity monitor must be performed every 92 days.

A channel calibration of the required containment atmosphere radioactivity monitor and a channel calibration of the required containment air cooler condensate level monitor must be performed every 18 months.

TECHNICAL REQUIREMENTS MANUAL

076G2.1.20 001/2/1/SW/C/A 4.2/NEW/FA011005/S/CVR/GTO

Unit 1 is at 38% power. LJ3, H2 TEMP HI, has come into alarm. H2 temp on the MCB reads 50°C and stable. Investigation reveals the following:

- SW TO TURB BLDG A(B) TRN Q1P16V514, V515, V516, & V517 are CLOSED and will not open.

Which one of the following is the correct action directed by AOP-7.0, Loss of Turbine Building Service Water?

A. Reduce Main Generator load until the alarm clears.

B. Maintain load as long as the Main Generator H2 temp is stable.

C. Trip the Reactor and enter EEP-0, Reactor Trip or Safety Injection.

D. Trip the Turbine and enter AOP-3.0, Turbine Trip Below P-9 Setpoint.

A. Incorrect. ARP-LJ3 directs reducing load until the alarm clears in the case of high temperatures if the temperatures are not due to Loss of SW.

B. Incorrect. Operating with this alarm in without reducing load to clear the alarm is not allowed by procedure.

C. Incorrect. This would be true since the alarm was caused by a loss of SW to the Turbine Building AND guidance was obtained from ARP-AF5, OR AOP-7.0 but power is >35%.

D. Correct. **Trip the Reactor and enter EEP-0, Reactor Trip or Safety Injection.** This is be true since the alarm was caused by a loss of SW to the Turbine Building and power level are >35%. It would be directed by ARP-LJ3 & AOP-7.0.

076G2.1.20 Service Water. Ability to execute procedure steps.
(CFR: 41.10 / 43.5 / 45.12)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. Evaluate abnormal plant or equipment conditions associated with the Service Water System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102F02).

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-7.0

LOSS OF TURBINE BUILDING SERVICE WATER

SAFETY RELATED

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	
Reference Use	ALL
Information Use	

Approved:

Todd Youngblood
Operations Manager

Date Issued: 11/25/2003

TABEL OF CONTENTS

A. Purpose

This procedure provides actions for response to a loss of service water cooling to the turbine building.

This procedure is applicable at all times.

B. Symptoms or Entry Conditions

I. This procedure is entered when a loss of turbine building service water is indicated by any of the following:

a. Rising temperatures on turbine building components supplied by service water

b. Actuation of SW TO TURB BLDG FLOW A OR B TRN HI annunciator AF5 (approximately 15,000 gpm)

Step	Action/Expected Response	Response NOT Obtained

*****

CAUTION: IF required to adequately cool running diesel generators, THEN any action previously taken to isolate service water to the turbine building to ensure an adequate cooling supply, should remain in effect during this procedure.

*****

NOTE:
- Step 1 is an IMMEDIATE ACTION step
- Steps 3, 4 and 5 should be performed in conjunction with FNP-1-ESP-0.1, REACTOR TRIP RESPONSE if sufficient personnel are available.
- SW TO TURB BLDG ISO A(B) TRN valves will automatically close if SW flow in either train is greater than 17,600 gpm.

Step	Action/Expected Response	Response NOT Obtained
1	Check at least one SW train aligned to turbine building. • Check A train SW - ALIGNED TO TURBINE BUILDING. SW TO TURB BLDG ISO A TRN [] Q1P16V515 open [] Q1P16V516 open OR • Check B train SW - ALIGNED TO TURBINE BUILDING. SW TO TURB BLDG ISO B TRN [] Q1P16V517 [] Q1P16V514	1 Perform the following. 1.1 Restore at least one SW train to turbine building. 1.1.1 Align A train SW to turbine building. SW TO TURB BLDG ISO A TRN [] Q1P16V515 open [] Q1P16V516 open OR 1.1.2 Align B train SW to turbine building. SW TO TURB BLDG ISO B TRN [] Q1P16V517 open [] Q1P16V514 open

Step 1 continued on next page.

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Step	Action/Expected Response	Response NOT Obtained
		1.2 IF Rx power is less than 35% AND SW flow cannot be immediately restored, THEN trip the turbine and refer to FNP-1-AOP-3.0, TURBINE TRIP BELOW THE P9 SETPOINT. 1.3 IF Rx power is greater than 35% AND SW flow cannot be immediately restored, THEN trip the reactor and refer to FNP-1-EEP-0.0, REACTOR TRIP OR SAFETY INJECTION.

NOTE: Step 2 is a continuing action.

Step	Action/Expected Response	Response NOT Obtained
___ 2	IF main generator on line, THEN check generator hydrogen temperature less than 46°C by the following: [] TI-4067	2 Perform the following: 2.1 IF Rx power is less than 35% THEN trip the turbine and refer to FNP-1-AOP-3.0, TURBINE TRIP BELOW THE P9 SETPOINT. 2.2 IF Rx power is greater than 35%, THEN trip the reactor and refer to FNP-1-EEP-0.0, REACTOR TRIP OR SAFETY INJECTION.
___ 3	Check SW HDR Pressure - GREATER THAN 110 psig. Train A [] PI-3001A Train B [] PI-3001B	3 CLOSE SW DIL BYP ISO. Train A [] Q1P16V558 Train B [] Q1P16V557

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
___ 4	Restore both trains of SW to turbine building.	
	4.1 Dispatch personnel to correct cause for loss of SW.	
	4.2 WHEN cause for loss of SW corrected, THEN verify both SW trains aligned to turbine building. SW TO TURB BLDG ISO A(B) TRN [] Q1P16V515 open [] Q1P16V516 open [] Q1P16V517 open [] Q1P16V514 open	
5	IF at least one SW train aligned to turbine building, THEN monitor turbine building component temperatures and go to procedure and step in effect.	5 Perform the following. 5.1 Stop all CNDS PUMPs. 5.2 Stop all HDPs. 5.3 Secure turbine building chillers using FNP-1-SOP-57.0, TURBINE BUILDING HVAC SYSTEM. 5.4 Secure steam dumps. 5.4.1 Manually control atmospheric relief valves to reduce steam dump demand to 0%. 1A(1B,1C) MS ATMOS REL VLV [] PC 3371A adjusted [] PC 3371B adjusted [] PC 3371C adjusted 5.4.2 WHEN steam dump demand is 0%, THEN place STM DUMP INTERLOCK A TRN and STM DUMP INTERLOCK B TRN switches to OFF RESET.

Step 5 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained
		5.5 Break condenser vacuum. COND VAC BKR MAN ISO (155 ft. TURB BLDG) [] N1N51V518A open [] N1N51V518B open COND VAC BKR VLVS [] N1N51V519A/519B open
		5.6 Secure gland sealing steam using FNP-1-SOP-28.4, GLAND SEALING STEAM SYSTEM.
		5.7 Begin cooldown to hot shutdown using FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY and FNP-1-UOP-2.2, SHUTDOWN OF UNIT FROM HOT STANDBY TO COLD SHUTDOWN.
		5.8 Return to step 3.

-END-

LOCATION LJ3

SETPOINT: 48°C

ORIGIN: 30X5 Relay actuated by Temperature Switch N1N43TE519-N

J3	
	H2 TEMP HI

PROBABLE CAUSE

1. Turbine overload.
2. Low Hydrogen purity.
3. High water temperature in Hydrogen Coolers.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. NOTIFY APPROPRIATE PLANT PERSONNEL.
2. VERIFY VALIDITY OF ALARM FROM MCB INDICATION.
3. **IF THIS ALARM WAS CAUSED BY A COMPLETE LOSS OF SERVICE WATER AND SERVICE WATER FLOW IS NOT IMMEDIATELY RESTORED, THEN PERFORM THE FOLLOWING:**

   3.1 IF Reactor Power is Greater than or equal to (≥) 35%, THEN trip the Reactor AND refer to FNP-1-EEP-0 **REACTOR TRIP OR SAFETY INJECTION.**

   3.2 IF Reactor Power is Less than (<) 35%, THEN trip the Turbine AND refer to FNP-1-AOP-3.0 **TURBINE TRIP BELOW P-9 SETPOINT.**

SUPPLEMENTARY ACTION

1. Determine and correct the cause of the High Hydrogen Temperature.
2. IF alarm valid, THEN reduce load until alarm clears.

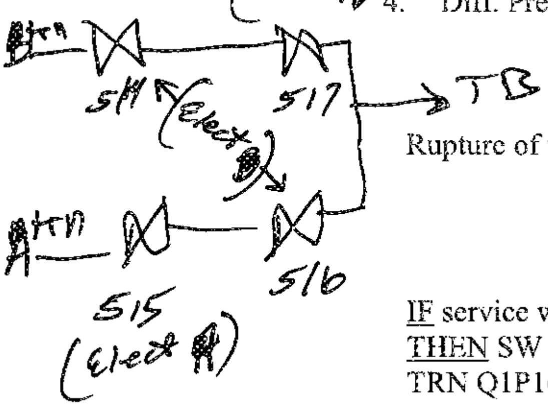
References: A-177100, Sh. 538; D-173150; C-173152; D-172784; D-173097; D-172700; U-162215; PCN 84-1-2657

LOCATION AF5

SETPOINT: 9.0 ± 1.5 PSID (~ 15,000 GPM)

F5
SW TO TURB BLDG A OR B TRN FLOW HI

ORIGIN:
1. Diff. Pressure Switch (Q1P16PDS565-A) ✓
2. Diff. Pressure Switch (Q1P16PDS566-B)
3. (Elect A) Diff. Pressure Switch (Q1P16PDS568-A) ✓
4. Diff. Pressure Switch (Q1P16PDS569-B)



PROMPT OPERATOR ACTION REQUIRED

PROBABLE CAUSE

Rupture of the Service Water piping in the Turbine Building.

AUTOMATIC ACTION

IF service water flow in the A Train exceeds 17,600 GPM (11 ± 1.5 PSID) THEN SW TO TURB BLDG A TRN Q1P16V516 and SW TO TURB BLDG B TRN Q1P16V514 will close.

IF service water flow in the B Train exceeds 17,600 GPM (11 ± 1.5 PSID) THEN SW TO TURB BLDG A TRN Q1P16V515 and SW TO TURB BLDG B TRN Q1P16V517 will close. ✓ ✓

NOTE: Consider closing Train A dilution bypass isolation valve Q1P16V558 and/or Train B dilution bypass valve Q1P16V557 (Diesel Bldg.) if SW HDR PRESS is less than 110 psig and SW Dilution Flow normal. (Ref. OR 2-99-336)

IMMEDIATE ACTION

1. IF SERVICE WATER FLOW HAS BEEN LOST TO THE TURBINE BUILDING, THEN ATTEMPT TO RESTORE SERVICE WATER FLOW.
2. IF SERVICE WATER FLOW CAN NOT BE IMMEDIATELY RESTORED, THEN PERFORM THE ACTIONS REQUIRED BY FNP-1-AOP-7.0, LOSS OF TURBINE BUILDING SERVICE WATER.

SUPPLEMENTARY ACTION

1. IF annunciator LJ3, H2 TEMP HI, alarms, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
2. Refer to FNP-2-AOP-7.0, LOSS OF TURBINE BUILDING SERVICE WATER.
3. Notify appropriate personnel to locate and correct the cause of the alarm.

References: A-177100, Sh. 80; D-172674, Sh. 1 & 2; D-170119, Sh. 2; A-170750, Sh. 19

93. G2.1.33 001/3/1/T. S./C/A 4.0/BANK/FA011005/S/

Which one of the following renders 1A Auxiliary Building battery bank INOPERABLE?
Reference Provided

A✓ Average cell float voltage of 2.10 volts.

B. Average electrolyte temperature of the representative cells is 62°F.

C. Electrolyte level greater than 1/4" above the top of the plates but not overflowing.

D. Specific gravity, corrected for electrolyte temperature of 77° F, of 1.191 for one of the connected cells.

Reference provided TS 3.8.6

A. Correct; **Average cell float voltage of 2.10 volts.**
This violates condition "B" of the TS with the required action being a declaration of inoperability. "*One or more required batteries with the average cell float voltage less than or equal to 2.13 volts.*"

B. Incorrect; this satisfies TS as stated in Condition B (>60°F).

C. Incorrect; this satisfies Category "C" criteria of Table 3.8.6-1 of 3.8.6.

D. Incorrect; this satisfies Category "C" criteria of Table 3.8.6-1 of 3.8.6.

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ¼ inch above maximum level indication mark(a)	> Minimum level indication mark, and ¼ inch above maximum level indication mark(a)	Above top of plates, and not overflowing
Float Voltage	2.08 V	2.08 V	> 2.02 V
Specific Gravity(b)	1.195(c)	1.190 AND Average of all connected cells > 1.195	If a cell is < 1.190, then it shall not have decreased more than 0.080 from the previous 92 day test. AND Average of all connected cells 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) Or battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

G2.1.33 Conduct of Operations Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.
(CFR: 43.2 / 43.3 / 45.3)

(2) Facility operating limitations in the technical specifications and their bases.
(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the DC Distribution System (OPS52103C01).

- 3.8.4 DC Sources - Operating
- 3.8.5 DC Sources – Shutdown
- 3.8.6 Battery Cell Parameters
- 3.8.9 Distribution Systems – Operating
- 3.8.10 Distribution Systems - Shutdown

DC DIST-52103C01 #10

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B Auxiliary Building and Service Water Intake Structure (SWIS) batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required batteries with one or more battery cell parameters not within Category A or B limits.	A.1	Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	2 hours
	AND		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND		
	A.3	Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more required batteries with average electrolyte temperature of the representative cells < 60°F for the Auxiliary Building batteries or < 35°F for the SWIS batteries. OR One or more required batteries with one or more battery cell parameters not within Category C values. OR -----------NOTE----------- Battery terminal voltage of 127.8 volts as measured by SR 3.8.4.1 is equivalent to average cell float voltage of 2.13 volts per cell. -------------------------------- One or more required batteries with the average cell float voltage ≤ 2.13 volts.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 7 days after a battery discharge < 110 V AND Once within 7 days after a battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 60°F for the Auxiliary Building batteries and ≥ 35°F for the SWIS batteries.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.08 V	≥ 2.08 V	> 2.02 V
Specific Gravity[b]	≥ 1.195[c]	≥ 1.190 AND Average of all connected cells > 1.195	If a cell is < 1.190, then it shall not have decreased more than 0.080 from the previous 92 day test. AND Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) Or battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

94. G2.2.22 001/3/2/T. S./C/A 4.1/BANK/FA011005/S/CVR

During a review of the surveillance schedules it is discovered that a 31 day surveillance was last performed 42 days ago.

Which one of the following describes the TECHNICAL SPECIFICATION implications of this?

A. The component must be declared inoperable and the associated action statement entered.

B. No Tech Spec actions are required as long as the redundant component in the opposite train remains operable.

C. The surveillance must be performed within 24 hours and the component is considered operable during this time.

D. If during the performance of the surveillance the equipment fails, the applicable conditions of the LCO must be entered immediately.

A. Incorrect. This is only done if the component or system fails its required STP or the performance can't be completed within the time allowed.

B. Incorrect. TS requirements include the surveillances being current for OPERABILITY. If surveillances are not current, and performed within 24 hours of discovering they are out of grace, the equipment is required to be declared INOPERABLE.

C. Incorrect. This would have been true prior to the most recent change to SR 3.0.3.

D. Correct. Based on SR 3.0.3.

SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then ***compliance with the requirement to declare the LCO not met*** **may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is GREATER.** This delay period is permitted to allow performance of the Surveillance........ If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. **When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.**

G2.2.22 Equipment Control- Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2)

(2) Facility operating limitations in the technical specifications and their bases.

5. Describe the application and Bases of LCO Section 3.0 and Section SR 3.0 of Technical Specifications (OPS52302A05).

INTRO TS-52302A09 #5

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

recent change

95. G2.2.29 001/3/2/REFUELING SRO/MEM 3.8/NEW/FA011005/S/CVR/GTO

Unit 2 is in a refueling outage and in MODE 6.

Which one of the following is required to be done by the SRO in charge of refueling in accordance with FNP-2-FHP-1.0, Refueling Operations?

A. To give permission prior to unlatching a fuel assembly in the Reactor Vessel.

B. To document current status of fuel handling operations by keeping status maps up to date.

C. To operate all Manipulator Crane bypass switches and initial for switch postions in FHP-1.0.

D. To be present in Containment **OR** the Spent Fuel Pool Room in constant communications with the refueling team prior to starting core unload or reload.

A. Correct. **To give permission prior to unlatching a fuel assembly in the Reactor Vessel.**
Per FNP-2-FHP-1.0 Step 3.7.

B. Incorrect. Engineering Support does this per FNP-2-FHP-1.0 Step 3.11.

C. Incorrect. The SRO in charge of refueling must give permission, but does not have to be the operator of the bypass switches per FNP-2-FHP-1.0 Step 3.15.

D. Incorrect. The SRO in charge of refueling must be in Containment (SFP is not allowed) prior to unload or reload per FNP-2-FHP-1.0A & B Steps 1.4.

G2.2.29. Equipment Control - Knowledge of SRO fuel handling responsibilities.
(CFR: 43.6 / 45.12)
(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

2. Evaluate abnormal plant or equipment conditions associated with the Fuel Storage, Handling and Refueling System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52108D02).

FARLEY NUCLEAR PLANT

FUEL HANDLING PROCEDURE

FNP-2-FHP-1.0

S
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D

REFUELING OPERATIONS

PROCEDURE USAGE REQUIREMENTS PER FNP-0-AP-6	SECTIONS
Continuous Use	
Reference Use	ALL
Information Use	

Approved:

RAY MARTIN
Operations Manager

Date Issued 6-30-04

FARLEY NUCLEAR PLANT
UNIT 2
REFUELING TOOLS AND EQUIPMENT
PRE-SERVICE INSPECTION/CHECKOUT

1.0 Purpose

1.1 This procedure provides instructions for checkout of the following fuel handling tools and equipment:

ATTACHMENT A	INITIAL CONDITIONS PRIOR TO UNLOAD
ATTACHMENT B	INITIAL CONDITIONS PRIOR TO RELOAD
ATTACHMENT C	INSTALLATION OF THE FUEL ASSEMBLY LOADING GUIDES (SHOE HORN)

2.0 Initial Conditions

2.1 The version of this procedure has been verified to be the current version and the correct unit for the task. (OR 1-98-498)

2.2 Adequate lighting is available for establishing visibility in the Spent Fuel Pool (SFP), fuel transfer canals, New Fuel Elevator and Reconstitution Basket during the fuel transfer equipment checkout.

2.3 An RWP has been obtained for any work in a Radiation Control Area (RCA).

2.4 Foreign Material Exclusion (FME) areas should be established and equipment accountability documented in accordance with FNP-0-ACP-7.0, Foreign Material Exclusion Program.

3.0 Precautions and Limitations

3.1 When performing fuel handling operations, guidelines established by Westinghouse F-5 specifications will be followed. These guidelines have been incorporated into Farley's FHP's. Additional guidelines can be found in Appendix 1 of Westinghouse F-5 specifications.

3.2 Tools and equipment which are withdrawn from the refueling water must be closely monitored for radiation and / or contamination. Coordinate with HP prior to removing tools or equipment from the water.

3.3 All refueling and fuel handling operations will be performed in accordance with Technical Specifications requirements at all times. The appropriate Unit Shift Supervisor's permission must be obtained prior to any fuel handling operations.

3.4 Use applicable fuel handling equipment operating instructions to operate equipment during performance of checkout.

3.5 All core alterations shall be directly supervised by an active licensed Senior Reactor Operator (SRO) in charge of refueling and he will have no other concurrent responsibilities during this operation.

3.6 The Control Room refueling station Engineer shall be stationed in the control room during all fuel loading evolutions. He shall continuously maintain and evaluate an Inverse Count Rate Ratio (ICRR) plot to monitor core criticality during fuel reload in the vessel core. If at any time the ICRR falls below 0.4, the fuel movement will be stopped, the situation will be analyzed, and the SRO in charge of refueling will give permission prior to resuming fuel movement. ICRR monitoring is not required during core unload activities.

3.7 The following shall be performed prior to unlatching a fuel assembly in the reactor vessel:

a) The ZZ axis tape position corresponds to the fuel assembly FULL DOWN position on the lower core plate.

b) A visual check of the fuel assembly is performed.

c) The Control Room refueling station Engineer has verified that source range counts are stable.

d) Digital height indication is correct and corresponds to the fuel assembly FULL DOWN position on the lower core plate.

e) The SRO in charge of refueling has directed that the fuel assembly may be unlatched.

3.8 Access to the refueling work stations will be restricted to members of the refueling team and observers approved by the SRO in charge of refueling.

3.9 A sample shall be taken from the spent fuel pool, reactor cavity, transfer canal, and RHR system and analyzed for boron concentration prior to fuel movement.

3.10 Lifting of the fuel assemblies by means of tools or adapters attached to the assembly must be performed with the assembly in the vertical position only.

3.11 The current status of fuel handling operations must be documented by maintaining up to date status maps and by initialing each step in the refueling sequence as it is accomplished (performed by ES).

3.12 During core off-load and re-load there shall be a copy of the Fuel Handling Data sheets in three locations: the Control Room, CTMT, and the SFP. ES will maintain the control room copy as the master/controlling document. The copies found in CTMT and SFP will be field copies and are only required to be maintained with place-keeping aides. Times are not required to be documented on the field copies.

3.13 The control room copy of the fuel movement record serves as the permanent history of fuel transfer and core loading.

3.14 Any inconsistencies in the loading sequence or in the recording of the fuel movements shall be cause to immediately cease fuel movements and notify the Shift Supervisor and SRO in charge of refueling.

3.15 No bypass switch operation of the manipulator crane shall be allowed without the permission of the SRO in charge of refueling.

3.16 It is permissible to move the manipulator crane bridge and/or trolley using the respective bypasses or jog controls or by disengaging bridge and trolley brakes and manually moving the crane within the reactor core confines, when indexing or when bowed and/or leaning fuel assemblies are encountered, with the hoist in a position other than FULL UP providing that the following conditions are strictly observed:

3.16.1 Adequate lighting has been established in the reactor core area such that the top of the reactor can be readily observed;

AND

3.16.2 The gripper assembly or fuel assembly bottom nozzle shall be a minimum height of 6 inches above the reactor core as read on the Gripper Elevation Indicator (ZZ axis tape) and the digital height indication;

AND

3.16.3 When performing this operation the SRO in charge of refueling and/or spotter shall be observing the movement in the core to protect against any unwanted gripper/fuel assembly or fuel assembly/fuel assembly interaction;

3.17 When encountering bowed or leaning fuel assemblies with adjacent core location(s) devoid of fuel, it is permissible to lower a fuel assembly into an empty adjacent core location and move the manipulator crane bridge and/or trolley by hand or inching device into the proper core location provided that the following conditions are strictly observed:

3.17.1 Adequate lighting has been established in the reactor core area such that the top of the reactor core can be readily observed. During reload, where possible, portable lighting should also be provided on the lower core plate for flow hole and fuel assembly guide pin observation.

AND

3.17.2 The fuel assembly bottom nozzle shall be a minimum of four (4) inches above the lower core plate as read on the Gripper Elevation Indicator (ZZ axis tape) or digital height indication,

AND

3.17.3 When performing this operation the SRO in charge of refueling and/or spotter will be observing the movement in the core to protect against any unwanted gripper/fuel assembly or fuel assembly/fuel assembly interaction,

AND

3.17.4 The manipulator crane bridge and/or trolley shall be positioned towards the correct index,

AND

3.17.5 When lowering the fuel assembly into its proper core location, fuel assembly guide pin engagement should be confirmed by the SRO in charge of refueling and/or spotter if possible.

3.18 Manipulation of the manipulator crane hoist cables by hand is permissible in order to align the gripper assembly with the fuel assembly top nozzle or align the bottom nozzle of the fuel assembly with the lower core plate guide pins.

3.19 The SNC Health Physics Group may perform a radiation survey during the transfer of the first irradiated fuel assembly from the reactor building to the spent fuel storage building and analyze the results before additional fuel transfers can be performed per FNP-0-RCP-4.

3.20 In addition to operation of the transfer system, the SFP upender operator shall:

3.20.1 Verify the spent fuel handling tool at full up (when attached to a fuel assembly) prior to spent fuel bridge or trolley movement. (This may be delegated as long as it is someone in addition to SFP crane operator.)

3.20.2 Do not attempt to raise or lower the upender when the fuel assembly and spent fuel handling tool are in the SFP transfer system canal (the refueling tools must be in the weir gate between the spent fuel pit and the fuel transfer system canal or in the SFP.)

3.21 Voice communications will be established prior to fuel movement and maintained between all refueling stations during all fuel movement. If communications are interrupted fuel movement will stop until communications have been restored. Momentary interruptions to replace head set batteries are allowed as long as no fuel movement between SFP and CTMT buildings is in progress.

3.22 During fuel movement, proper communications shall be strictly maintained as follows:

3.22.1 The minimum number of refueling stations in communication during fuel movement between the reactor vessel and the SFP shall include:

a) SFP upender operator

b) Manipulator crane operator

c) Containment upender operator

d) Control Room refueling station

For all other fuel movements, proper communication will be maintained between the operators involved in the fuel movement.

NOTE: There is a spare sound powered phone circuit between the CTMT and SFP upender operators that may be utilized if the normal circuit is lost.

3.22.2 At least one communication station will be manned in the SFP and CTMT during any unusual occurrence/event.

3.22.3 At no time will any station operator discontinue communication unless first informing the Control Room refueling station.

3.22.4 The Containment upender operator shall remain in communication except if necessary to assist in observation of assemblies in the reactor core.

3.22.5 Manipulator crane and Fuel Transfer System operators shall report illumination of system lights indicating upender position and manipulator mast position during refueling operations. In addition, the above positions shall be visually checked prior to reporting to other station operators in communication.

3.22.6 The Control Room refueling station shall be informed of all movements of the Fuel Transfer System and manipulator crane.

3.23 No fuel shall be moved in CTMT without at least two people on the manipulator crane -- the manipulator crane operator and one other qualified fuel handler.

3.24 All nonessential personnel shall remain off the manipulator crane and no one will board or leave the crane while the crane is in motion.

3.25 If the Tri-Nuclear Filtration System is in use during fuel movement in the Spent Fuel Pit, the following precautions must be observed as stated in SOP-54.3.

3.25.1 If the filter is in the SFP, then fuel assemblies must not be moved over the filter at anytime during fuel movement.

3.25.2 If fuel assemblies are being moved while the filter unit is in the pool, the fuel assemblies may pass over the power cord, hoses, cables, or ropes associated with the filter, provided clearances are maintained.

3.25.3 Prior to fuel movement verify that the Tri-Nuclear System will not interfere with fuel movement.

3.26 Spent Fuel Pit Bridge Crane operator shall perform a self verification of designated storage rack before entering the designated rack per the guidance below. This self verification is to be performed during fuel assembly handling activities and insert change outs.

3.26.1 Self verification: a positive identification of desired location by counting spent fuel racks both for number and letter designation.

3.26.2 Verify desired location by use of alpha/numeric index information.

3.26.3 Prior to lowering any fuel assembly, the operator must verify that the desired location contains NO FUEL ASSEMBLY; i.e. the rack is empty.

3.26.4 SFP Bridge Crane operator should communicate desired and verified location to the SFP upender operator or designated verifier. The upender operator or designated verifier shall concur on the location prior to lowering the fuel assembly.

3.26.5 The Spent Fuel Pit Upender Operator or designated verifier should verify proper location of bridge crane by (1) observing the alpha denoted position from a viewpoint which reduces parallax to ensure correct position and (2) locating himself at the bridge crane and observing numeric position from a viewpoint which reduces parallax.

3.27 The following considerations should be incorporated during fuel loading.

3.27.1 Look for non-uniformity or substantial misalignment after each fuel assembly is seated. Take a global view, look at surrounding assemblies and adjacent rows.

3.27.2 When completing rows (i.e., loading the last assembly in a row, adjacent to the baffle), complete the third row out from the baffle first, the second row out from the baffle next, and the baffle row last.

3.27.3 Previous core verification tapes should be given attention for prior assembly-to-assembly gaps, assembly-to-baffle gaps, and top-nozzle to top-nozzle corner-to-corner offsets. Take note of the size and distribution of the gaps and offsets. This should be done to establish normal gaps.

3.27.4 Inspect all assembly-to-assembly gaps, assembly-to-baffle gaps, and top nozzle corner-to-corner junctions.

NOTE: If any gap exceeding .25 inch is noticed, refer to Westinghouse Technical Bulletin - Damaged Fuel Assemblies During Refueling. For further action, notify Westinghouse refueling shift supervisor or service lead.

3.28 Determine if any fuel assemblies' top nozzles (corner-to-corner) are misaligned or assembly-to-assembly gaps and assembly-to-baffle gaps exceed previously documented levels.

If misalignments exist, the following actions are recommended:

3.28.1 Review all gaps and nozzle-to-nozzle misalignments in the row and at the baffle to determine the source of the misalignment, i.e. is it a build-up across the row involving several assemblies locked in mechanically by the crossing row?

3.28.2 After determining the source(s) of any misalignment and deciding to take action, use normal fuel handling procedures and remove designated assemblies and re-seat them in the core as appropriate. Consider seating the assembly(s) judged to be the sources(s) of the displacement last.

3.28.3 Re-inspect all gaps and misalignments that may have been changed by reseating assemblies.

3.28.4 If a gap is observed that is greater than 1/4" (inches), notify the SRO in charge of refueling.

3.29 A step by step fuel movement sequence verification shall be performed on the Fuel Status Map prior to actual fuel movement.

3.30 A camera may be used to verify the fuel assembly number and insert type (if any) of the fuel assembly at the Spent Fuel Pit upender prior to transferring the fuel assembly to containment for reloading into the Reactor Core.

3.31 It is permissible to temporarily store a fuel assembly at any empty core location provided the fuel assembly is supported on at least one (1) face by a baffle. Temporary storage of fuel assemblies is also permitted in the upender, RCC change fixture compartments or new fuel elevator. Prior to storing a fuel assembly in a temporary location, permission must be obtained from the SRO in charge of refueling and the Control Room refueling station Engineer.

3.32 **If** a fuel assembly or fuel component should be placed in any location or handled in any way not directed by this procedure or a Fuel Assembly Handling Deviation Form (FAHDF), **THEN all fuel handling activities will stop.** The fuel assembly, component, and equipment should be placed in a safe condition and the need for a recovery plan evaluated. Recovery plan steps may be written as additional steps to this procedure or written as a separate procedure using FNP-0-SOP-0.2 as guidance. In either case, the recovery plan and actions to prevent reoccurrence should be discussed with the Shift Supervisor and the OSS. The recovery plan and approval documentation will be attached to the FAHDF and/or fuel transfer forms. **Prior** to continuing with fuel movement after recovery operations, permission shall be obtained from the OSS.

NOTE:	**If the new fuel elevator is to be used for temporary storage, the guidance listed below must be strictly followed.**

3.33 If fuel is temporarily stored in the new fuel elevator, establish the following conditions:

3.33.1 The elevator must be in the full DOWN position.

3.33.2 The electrical supply breaker for the elevator must be OPEN and TAGGED OUT. This step need not be accomplished if a new fuel element with no irradiated insert is temporarily stored.

3.34 All irregularities noted during fuel movement shall be logged by ES or annotated on a Fuel Assembly Handling Deviation Form. These irregularities include temporary fuel assembly storage, temporary insert storage, bridge and/or trolley OFF index movement, fuel assembly twist or bow, equipment malfunctions, etc., but the final core loading shall not deviate from the Reactor Engineering final core drawing.

3.35 An independent verification of placement in each respective storage location by the upender operator or designated verifier shall be performed in accordance with FNP-2-STP-107 and initialed on the core unload data sheets in the Spent Fuel Pit copy of this procedure.

3.36 Ensure the Fuel Assembly Loading Guide (FLG or shoe horn) cables are secured by some means to maintain the cables clear of fuel movements.

3.37 Due to spatial storage constraints relative to fuel assembly enrichment and burn up relative to Technical Specification requirements, two storage locations have been evaluated and reviewed for contingency storage locations for use with FAHDFs. These contingency storage locations are designated by Engineering Support. These two locations have been reviewed and determined to meet Technical Specification storage requirements for all potential fuel movements for this refueling procedure. If an FAHDF is necessary in the SFP area during fuel offload, the preferred placement for an assembly is the designated destination storage cell in the SFP for the affected assembly. If an FAHDF is necessary in the SFP area during fuel reload, the preferred placement for an assembly is the cell where the assembly originated from. In both of these cases or in any other scenario, the pre-evaluated cells described above should be used wherever possible. In the event an FAHDF is required for fuel movement into or within the SFP, the independent verification requirements of FNP-2-STP-107 must also be provided for and documented by the FAHDF. In the event of an FAHDF of the above nature, a copy of the completed FAHDF with initials of assembly placement verification shall be forwarded to an FNP Reactor Engineer by the end of the shift, to allow completion of an FNP-2-STP-107 review within the required seven (7) day period.

3.38 Moving the dummy fuel assembly or dummy RCCA from the spent fuel pool to transfer system or from the transfer system to the SFP requires a Fuel Transfer Form. Movements of dummy assembly or dummy RCCA within the Fuel Transfer System, including moves to the CTMT RCC change fixture, do not require a Fuel Transfer Form.

3.39 When using the dummy fuel assembly, ensure the dummy fuel assembly is inserted in the ELEVATOR/BASKET or the upender in the correct orientation.

Unit 1 and Unit 2 - Reference hole in Southeast corner

3.40 Observe and note any abnormal equipment conditions.

3.40.1 Check for twists, kinks, distortion, excessive wear and improper deadening of all wire ropes.

3.40.2 Check for any deformation/corrosion of Crane hooks.

3.40.3 Check for damage to controllers or improperly adjusted limit switches.

3.41 Complete reliance on limit switches and indicating lights to protect the fuel handling tools and equipment during checkout is not recommended. Visual observation is required during the handling of the various tools and equipment to preclude possible damage.

3.42 All tools and equipment listed in FNP-2-FHP-1.0 are not required for a particular phase of work. Initial the work in FNP-2-FHP-1.0 for which the checkout is required and N/A all subsection sign-offs not required.

3.43 Any equipment, tools, etc. that comes in contact with the fuel or is used in the vicinity of the Spent Fuel Pools, Fuel Transfer Canals, the Cask Wash Pit or the Cask Storage Pit shall be inspected and determined to be free of debris and loose parts.

3.44 If a loose part is found, an attempt to identify the source of the part shall be made along with efforts to ensure that all pieces are accounted for.

3.45 Nuts and bolts identified as requiring tightening are to be tightened in accordance with approved procedures. Tools used are to be made fail-safe. (Fail-safe tools are too large to fit through system openings, have no loose parts and are not easily broken. They may be used if they are tied off with lanyards).

4.0 Instructions

4.1 Perform the desired attachment(s) as necessary to complete a core unload or reload.

5.0 REFERENCES

New Fuel Handling Tool	FNP-0-FHP-5.12
New RCCA Handling Tool	FNP-0-FHP-5.6
Spent Fuel Assembly Handling Tool	FNP-1/2-FHP-5.4
BPRA Handling Tool	FNP-0-FHP-5.10
Irradiation Sample handling Tool	FNP-0-FHP-5.5
Control Rod Drive Shaft Unlatching Tool	FNP-0-FHP-5.1
New Fuel Elevator (NFE)	FNP-1/2-FHP-5.17
Manipulator Crane	FNP-1/2-FHP-5.13
Thimble Plug Handling Tool	FNP-0-FHP-5.3
Rod Cluster Control Assembly (RCCA) Change Tool	FNP-0-FHP-5.14
Manipulator Crane Q2F15K0001 Load Measuring System Calibration	FNP-2-IMP-270.1

U279889 "Manipulator Crane Field Checkout"

Westinghouse Drawing 1879E48, Rev 2, Right Angle Fuel Assembly Loading Guide

FNP-2-FHP-1.0A
June 16, 2004

FARLEY NUCLEAR PLANT
UNIT 2
FNP-2-FHP-1.0A

CONTROLLING PROCEDURE FOR CORE UNLOAD

Completed By ______________________________ Date ______________

Reviewed By ______________________________ Date ______________

This appendix consists of 5 pages.

CONTROLLING PROCEDURE FOR CORE UNLOAD

1.0 Initial Conditions

NOTE: These Initial Conditions MUST be met (with the exception of tool checkouts for tools that will not be used for fuel offload) before starting the core unload and may be signed in any order.

Initials

_____ 1.1 FNP-2-STP-35.1F for mode 5 and 6 surveillances are met.

_____ 1.2 FNP-2-STP-35.1G for mode 5 and 6 surveillances are met.

_____ 1.3 Verify that no LCO's exist that prevent fuel movement.

_____ 1.4 The SNC Refueling SRO and the refueling team are in containment to monitor core alterations.

_____ 1.5 Underwater lights are positioned in the vessel such that they will not interfere with core alterations.

1.6 Ensure that the following equipment checks have been completed (checks not needed (i.e., equipment not to be used for fuel unload) may be marked N/A):

_____ Manipulator Crane Pre-Operational Checks (FNP-2-FHP-5.13A)

_____ Manipulator Crane Check Out (FNP-2-FHP-5.13B) or Manipulator Crane Field Checkout Procedure covered in U279889.

_____ Fuel Transfer System Checkout (FNP-2-FHP-5.11A)

_____ RCCA Latch/Unlatch Tool Checkout (FNP-0-FHP-5.1A)

_____ RCCA Latch/Unlatch Operation Checkout (FNP-0-FHP-5.1B)

_____ Spent Fuel Assembly Handling Tool Checkout (FNP-2-FHP-5.4A)

_____ RCCA Change Fixture Checkout (FNP-0-FHP-5.9A)

_____ BPRA Handling Tool Checkout (FNP-0-FHP-5.10A)

_____ Thimble Plug Handling Tool Checkout (FNP-0-FHP-5.3A)

_____ Portable RCCA Change Tool Checkout (FNP-0-FHP-5.14A)

1.7 Check the manipulator crane limit switches for proper orientation by physical observation of limit switch. (Reference Figure 1)

_____ LS-4 Gripper Tube Up Backup Limit

_____ LS-6 Bridge Left Reactor Limit

_____ LS-7 Trolley on Transfer System Centerline

_____ LS-11 Bridge Right End Limit

_____ LS-12 Bridge/Trolley Permission in Basket Area

_____ LS-14 Trolley Forward End Limit

_____ LS-17 Trolley Reverse Limit in Transfer Area

_____ 1.8 Position the Tri-Nuclear filtration system to provide the least interference with fuel movement in the spent fuel pool.

_____ 1.9 Position the Tri-Nuclear filtration system to provide the least interference with fuel movement in the lower reactor cavity.

_____ 1.10 The initial conditions of FNP-2-UOP-4.1 are satisfied.

_____ (ES) 1.11 A step by step unloading sequence has been developed and independently verified.

_____ (ES)

_____ 1.12 A pre-job briefing has been conducted with all personnel who will be involved in fuel handling activities during the core unload, (per FNP-0-AP-92, Attachment 1).

_____ 1.13 Health Physics Foreman notified that core unload prerequisites are complete.

_____ 1.14 Index the manipulator crane over core location H-8, grip the assembly, raise it six (6) inches, and replace it in the core and unlatch the assembly.

_____ 1.15 Permission to begin core unload by Operations Shift Supervisor.

2.0 Precautions and Limitations for Fuel Assembly and Core Component Movement

_____ 2.1 Review Precautions and Limitations for Fuel Assembly and Core Component Movement at the beginning of FNP-2-FHP-1.0.

3.0 Core Unload

_____ 3.1 Unload in accordance with the following procedures:

Position to be manned	Procedure to be utilized	Procedures used by all
SFP Upender Operator & CTMT Upender Operator	FNP-2-FHP-5.11 Copy of Off-Load Sequence	FNP-0-FHP-1.0 FNP-0-FHP-7.0
SFP Crane Operators	FNP-2-FHP-5.4 FNP-2-FHP-5.18	
CTMT Manipulator Operator	FNP-2-FHP-5.13 Copy of Off-Load Sequence FNP-2-AOP-30.0	
Control Room Engineering Team	FNP-0-ETP-3637 FNP-0-ETP-4423 FNP-2-STP-107.0 Master Copy of Off-Load Sequence	
Nuclear Fuels Inspector in SFP	FNP-0-ETP-3636	

NOTE: **The intent of the following step is to have the dummy assembly in CTMT for equipment checkouts while the transfer tube gate valve is closed during mid-loop operations. IF for some reason the assembly can not be placed in the RCC change fixture, THEN the following may be marked N/A and place a comment on why it could not be stored in the RCC change fixture and where it is currently located.**

_____ 3.2 Verify the dummy assembly is stored in the RCC change fixture basket in CTMT.

_____ 3.3 Verify transfer cart is at the SFP per FNP-2-FHP-5.11.

_____ 3.4 Close the SFP transfer tube gate valve.

3.5 Perform insert shuffle per the following procedures:

3.5.1 ES Insert Shuffle Move Sheets

3.5.2 FNP-2-FHP-5.14

3.5.3 FNP-2-FHP-5.18

FNP-2-FHP-1.0B
June 16, 2004

FARLEY NUCLEAR PLANT
UNIT 2
FNP-2-FHP-1.0B

CONTROLLING PROCEDURE FOR CORE RELOAD

Completed By ______________________________ Date ______________

Reviewed By ______________________________ Date ______________

This appendix consists of 5 pages.

CONTROLLING PROCEDURE FOR CORE RELOAD

2.0 Initial Conditions

NOTE: These Initial Conditions MUST be met (with the exception of tool checkouts for tools that will not be used for fuel offload) before starting the core reload and may be signed in any order.

Initials

_____ 1.1 FNP-2-STP-35.1F for mode 5 and 6 surveillances are met.

_____ 1.2 FNP-2-STP-35.1G for mode 5 and 6 surveillances are met.

_____ 1.3 Verify that no LCO's exist that prevent fuel movement.

_____ 1.4 The SNC Refueling SRO and the refueling team are in containment to monitor core alterations.

_____ 1.5 Underwater lights are positioned in the vessel such that they will not interfere with core alterations.

NOTE: Equipment/tool checkouts should be performed at least once per outage. This is normally done at the beginning of the outage. It is at the discretion of the Refueling Supervisor whether the equipment/tool checkouts should be performed again prior to core reload. Maintenance performed or other activities involving the individual tools and equipment should be addressed when making this determination. Steps may also be omitted and marked N/A when they are not feasible to be performed.

1.6 Ensure that the following equipment checks have been completed (checks not needed (i.e., equipment not to be used for fuel reload) may be marked N/A):

_____ Manipulator Crane Pre-Operational Checks (FNP-2-FHP-5.13A)

_____ Manipulator Crane Check Out (FNP-2-FHP-5.13B)

_____ Fuel Transfer System Checkout (FNP-2-FHP-5.11A)

During a reentry, the reentry team finds an injured employee. The rescue will take the team on a different route than planned.
The reentry team:

A. is required to withdraw to a safe area and contact the TSC for further instructions.

B. is allowed to deviate from the planned route only if the OSC agrees to the new route.

C. is allowed to deviate from the planned route and should call the OSC as soon as possible.

D. is required to withdraw to a safe area until a radiation monitoring team conducts a survey to accurately determine dose rates.

A. Incorrect, This is not required if an injured person is involved and does not exceed emergency dose rates.

B. Incorrect, EIP 14 allows deviation for injured personnel.

C. Correct - **is allowed to deviate from the planned route and should call the OSC as soon as possible.**
EIP-14 states:
Re-entry personnel shall not deviate from a planned route **unless unanticipated conditions such as rescue,** performing an operation which would minimize the emergency condition, etc., require such a deviation.

If emergency dose rates observed during re-entry exceed the limits established by the re-entry guideline or other adverse conditions are encountered, re-entry personnel shall return to a safe area and contact the OSC/TSC/Control Room.

Figure 3 of EIP-14 also says it is acceptable for the team to modify the transit route based on the conditions encountered during the reentry. If the route is modified, the OSC or CR should be notified asap if the change places the team in areas that are not on the route.

D. Incorrect- This is not required with an injured person involved. In any reentry it is not required to get an HP monitoring team to come do surveys for you. The team should have HP with them anyway.

G2.3.1 Radiation Control
2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements.(CFR: 41.12 / 43.4. 45.9 / 45.10)

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

5. Using the appropriate EIPs, classify abnormal events, make notifications, and apply the required actions (OPS-52504A05).

modified from EPIP PERS-40501B09 #3

QUESTIONS REPORT

for Plant Systems Questions 6-11-2004

EPIP PERS-40501B09 003/HLT/LOCT/SOIT/SOCT/RO/C/A (LEVEL 2/3) PROC/G2.3.1///LOCT/SOCT/

During a reentry, the reentry team encounters dose rate higher than the limits established by the emergency director. The reentry team should:

A. Seek an alternate route.

B. Proceed rapidly through the area and check their dosimetry when they are in a lower dose rate.

C. Withdraw to a safe area until a radiation monitoring team conducts a survey to accurately determine dose rates.

D. Withdraw to a safe area and contact the technical support center (TSC) for further instructions.

not entirely accurate if Fig 3 is Coded @

Reference
EIP-14
EIP-14 states:
Re-entry personnel shall not deviate from a planned route unless unanticipated conditions such as rescue, performing an operation which would minimize the emergency condition, etc., require such a deviation.

If emergency dose rates observed during re-entry exceed the limits established by the re-entry guideline or other adverse conditions are encountered, re-entry personnel shall return to a safe area and contact the OSC/TSC/Control Room

A. Incorrect, This could result in passing through areas that have a higher dose rate than allowed.

B. Incorrect, EIP 14 requires reenetry personnel to withdraw to a safe area.

C. Incorrect, The team should contact the OSC/TSC/Control Room without exposing any personnel to additional dose

D. Correct

7.0 Requirements for Re-entry - General Guidance

7.1 TLD badges of personnel who receive an emergency exposure in excess of the 10CFR20 limits of step 7.10 will be pulled and read prior to receiving further non-emergency exposure.

7.2 Re-entry personnel shall not deviate from a planned route unless unanticipated conditions such as rescue, performing an operation which would minimize the emergency condition, etc., require such a deviation.

7.3 If emergency dose rates observed during re-entry exceed the limits established by the re-entry guideline or other adverse conditions are encountered, re-entry personnel shall return to a safe area and contact the OSC/TSC/Control Room for further instructions.

7.4 If the Plant Emergency Alarm (PEA) is sounded while a re-entry team is involved in their assigned tasks, the re-entry team shall call the Control Room/TSC/OSC and request further instructions for assembly requirements.

7.5 The re-entry guideline/log (Figures 3/4) will serve as a tracking mechanism for re-entries. One copy of the guideline will remain with the OSC and, if desired, another copy will be given to the re-entry team leader. The guideline may be photocopied, or a two-part form may be used. The re-entry guideline will be sequentially numbered.

7.6 Individuals listed on the re-entry guideline as responsible for completion of guideline items are not required to personally initial the guideline, but are responsible for ensuring that each requirement is performed and initialed by the person performing or ensuring performance of the task.

7.7 Radiological monitoring will be established for each re-entry. The following parameters will be considered when determining the degree of radiological monitoring:

- Releases in progress
- Dose rates, airborne and contamination levels
- Stability of plant radiological conditions

7.8 Re-Entry is the responsibility of the Emergency Director, and requires verbal ED approval to execute a re-entry. Re-entries may be authorized and executed by the OSC Manager or Maintenance Supervisor, with ED approval. Approval to exceed 10CFR20 radiation exposure limits listed in step 7.10 must be approved by the

COMBINED BRIEF SECTION III

Conduct a pre-job brief of the Re-Entry. The following information must be included:

- Duties for the re-entry including required procedures and safe work practices. Reference the OSC managers section and the re-entry duties section.
- Hazards associated with the assigned tasks (Radiological and Non Radiological)
- Dose and dose rate limits while performing the re-entry (per Health Physics section)
- Personnel protective equipment required (per Health Physics section if radiological)
- Isolation and control of energy sources (Clearance)
- Special support needs and precautions
- Transit route. It is acceptable for the team to modify the transit route based on the conditions encountered during the re-entry. If the route is modified, the OSC or control room should be notified as soon as possible if the change places the team in areas that are not on the route.
- Communications and actions to take if communications cannot be established

The following information may be considered in the pre-job briefing:

- Industry experience
- Plant or equipment conditions including potential radiological or industrial safety hazards and precautions
- Each person's job or task assignment
- Expected sequence of events and results
- Problems to be anticipated
- Criteria to be used to stop the evolution
- Contingencies if the evolution is stopped or the expected result is not achieved
- Potential distractions and how they will be minimized
- Housekeeping and fluid system cleanliness requirements
- Chemical control and disposal requirements
- Foreign Material Exclusion (FME) Controls

7.0 Requirements for Re-entry - General Guidance

7.1 TLD badges of personnel who receive an emergency exposure in excess of the 10CFR20 limits of step 7.10 will be pulled and read prior to receiving further non-emergency exposure.

7.2 Re-entry personnel shall not deviate from a planned route unless unanticipated conditions such as rescue, performing an operation which would minimize the emergency condition, etc., require such a deviation.

7.3 If emergency dose rates observed during re-entry exceed the limits established by the re-entry guideline or other adverse conditions are encountered, re-entry personnel shall return to a safe area and contact the OSC/TSC/Control Room for further instructions.

7.4 If the Plant Emergency Alarm (PEA) is sounded while a re-entry team is involved in their assigned tasks, the re-entry team shall call the Control Room/TSC/OSC and request further instructions for assembly requirements.

7.5 The re-entry guideline/log (Figures 3/4) will serve as a tracking mechanism for re-entries. One copy of the guideline will remain with the OSC and, if desired, another copy will be given to the re-entry team leader. The guideline may be photocopied, or a two-part form may be used. The re-entry guideline will be sequentially numbered.

7.6 Individuals listed on the re-entry guideline as responsible for completion of guideline items are not required to personally initial the guideline, but are responsible for ensuring that each requirement is performed and initialed by the person performing or ensuring performance of the task.

7.7 Radiological monitoring will be established for each re-entry. The following parameters will be considered when determining the degree of radiological monitoring:

- Releases in progress
- Dose rates, airborne and contamination levels
- Stability of plant radiological conditions

7.8 Re-Entry is the responsibility of the Emergency Director, and requires verbal ED approval to execute a re-entry. Re-entries may be authorized and executed by the OSC Manager or Maintenance Supervisor, with ED approval. Approval to exceed 10CFR20 radiation exposure limits listed in step 7.10 must be approved by the

Emergency Director. Approval to exceed plant administrative dose limits listed in step 7.10 must be approved by the HP Supervisor, or the Emergency Director in the HP Supervisor's absence.

7.9 An Emergency Repair Party which functions as a re-entry team shall consist of at least two (2) persons.

7.10 Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to the following 10CFR20 limits:

	10CFR20 limit	Administrative limit
Whole body (TEDE)	- 5 rem	- 2 rem
Lens of the eyes	- 15 rem	- 6 rem
Skin of the whole body	- 50 rem	- 20 rem
Extremities	- 50 rem	- 20 rem
Internal organs	- 50 rem	- 20 rem

7.11 Dosimetry records for potential re-entry team members are available in the Dosimetry Lab.

CAUTION: EMERGENCY EXPOSURE LIMITS SHALL ONLY BE AUTHORIZED BY THE E.D.

7.12 Emergency situations may transcend the normal requirement of maintaining personnel exposures below 10CFR20 limits, as noted in step 7.10. Emergency exposures shall be minimized to every degree practicable. Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to 25 rem TEDE for the activity and conditions described below. For those same personnel to receive in excess of 25 rem, they must voluntarily agree to receive an emergency dose in excess of 25 rem, but less than 100 rem. Persons volunteering to receive in excess of 25 rem must be made fully aware of the risks involved. Emergency exposure limits are as follows:

TEDE DOSE	ACTIVITY	CONDITION
10 REM	PROTECTING VALUABLE PROPERTY	LOWER DOSE NOT PRACTICAL
25 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	LOWER DOSE NOT PRACTICAL
>25, <100 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	VOLUNTEERS ONLY THAT ARE FULLY AWARE OF THE RISKS INVOLVED
Limit the dose to the lens of the eyes to 3 times the listed value. Limit the dose to other organs, including skin and extremities to 10 times the listed values.		

NOTE: THERE IS CURRENTLY NO METHOD AVAILABLE TO ASSESS INTERNAL EXPOSURE ON A REAL TIME BASIS; THEREFORE, FARLEY NUCLEAR PLANT WILL UTILIZE AN ADMINISTRATIVE DEFAULT CORRECTION FACTOR OF TWO TO RELATE DEEP DOSE TO EXTERNAL EXPOSURE.

7.13 If an internal hazard is present, the limit for the re-entry should be reduced by a factor of two, unless a compensatory measure has been used to eliminate the internal hazard.

8.0 Requirements for Re-entry - Specific Guidance

8.1 The Emergency Director must verbally approve all re-entries.

8.2 The ED must approve doses that exceed the 10CFR20 limits of step 7.10.

8.3 The HP Supervisor or ED will complete the applicable portions of section II of the re-entry Guideline.

8.4 The HP Supervisor or designee will complete section IV of the Re-Entry Guideline.

8.5 The OSC Manager or Maintenance Supervisor (or ED, if OSC and Maintenance Manager are not available) will coordinate the re-entry and complete sections I (OSC Manager Section) and V (Debrief Section) of the Re-Entry Guideline. Section III is a combined brief section to be completed by the OSC Manager or designee and the HP Supervisor or designee.

97. G2.3.6 001/3/3/LWRP/MEM 3.1/NEW/FA011005/S/CVR/GTO

Both Units are at 100% power with normal Service Water system alignments and Dilution Flow rates. The Shift Radiochemist has issued a Liquid Waste Release Permit (LWRP) for Unit 1 Waste Monitor Tank (WMT) #1. The SSS is reviewing the LWRP in preparation for issuing the key to the Locked closed discharge valve for WMT #1.

Which one of the following conditions requires the SSS to **NOT** issue the Key and prevent the release per FNP-1-SOP-50.1, APPENDIX 1, Waste Monitor Tank 1 Release to the Environment?

A. There is a Unit 2 WMT release in progress.

B. The Unit 1 SW dilution flow recorder is INOPERABLE.

C. The Unit 1 WMT Flow indicator, FI-1085B, is INOPERABLE.

D. The key for the Unit 1 WMT #2 Discharge Valve has been returned and the valve position is logged as "Unlocked Open".

A. Incorrect. Chemistry would require only one WMT tank per unit released at a time per FNP-0-CCP-212, LIQUID WASTE RELEASE PROGRAM, Step 4.10 only IF either unit was <20,000 gpm SW dilution flow. This occasionally occurs. In the case given, normal 100% power dilution flow was stated which is well above 20,000 gpm on each unit. The operations procedure SOP-50.1 does not address this low dilution flow condition and the requirement for one WMT release at a time between both units is directed by Chemistry when necessary (i.e. when either unit dilution flow is <20,000). Normally one WMT per unit can be released at one time.

B. Incorrect. This does not prohibit releasing the WMT. As long as the SW Dilution totalizer is operable the SSS does not even have to be notified by the SO per Note prior to step 2.1 fifth bullet in SOP-50.1 APP 1. The ODCM table 2.1 states that as long as flow is extimated at least once every 4 hours, the release can continue in this condition.

C. Incorrect. This does not prohibit releasing the WMT. As long as the the pump curves are used to estimate the flowrate per SOP-50.1 APP 1 Step 2.5.14 the release can occur. This is also allowed per the ODCM table 2.1.

D. Correct. **The key for the Unit 1 WMT #2 Discharge Valve has been returned and the valve position is logged as "Unlocked Open".**

The main function of the SRO during approval of the release is the initialed step 2.5.3. It states the the release permit is reviewed, the correct tank is being released, and the other tank key is signed in with the valve logged "locked closed". If this is the case, the key may be issued. With the other WMT discharge valve "unlocked and open" as given in this situation the wrong tank could be released.

G2.3.6 Radiation Control, Knowledge of the requirements for reviewing and approving release permits. (CFR: 43.4 / 45.10)

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

2. Identify any special considerations such as safety hazards and plant condition changes that apply to the Liquid and Solid Waste System (OPS52106A04).

FARLEY NUCLEAR PLANT

UNIT 1

APPENDIX 1

WASTE MONITOR TANK 1

RELEASE TO THE ENVIRONMENT

NOTES:
1. **This procedure and the liquid waste release permit, when issued, must be in the possession of the systems operator at all times during the performance of this procedure.**
2. **Each step must be signed off immediately after completion of that step.**
3. **______/______ indicates verification required.**

Completed by: ________________________________ Date ______________

Verified by: ________________________________ Date ______________

Reviewed by: ________________________________ Date ______________

This appendix consists of 10 pages.

FARLEY NUCLEAR PLANT

UNIT 1

APPENDIX 1

WASTE MONITOR TANK 1

RELEASE TO THE ENVIRONMENT

1.0 Initial Conditions

_____ 1.1 The version of this appendix has been verified to be the current version and correct unit for the task. (OR 1-98-498)

2.0 Instructions

CAUTION: DO NOT enter the "TEST/CALIB" or the "PARAM SET" modes of the WMT totalizer processor. Any changes in these modes will affect the channel calibration.

NOTE:

- **Either WMT totalizer (N1G21FQI1085) OR WMT level indicator (N1G21LIS1082) should be operable to discharge a WMT. IF both instruments are inoperable, THEN the Shift Support Supervisor should be notified.**
- **IF the WMT totalizer (N1G21FQI1085) is out of service, THEN the totalizer's reading may be derived from the WMT level indication (N1G21LIS1082). One gallon discharged from the WMT equals one gallon on the WMT totalizer. (Either the WMT totalizer or the WMT level indication should be operable)**
- **IF the WMT level indicator (N1G21LIS1082) is out of service, THEN the volume released (Net Tank Level) may be derived from the net "INVENT" totalizer reading or the "TOTAL" totalizer reading.**
- **IF the service water dilution totalizer is (N1P16FQI4126) is out of service, THEN the dilution totalizer reading may be derived by taking the average dilution flowrate during the release, multiplying this by the release duration (net time), and dividing by 10,000. Each totalizer integration equals 10,000 gal of water.**
- **Either the SW Dilution totalizer (N1P16FQI4126) OR the SW Dilution flow recorder (N1P16FR4107) should be operable to discharge a WMT. IF both instruments are inoperable, THEN the Shift Support Supervisor should be notified.**
- **Technical Specification 5.5.1 and Chapter 2 of FNP-0-M-011, OFFSITE DOSE CALCULATION MANUAL, are applicable to the WMT pump flow indicator (N1G21FI1085B) AND the SW Dilution flow recorder (N1P16FR4107).**
- **Initial each step as completed. Instructions for completion of liquid waste release permit are specified in FNP-0-CCP-212, LIQUID WASTE RELEASE PROGRAM.**

Initial

_____ 2.1 Waste monitor tank 1 is aligned per system check list FNP-1-SOP-50.1A.

_____ 2.2 IF waste monitor tank 1 is to be recirculated through the Demineralizer, THEN proceed to step 2.3. IF normal recirculation is to be used, THEN proceed to step 2.4.

CAUTION: Only one tank may be recirculated through the demineralizer at a time.

CAUTION: IF WMT is aligned for recirc through the demin while WCT is being pumped to the Disposable Demineralizer System, THEN the two tanks will be cross-connected.

NOTE: Recirculation time through the demineralizers does not count towards recirculation time necessary to obtain a representative sample. Section 2.4 must be performed. (Ref. FNP-1-CCP-212)

2.3 Recirculate through Demineralizer as follows:

_____ 2.3.1 Verify closed FDT Strainer Discharge valve 1-LWP-V-7421 (N1G21V118), then start Waste Monitor Tank Pump #1.

_____ 2.3.2 Close FDT disch to Waste Evap 1-LWP-V-7427 (Q1G21V267).

_____ 2.3.3 Close WMT demineralizer bypass valve 1-LWP-V-7428 (Q1G21V125).

_____ 2.3.4 Open FDT filter outlet valve 1-LWP-V-7456 (Q1G21V189).

_____ 2.3.5 Open floor drain tank discharge valve 1-LWP-V-7452B (Q1G21V091B) to WMT demineralizer.

_____ 2.3.6 Verify closed WMT demin outlet valve to WMT #2, 1-LWP-V-7438B (Q1G21V093B).

_____ 2.3.7 Open WMT demineralizer outlet valve 1-LWP-V-7438A (Q1G21V093A) to WMT #1.

_____ 2.3.8 Open WMT demin outlet to WMT filter 1-LWP-V-7434 (Q1G21V094).

_____ 2.3.9 Verify open WMT #1 inlet iso 1-LWP-V-7413 (Q1G21V090).

_____ 2.3.10 Open WMT #1 discharge valve 1-LWP-V-7454 (Q1G21V115) to FDT filter.

_____ 2.3.11 Throttle open WMT #1 pump discharge valve 1-LWP-V-7443B (N1G21V108B).

____ 2.3.12 Recirculate tank contents for a minimum of two (2) tank volumes based on the flow rate from either FI1085B, totalizer processor "RATE" display or the release flow rate corresponding to the discharge pressure obtained using Table 1, and the tank volume.

Disch Press _______ psig Flow Rate ______ gpm

Recirculation Time ___________ hr. _________ min.

____ 2.3.13 Verify WMT filter ΔP < 20 psid.

Inlet Press __________ (N1G21PI2907) - Outlet Press __________ (N1G21PI1089) = __________ psid

____ 2.3.14 IF WMT filter ΔP ≥ 20 psid, THEN contact control room and submit a DR to change the filter.

____ 2.3.15 WHEN recirculation is completed, THEN close waste monitor tank #1 pump discharge valve 1-LWP-V-7443B (N1G21V108B).

____ 2.3.16 Close WMT #1 discharge valve 1-LWP-V-7454 (Q1G21V115) to FDT filter.

____ 2.3.17 Close WMT #1 inlet valve 1-LWP-V-7413 (Q1G21V090).

____ 2.3.18 Close WMT demin outlet to WMT filter 1-LWP-V-7434 (Q1G21V094).

____ 2.3.19 Close WMT demineralizer outlet valve 1-LWP-V-7438A (Q1G21V093A) to WMT #1.

____ 2.3.20 Close FDT discharge valve 1-LWP-V-7452B (Q1G21V091B) to WMT demineralizer.

____ 2.3.21 Open FDT disch to Waste Evap 1-LWP-V-7427 (Q1G21V267).

____ 2.3.22 Open WMT demineralizer bypass valve 1-LWP-V-7428 (Q1G21V125).

____ 2.3.23 Close FDT filter outlet valve 1-LWP-V-7456 (Q1G21V189).

____ 2.3.24 Proceed to step 2.4 for recirculation to obtain sample.

2.4 Align normal recirc to #1 WMT as follows:

____ 2.4.1 Close waste monitor tank inlet valve 1-LWP-V-7413 (Q1G21V090).

____ 2.4.2 Verify closed, WMT #1 Pump Discharge valve 1-LWP-V-7443B (N1G21V108B).

____ 2.4.3 Start waste monitor tank pump #1 and record #1 WMT PUMP DISCH PRESS N1G21PI1084A.

_______________ psig

____ 2.4.4 <u>IF</u> pressure recorded in step 2.4.3 ≥ 103 psig, <u>THEN</u> continue with 2.4.5, <u>IF NOT</u>, the flow value specified in step 2.4.5 is not conservative and will have to be re-determined using Appendix 3 prior to proceeding with this release.

____ 2.4.5 Recirculate tank contents for a minimum of two (2) tank volumes based on the following flow rate and the volume of liquid in the tank.

Flow Rate ___38.5_____ gpm

Recirculation Time ___________ hr. ________ min.

____ 2.4.6 Sample has been drawn and analyzed after required recirculation time.
CHEM

____ 2.4.7 Proceed to step 2.5.

2.5 Liquid Waste Release

____ 2.5.1 Liquid waste release permit no. ______________________________ has been issued.

____ 2.5.2 Calculate time that tank low level and pump shutoff will occur by using the following formula.

$$\text{Time (minutes)} = \frac{A - 750\ (\text{gal})}{B}$$

A = Quantity of liquid to be released from permit, gal.

B = Expected Release rate, gal/min.

750 gal = pump lo lo level trip setpoint

_____ SSS 2.5.3 The Shift Support Supervisor has reviewed the release permit and has verified that waste monitor tank No. 1 is the tank to be released and that Key Z-19 has not been issued and its position is logged as "Locked Closed".

_____ 2.5.4 Obtain key Z-22 from Shift Support Supervisor.

NOTE:	With R-18 out of service, FNP-0-M-011, OFFSITE DOSE CALCULATION MANUAL Chapter 2 should be referred to for release limitations.

2.5.5 IF LIQ WASTE DISCH R-18 is operable, THEN perform test on LIQ WASTE DISCH R-18 per the following:

_____ 2.5.5.1 IF operable, THEN verify R-18 recorder in operation.

_____ 2.5.5.2 Verify meter responds in the top scale direction on channel R-18 by inserting check source. IF check source is NOT sufficient, THEN have Health Physics source check with a portable source.

_____ 2.5.5.3 IF R-18 recorder is operable, THEN verify source check caused on upscale on R-18 recorder.

_____ 2.5.5.4 Document R-18 source check in FNP-1-STP-1.0, OPERATIONS DAILY AND SHIFT SURVEILLANCE REQUIREMENTS.

_____ 2.5.5.5 Coordinate with the Control Room to perform test on LIQ WASTE DISCH R-18 per section 4.3.

___/___ P.O. 2.5.6 IF R-18 is operable, THEN adjust the monitor setpoint for RE-18 to the value obtained from the discharge permit and the pot setting in Unit 1 Volume III Curve R18.

NOTE:	IF the Service Water dilution flow recorder is out of service, THEN the limitations of FNP-0-M-011, OFFSITE DOSE CALCULATION MANUAL Chapter 2 apply.

_____ 2.5.7 Verify the dilution flow rate is greater than or equal to the minimum, as stated in Part II of the liquid waste release permit. (Utilize the largest dilution flow which can practically be attained.)

_____ 2.5.8 IF the WMT totalizer is operable, THEN verify the "DISPLAY" light is lit, (If not lit, then depress the "DISPLAY" pushbutton), depress the "TOTAL" pushbutton and verify the display reads "TOT GAL" in the message display, then reset the numeric display to "0" by depressing the "TOTAL RESET" pushbutton.

_____ 2.5.9 Record tank level, waste monitor tank totalizer, dilution totalizer, and R-18 setpoint on release permit.

_____ 2.5.10 Open waste monitor tank discharge to the environment 1-LWP-RCV-18 (N1G21V113).

_____ 2.5.11 Unlock and open waste monitor tank 1 discharge valve 1-LWP-V-7446 (Q1G21V111) (Key #Z-22).

_____ 2.5.12 Obtain max release rate from CHM FORM 1700.1, Liquid Waste Release Permit, Part II, and record. __________ gpm.

_____ 2.5.12.1 Subtract 5 gpm from max release rate entered in step 2.5.12.

__________ gpm - 5 gpm = __________
(Step 2.5.12) (MAX RATE)

_____ 2.5.13 <u>IF</u> WMT FI-1085B is operable, <u>THEN</u> slowly open waste monitor tank #1 pump discharge valve 1-LWP-V-7443B (N1G21V108B) until FI-1085B or the totalizer processor "RATE" display indicates the desired release rate, not to exceed the MAX RATE calculated in Step 2.5.12.1 above.

2.5.14 <u>IF</u> WMT FI-1085B is inoperable, <u>THEN</u> perform the following:

_____ 2.5.14.1 Subtract 10 gpm from max release rate entered in step 2.5.12.

__________ gpm - 10 gpm = __________
(step 2.5.12) (MAX RATE)

_____ 2.5.14.2 Determine minimum pump discharge pressure corresponding to the MAX RATE entered in step 2.5.14.1, using Table 1 and record.

__________ psig

_____ 2.5.14.3 Slowly throttle open 1-LWP-V-7443B such that pump discharge pressure does <u>not</u> drop below the pressure recorded in step 2.5.14.2.

_____ 2.5.15 Record start time and release flow rate on release permit.

NOTE: Local indication may be used from N1D11RI018, LIQUID RADWASTE DISCHARGE INDICATOR, located on the liquid waste panel.

____ 2.5.16 IF R-18 is operable, THEN record monitor RE-18 count rate 10 minutes into discharge on release permit.

____ 2.5.17 Verify pump tripped at low level set point or secure pump.

____ 2.5.18 Close waste monitor tank discharge valve to the environment 1-LWP-RCV-18 (N1G21V113).

____ 2.5.19 Record stop time, dilution totalizer, WMT totalizer and tank level on release permit and stop time and date on the WMT flow recorder paper.

__/__ 2.5.20 Close and lock W.M.T. discharge valve 1-LWP-V-7446 (Q1G21V111) (Key #Z-22).

____ 2.5.21 IF locking device removed to verify valve position in previous step, THEN verify locking device properly installed on 1-LWP-V-7446 (Q1G21V111).

____ 2.5.22 Close WMT #1 Pump Discharge Valve 1-LWP-V-7443B (N1G21V108B).

____ 2.5.23 Open WMT inlet valve 1-LWP-V-7413 (Q1G21V090).

____ 2.6 Obtain the pre-release background count rate on R-18 from Liquid Waste Release Permit and record ______________ cpm.

2.7 IF the post-release count rate obtained from the control room is less than or equal to the pre-release count rate, THEN proceed to step 2.8. IF the post-release count rate is greater than the pre-release count rate, THEN back flush radiation monitor R-18 as follows:

____ 2.7.1 Close WMT disch to environment isolation N1G21V281.

____ 2.7.2 Open R-18 flushing drain N1G21V283.

____ 2.7.3 Attach hose between demin water connection N1P11V024 and DW backflush valve 1-LWP-V-7447 (Q1G21V112).

____ 2.7.4 Open DW isolation N1P11V024.

____ 2.7.5 Open DW backflush valve 1-LWP-V-7447 (Q1G21V112).

____ 2.7.6 Throttle open demin water valve N1P11V066.

CAUTION:	**Leaving the R-18 flushing drain N1G21V283 closed, with demin water valve N1P11V066 throttled open could overpressurize the flushing water supply hose.**

____ 2.7.7 While flushing for a minimum of 5 minutes, cycle R-18 flushing drain N1G21V283 open and closed several times to ensure the chamber volume is filled completely.

____ 2.7.8 After 5 minutes have elapsed contact the control room for the R-18 background reading and compare it to the reading recorded in step 2.6.

2.7.8.1 IF the two readings are the same then flushing may be secured.

2.7.8.2 IF the reading is above pre-release levels but has indicated a trend downward on the R-18 recorder, THEN continue flushing for an additional 5 minutes.

2.7.8.3 IF the reading on R-18 recorder has not indicated a downward trend and is above pre-release levels, THEN submit a work request to have R-18 cleaned.

____ 2.7.9 Close DW isolation N1P11V024.

____ 2.7.10 Close demin water valve N1P11V066.

____ 2.7.11 Close DW backflush valve 1-LWP-V-7447 (Q1G21V112).

____ 2.7.12 Close R-18 flushing drain N1G21V283.

____ 2.7.13 Remove hose between demin water connection N1P11V024 and DW backflush valve 1-LWP-V-7447 (Q1G21V112).

____ 2.7.14 Open WMT disch to environment isolation N1G21V281.

NOTE:	**Local indication may be used from N1D11RI018, LIQUID RADWASTE DISCHARGE INDICATOR, located on the liquid waste panel.**

____ 2.8 IF R-18 is operable, THEN observe R-18 count rate for approximately two minutes. Record the highest observed count rate on the release permit and complete all applicable post release information.

___/___ 2.9 <u>IF</u> R-18 is operable, <u>THEN</u> reset R-18 to the value specified on the release permit.
P.O.

_____ 2.10 Return key Z-22 to Shift Support Supervisor.
S.O.

_____
SSS

TABLE 1

#1 WMT PUMP Discharge Pressure
vs. Release Flow Rate

Disch Pressure (psig)	Release Flow Rate (gpm)
106	0
105	5
104	7
103	10
102	12.5
101	14.5
100	18
99	20.5
98	21
97	24.5
96	27
95	29
94	31.5
93	33
92	35.5
91	37.5
90	39
89	41
88	43.5
87	46
86	47.5
85	49
84	> 50

LIQUID WASTE RELEASE PERMIT
FARLEY NUCLEAR PLANT

Unit ________ LWRP# ______________ ______________

PART I: ANALYSIS REQUEST

DISCHARGE SOURCE TANK ______________

Date __/__/__ a. Tank Content ________ (gal) d. Recirc start ________
Time __________ b. Recirc Rate ________ (gpm) e. Sample After ________
Initials ______ c. Recirc Time Minimum (2xa/b) __________ (min)
f. Reviewed by: ______________

SOP 50.1 OTC #______________________

2 SO'S

PART II: PRE-RELEASE CALCULATIONS (CHM)

Unit#__ Tank#___ Sampled @ __________ Recirc Duration ________ (min)
RE-18 Background __________ cpm (must exceed PART 1 c. Limit)

Maximum Permissible Release Rate ______________ GPM

Possible Release Conditions and Dose Projections - SEE ATTACHMENTS

SPECIAL CONDITIONS: Set RE-18 to ≤ ______ cpm prior to release. Reset RE-18 to ______ cpm after release. Use ≥ ______ gpm dilution flow

Composite Stored By ________ ID of Sample, Prerelease & LWRP Verified and Release Approved By ______________
Shift Radiochemist
Date/Time __/__/__

PART III: ACTUAL RELEASE DATA (OPS)

Dilution Flow Rate __________ gpm Monitor Trip Setpoint ____________ cpm
Tank Flow Rate __________ gpm

	Date	Time	Tank Level (gal)	Totalizer	
				Dilut	WMT
Start					
Finish					
Net					

Monitor Reading 10 minutes into Discharge __________ cpm
Post Monitor Reading __________ cpm

Discharge Completed By ______________ Data Reviewed By ______________

SO

Date/Time __/__/__ SSS

PART IV: Release Records Update (CHM)

(already done w/ release by this time,

Actual Release Conditions and Dose Calculations - SEE ATTACHMENTS
Data Updated & Checked By ______________ Date/Time __/__/__

Unit ______ LWRP# ____________________

Detailed Guidance for Review of Liquid Pre-Release Permits

Init's

1. Check spectrum analysis inputs as follows:

_____ A. Correct Detector No.(3 thru 6)
_____ B. Correct Geometry
_____ C. Correct Sample Size (Normally 1000 Mls)
_____ D. Correct Sample Date/Time as compared to Permit
_____ E. Correct Count Time (at least 1000 seconds)
_____ F. Verify Percent (%) dead time is < 10%

_____ 2. Scan Radionuclide Report and compare identified isotopes to those of the source process tank (if available) - each counted on a different detector if both detectors in the applicable unit are operable.

_____ 3. Verify that the Unidentified peak(s) (if any) are not the primary peak(s) of principle gamma emitters as listed in Kev here: 81, 134, 145, 196, 233, 250, 258, 365, 403, 605, 662, 740, 811, 835, 1099, 1115, & 1332.

_____ 4. Verify the following parameters on the computer generated Pre-Release Permit:

_____ A. Correct Monitor Tank number input
_____ B. Same Sample ID No. as WMT sample analysis
_____ C. Estimated Dilution Flow Rate is >= 20,000 GPM or obtain approval from supervision if less.
_____ D. Anticipated (Maximum Possible) Release Rate as determined by specific release conditions.
_____ E. Correct Re-18 Radiation Monitor Background (Part II, CHM Form 1700.1)
_____ F. Release Volume same as permit

_____ 5. Confirm that the Total Gamma Concentration on Part II of the LWRP is equal to the Sum Totals of the Radionuclide Report.

_____ 6. Confirm that the Individual (principle and others) Gamma Emitters listed on the Gamma Spectrum Analysis results are the same as those listed on page 2 of the Liquid Waste Release Permit.

_____ 7. Confirm that Tritium, Gross Alpha, Fe55, Sr89&90 Composite Values are listed and correct on page 2 of the LWRP as compared with page 4 of STP-714.

_____ 8. Check batch doses against quarterly dose criteria.

Unit ________ LWRP#____________________

Detailed Guidance for Review of Liquid Post Release Permits

Init's

_____ 1. Verify that the Pre-Release and Post-Release Sample ID Number agrees.

_____ 2. Verify the following parameters on the computer generated Post-Release Printout and the LWRP part III (Actual Release Data) agree:

_____ A. Start Date/Time.

_____ B. Stop (Finish) Date/Time.

_____ C. Release Volume (Net WMT Tank Level).

_____ D. Dilution Volume (Net Dilution Flow Totalizer * 1E4).

_____ 3. Check batch doses against quarterly dose criteria.

CHAPTER 2
LIQUID EFFLUENTS

2.1 LIMITS OF OPERATION

The following Liquid Effluent Controls implement requirements established by Technical Specifications Section 6.0 {5.0}. Terms printed in all capital letters are defined in Chapter 10.

2.1.1 Liquid Effluent Monitoring Instrumentation Control

In accordance with Technical Specification 6.8.3.e(i) {5.5.4.a}, the radioactive liquid effluent monitoring instrumentation channels shown in Table 2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits specified in Section 2.1.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Section 2.3.

2.1.1.1 Applicability

This limit applies at all times.

2.1.1.2 Actions

With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above control, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperable, or change the setpoint to a conservative value.

With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 2-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report {Radioactive Effluent Release Report} pursuant to Section 7.2 why this inoperablility was not corrected in a timely manner.

This control does not affect shutdown requirements or MODE changes.

2.1.1.3 Surveillance Requirements

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK {source check}, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST {CHANNEL OPERATIONAL TEST (COT)} operations at the frequencies shown in Table 2-2.

2.1.1.4 Basis

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in Section 2.3 to ensure that the alarm/trip will occur prior to exceeding the limits of Section 2.1.2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 2-1. Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	OPERABILITY Requirements[a]	
	Minimum Channels Operable	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE-18)	1	28
b. Steam Generator Blowdown Effluent Line (RE-23B)	1	29
2. Flowrate Measurement Devices		
a. Liquid Radwaste Effluent Line		
1) Waste Monitor Tank No. 1	1	30
2) Waste Monitor Tank No. 2	1	30
b. Discharge Canal Dilution Line (Service Water)	1	30
c. Steam Generator Blowdown Effluent Line	1	30

a. All requirements in this table apply to each unit.

Table 2-1 (contd). Notation for Table 2-1 – ACTION Statements

ACTION 28 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Section 2.1.2.3, and

b. At least two technically qualified members of the Facility Staff independently verify the discharge line valving and

   (1) Verify the manual portion of the computer input for the release rate calculations performed on the computer, or

   (2) Verify the entire release rate calculations if such calculations are performed manually.

Otherwise, suspend release of radioactive effluents via this Pathway.

ACTION 29 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a MINIMUM DETECTABLE CONCENTRATION no greater than 1×10^{-7} μCi/mL:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 μCi/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 μCi/gram DOSE EQUIVALENT I-131.

ACTION 30 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

4.9 Analyses are not performed for C-14 or P-32 since they are not required by the Technical Specifications/ODCM and by recent NRC guidance in NUREG-0472.

4.10 The dilution flow rate for release of liquid radioactive waste should be ≥ 20000 gpm for batch releases (WMT) and ≥ 10000 gpm for continuous releases (SGBD and TBS). Release of liquid radioactive waste with inadequate dilution flow rate must be approved by Chemistry or Environmental Supervision on a case by case basis. If for any reason the dilution flow rate is reduced to less than 20000 gpm during a WMT release or to less than 10000 gpm during SGBD and TBS releases, the release should be immediately terminated and Chemistry or Environmental Supervision notified. When release of liquid radioactive waste is authorized at a dilution flow rate of less than 20000 gpm or 10000 gpm as appropriate a new LWRP should be issued showing the acceptable release parameters.

If dilution water is inadequate for either Unit:

A. Release WMT's from only one Unit at a time and only one WMT at a time, using the dilution water from the available Unit. Include this information in the comments section of the LWRP.

B. For TBS and SGBD updates, use the dilution flow from the available Unit divided by 2, for the period the dilution water was out of service.

5.0 General

5.1 The Effluent Management System (EMS) is a software package which keeps track of effluent emissions by performing complicated dose calculations, and printing reports required by the NRC.

5.2 The two main sources of regulation affecting effluent releases are:

5.2.1 10 CFR Part 20, which governs the instantaneous rate of effluent release through the plant's liquid or gaseous discharge points.

5.2.2 10 CFR Part 50, which specifies the maximum permissible dose (both quarterly and annually) to a hypothetical individual who could be exposed to radiation as the result of effluent releases.

98. G2.4.41 001/3/4/EIP/MEM 4.1/NEW/FA011005/S/CVR/GTO

A Unit 2 Reactor Trip and Safety Injection has occured. The Shift Manager is classifying the event in accordance with FNP-0-EIP-9.0, Emergency Classification and Actions.

Which one of the following is correct with regards to the order of requesting and evaluating Dose Assesement, evaluating Plant Conditions & classifying the event?

A. Request Dose Assessment first, evaluate Plant Conditions next. Classify the event only after obtaining and evaluating both the Dose Assessment and Plant Conditions.

B. Evaluate Plant Conditions first, then request Dose Assessment. Classify the event only after obtaining and evaluating both the Dose Assessment and Plant conditions.

C. Request Dose Assessment first, evaluate Plant Conditions next. Classify the event as soon as either Plant Conditions or Dose Assessment meet criteria for an Emergency Classification.

D. Evaluate Plant Conditions first, then request Dose Assessment. Classify the event as soon as either Plant Conditions or Dose Assessment meet criteria for an Emergency Classification.

A. Incorrect- The second part is wrong. Do not wait for DA and PC to classify.

B. Incorrect- Both first and second part are wrong. DA is called for first.

C. Correct - **Request Dose Assessment first, evaluate Plant Conditions next. Classify the event as soon as either Plant Conditions or Dose Assessment meet criteria for an Emergency Classification.**
4.1 Plant Conditions
<u>While performing the remainder of step 4.1, have the On Shift Dose Analyst commence performing the calculations for dose assessment per step 4.2</u>. Use the following guidelines to determine
4.3.1 Compare the emergency classifications determined from steps 4.1 and 4.2 to determine the highest required emergency classification and declare the emergency. <u>Do not wait for dose assessment results from step 4.2 to classify the event if plant conditions require an initial classification</u> or an upgrade classification. As soon as a criteria for classification has been met, the event should be classified by the Operations Shift Superintendent or ED and an upgrade can be done later if required.

D. Incorrect-The first part is wrong. DA is called for first

G2.4.41 Emergency Procedures / Plan
2.4.41 Knowledge of the emergency action level thresholds and classifications.
(CFR: 43.5 / 45.11)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1. Using plant procedures/references, analyze a set of plant conditions to classify the emergency condition as being a NOUE, Alert, Site Area, or General Emergency (OPS53002C01) (EIP- 9.0).

2. Using plant procedures/references, determine the appropriate actions that are to be performed by the OSS/ED during a NOUE, Alert, Site Area, or General Emergency including the consequences of inadequate actions. (OPS53002C02) (EIP-9.0).

COMMITTED DOSE EQUIVALENT (CDE)
means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

COMMITTED EFFECTIVE DOSE EQUIVALENT (CEDE)
is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs of tissues.

3.5 Protective action recommendation guidance is provided to aid in establishing protective action recommendations. The Emergency Director will exercise his own judgment in recommending protective actions to offsite agencies.

3.6 If steam generator water level falls below the break point during a steam generator tube rupture, off-site dose rate may be significantly higher (up to 10 times) due to volatilization of iodine.

3.7 Initial Notification or upgrade should be made from the Control Room or TSC. It is not necessary to transfer the information to the EOF to make the upgrade notification. The EOF, if staffed, should be informed as soon as possible.

3.8 Communication guidance for making the initial verbal notification is on the Emergency Initial Notification Form, in the appropriate guideline.

3.9 Guidance for when the emergency response facilities should be manned and the level of manning required is included in Table 2. It is recommended that the TSC and the EOF be fully staffed initially at the ALERT level. If the full staff is not required, individuals can be released on a case-by-case basis.

3.10 At the NOUE level or below, it may be desirable to partially staff the TSC in order to relieve the Control Room staff of offsite communications and notifications. FNP-0-EIP-6.0 provides a listing of positions that should be considered for partial TSC activation.

3.11 EIP-6, Figure 3, provides a list of information that should be considered when updating plant staff over the public address system.

4.0 Classify emergency based on the most severe plant conditions OR projected off-site dose/dose rate conditions, WHICHEVER results in the higher emergency classification. Figure 2 provides a flowpath for dose assessment methods and plant conditions criteria.

4.1 Plant Conditions

While performing the remainder of step 4.1, have the On Shift Dose Analyst commence performing the calculations for dose assessment per step 4.2. Use the following guidelines to determine the highest indicated emergency classification based on plant conditions:

Guideline 1, Section I, General Emergency Classification Criteria

Guideline 2, Section I, Site Area Emergency Classification Criteria

Guideline 3, Section I, Alert Criteria

Guideline 4, Section I, NOUE Criteria

4.2 Dose Assessment

CAUTION: DOSE CALCULATIONS FROM EIP-9.1 OR EIP-9.3 ARE NOT TO BE USED TO DECLARE A NOUE OR ALERT SINCE EIP-9.1 AND EIP-9.3 ARE BASED ON EDCM METHODOLOGY, AND NOUE AND ALERT LIMITS ARE BASED ON ODCM METHODOLOGY.

NOTE: **Due to the differences in the met data used for EDCM and ODCM calculations, the following sequence of step 4.2 substeps must be followed. The Top Down approach must be used for dose assessment.**

NOTE: **EDCM dose assessment can only be done from an ERDS terminal or a MIDAS terminal. The only location in the power block where these terminals are available is in the TSC.**

NOTE: **All of the step 4.2 substeps will normally be accomplished by the On Shift Dose Analyst with the exception of steps 4.2.8 and 4.2.11. Steps 4.2.8 and 4.2.11 must be performed by the Shift Supervisor, Operations Shift Superintendent, or Emergency Director.**

4.2.1 Initial evaluation of off-site dose.

The On Shift Dose Analyst when asked to perform dose assessment should initially evaluate effluent monitors (R-14, R-21, R-22, R-29, an R-60 series or an R-15 series) as follows:

4.2.1.1 If there are no effluent radiation monitors that are in alarm or have up-scaled by a factor of 10 or more,
<u>and</u>
there are no other indications of an off-site radioactive release in progress,
<u>then</u>
The On Shift Dose Analyst should report to the ED that there is no indication of a radioactive release based on effluent monitors and NO additional dose assessment per step 4.2 is required. Continue to perform this assessment periodically not to exceed 30 minutes until the requirement is terminated by the ED.

4.2.1.2 If any effluent radiation monitor is in alarm or has up-scaled by a factor of 10 or more,
<u>or</u>
there are other indications of an off-site radioactive release in progress,
<u>then</u>
For initial dose assessment from the TSC, proceed to step 4.2.4.

4.2.2 For dose assessment from the EOF or long term dose assessment from the TSC, go to EIP-9.3, PERSONNEL COMPUTER-AUTOMATED DOSE ASSESSMENT and perform dose assessment using the MIDAS program. Return to step 4.2.8 for evaluation of dose information and continue with step 4.2.9.

4.2.3 If the MIDAS program is inoperable, then for dose assessment from the EOF or from the TSC, go to EIP-9.1, AUTOMATED DOSE ASSESSMENT and perform dose assessment using the ARDA program to obtain dose information. Return to step 4.2.8 for evaluation of dose information and continue with step 4.2.9.

4.2.4 If the ARDA System is operable and has been automatically activated, then go to EIP-9.1, AUTOMATED DOSE ASSESSMENT and perform dose assessment using the ARDA program to obtain dose information. Return to step 4.2.8 for evaluation of dose information and continue with step 4.2.9.

4.2.5 If the ARDA System per EIP 9.1 is operable, has not automatically activated, and one of the following rad monitors has alarmed:

- R-29
- R-15C
- R-60 A, B, C, or D
- R-14
- R-21
- R-22

Then go to EIP-9.1, AUTOMATED DOSE ASSESSMENT, manually start ARDA and perform dose assessment using the ARDA program to obtain dose information. Return to step 4.2.8 for evaluation of dose information and continue with step 4.2.9.

4.2.6 If the ARDA system per EIP 9.1, AUTOMATED DOSE ASSESSMENT is NOT operable, then go to EIP-9.3, PERSONAL COMPUTER-AUTOMATED DOSE ASSESSMENT and perform dose assessment using the MIDAS program. Return to step 4.2.8 for evaluation of dose information and continue with step 4.2.9.

4.2.7 If the ARDA system per EIP 9.1 AUTOMATED DOSE ASSESSMENT is operable, has not automatically activated, and none of the alarms listed in step 4.2.5 have alarmed then go to EIP-9.5, EMERGENCY CLASSIFICATION BASED ON ODCM to perform dose assessment. Return to step 4.2.11 for evaluation of doserate information.

NOTE: Evaluating the dose assessment information in Step 4.2.8 must be performed by the Shift Supervisor, Operations Shift Superintendent, or Emergency Director in the Control Room or TSC, the Dose Assessment Supervisor or EOF Manager in the EOF.

4.2.8 Using the dose information obtained from EIP-9.1 or EIP-9.3, determine the highest indicated emergency classification from the "High Effluent" criteria in Guideline 1, Section I, or Guideline 2, Section I.

NOTE: If a General Emergency or site area emergency is indicated in the following step, the Emergency Director should consider directing long term dose assessment be performed from the TSC per step 4.2.2.

4.2.9 If a General Emergency or Site Area Emergency was indicated from step 4.2.8, then go to step 4.3.

4.2.10 If a General Emergency or Site Area Emergency was not indicated in step 4.2.8, then go to EIP-9.5, EMERGENCY CLASSIFICATION BASED ON ODCM. Return to step 4.2.11 for evaluation of doserate information.

NOTE:	**Evaluating the dose assessment information in Step 4.2.11 must be performed by the Shift Supervisor, Operations Shift Superintendent, or Emergency Director in the Control Room or TSC, and by the Dose Assessment Supervisor or EOF Manager in the EOF.**

4.2.11 Using the dose rate information obtained from EIP-9.5, determine the highest indicated emergency classification from the "High Effluent" criteria in Guideline 3, Section I, and Guideline 4, Section I.

4.3 Determine the correct emergency classification, declare the emergency at the time the classification was verified in the Guideline, determine PARs and make notifications.

4.3.1 Compare the emergency classifications determined from steps 4.1 and 4.2 to determine the highest required emergency classification and declare the emergency. **Do not wait for dose assessment results from step 4.2 to classify the event if plant conditions require an initial classification or an upgrade classification.** As soon as a criteria for classification has been met, the event should be classified by the Operations Shift Superintendent or ED and an upgrade can be done later if required.

4.3.2 Using section L of the guideline for the highest emergency classification determined in step 4.3.1, determine the required protective action recommendations.

4.3.3 Complete the initial notification form at the back of the guideline and perform required notifications.

5.0 Perform actions and initial notification to offsite authorities upon initial entry or upgrade into a classification using the applicable guideline:

Guideline 1, Section II - General Emergency

Guideline 2, Section II - Site Area Emergency

Guideline 3, Section II - Alert

Guideline 4, Section II - Notification of Unusual Event

6.0 Continue reassessment of emergency classification per step 4.0 or 7.0, as appropriate, and transmit follow-up message/periodic update message as follows:

6.1 Transmit Follow-up Messages:

6.1.1 Transmit a follow up message as soon as possible following an initial or upgrade verbal notification but no longer than 30 minutes after the verbal notification has been transmitted over the ENN. Refer to step 6.2 for time limits for other follow-up messages.

6.1.2 Use, if desired, Attachment 1, for guidance in completing and transmitting the "Emergency Message" for Follow Up/Periodic Update (Figure 6).

6.1.3 When performing dose assessment, transcribe dose information from the form being printed on a blank Figure 6 or use the form being printed. Fill in the remaining information. Transmit follow up message by telecopy.

NOTE: EFFORTS WILL BE MADE TO TRANSMIT FOLLOW-UP REPORTS EVERY HALF HOUR.

6.2 Transmit subsequent "Follow Up Message/Periodic Update Message" reports per step 6.1.

6.2.1 At a minimum of once per hour. The hourly requirement may be waived while in a NOUE declaration, if this is agreed to by the state and local agencies.

6.2.2 Following a significant change in dose rate that does not require a change in emergency classification.

6.2.3 Following a significant change in plant conditions that does not require a change in emergency classification.

7.0 Downgrade or closeout an emergency classification after determining, through the use of the guidelines, that the current emergency classification is no longer required. FNP-0-EIP-28.0 will be used to downgrade or closeout an emergency class

ATTACHMENT 1

EMERGENCY NOTIFICATION FORM GUIDANCE

1.0 Emergency Notification Form Guidance is provided below. The guidance is provided for filling out the Emergency Notification Form (FIGURE 6). This attachment can be used as reference guidance when filling out the form or for training.

1.1 **A VERBAL NOTIFICATION** of a declared emergency or an upgrade to a higher level declared emergency shall be performed in accordance with Guidelines 1 through 4 using the appropriate Verbal Notification Form.

1.2 **A FOLLOW-UP MESSAGE** will be used to make periodic plant status updates using Figure 6. This attachment is a REFERENCE to use as necessary for completing Figure 6.

NOTE: THE BOXES CAN BE INDICATED BY A CHECK, AN "X", OR BY FILLING OR BLACKENING IN.

Line 1
- Check A or B.
- Except for actual emergencies A should be checked in training and for drills/exercises.
- Check D.
- D should be checked for Follow-up Messages described in step 1.2 above. Note the single asterisk associated with box D for Follow-up Messages.
- The Follow-up message numbers are sequential, starting with 001.

Line 2
- Specify Unit 1, 2, or both. If both is entered consider describing the least affected unit reactor status in line 7.
- Reported by is the individual who filled out the form.

Line 3
- Fill in the time/date of this message just prior to transmitting.
- Fill in an extension number that will be staffed so that the state and county agencies have a place to talk to someone that can help them. A Direct inward dial number may be entered. In the TSC, EOF or Control Room (334) 814-4662 and (334) 814-4663are dedicated for call in by the states.
- Select TSC, EOF or fill in a separate fax number that will be staffed, so that the state and county agencies have a location where they can fax information if necessary

Line 4
- This line is not used at FNP. It remains on the form to maintain conformity with other state forms.

ATTACHMENT 1

Line 5

- Check A, B, C or D (only one) as appropriate. Check no boxes if this is a termination.

Line 6

- Check A or B. Time is for the declaration checked in line 5 or the time of termination.. Termination should only be done after conversation with off-site authority per FNP-0-EIP-28.0. If this is a termination, go to line 16.

Line 7

- Provide a brief, concise summary of plant conditions that requires the classification and other pertinent information.
- Enter the classification criteria used in words; the criteria code may also be included.
- If more space is needed, check box E and use the continuation sheet.
- If any of the items in block A, B, C or D apply, check them as well.

Line 8

- Check A, B or C.
- A should be checked if mitigation efforts appear successful , progressing toward termination/recovery.
- B should be checked if escalation to a higher classification is unlikely based on current conditions.
- C should be checked if escalation to a higher emergency classification or PAR change is likely.
- Check D if FMTs have been dispatched off-site
- Check E if personnel not needed to support recovery efforts have been released from the site. This provides the state and county information as to who to expect to see in the 10 mile EPZ

Line 9

- Check A or B to provide information about the affected unit.
- If both units are affected provide the least affected unit information in line 7 and the most affected unit information here
- If A is checked complete the time that the reactor was shutdown
- If B is checked, provide the current reactor power from N41, 42, 43 or 44.

ATTACHMENT 1

Line 10

- Check A, B, C or D (only one).
- Check A if the criteria for a (B) Potential Release, (C) Is Occurring or (D) Has Occurred has not been met
- If A is checked, go to line 14 to provide meteorological data. Off-site agencies still want line 14 information in order to know what zones might be affected. No information should be filled in for lines 11, 12 or 13.
- Check B for potential failures of the final barrier to the release of fission fragments. Examples of potential releases are the RCS and clad barriers that are known to be failed and containment pressure is above design value, or the clad and containment barriers are failed and a pressurizer safety valve is stuck with PRT pressure rising. For potential releases, do not put estimated or potential values for the release information in lines 11, 12, or 13, because there is no release in progress.
- If B is checked, go to line 14 to provide meteorological data. Off-site agencies still want line 14 information in order to know what zones might be affected. No information should be filled in for lines 11, 12 or 13.
- Check box C If an emergency release is occurring. An emergency release should be considered to exist if an abnormal increase in radiological release rate by a factor of 10 over and above normal operating levels has occurred. Normal operating levels are average release levels for steady-state operations (may be obtained from ERDS, RMDA, control room recording, or plant computer).
- If A Site Area Emergency or General Emergency has been declared and a release is in progress, complete the information in lines 11, 12 and 13 based on the values obtained from an EDCM calculation (MIDAS or ARDA)
- If a NOUE or an ALERT has been or can be declared based on the release in progress then provide the release information in line 7 as directed by step D of Guideline 3 or 4. Do not provide any information in lines 11, 12 or 13 except for checking 12D to indicate that the release is above Technical Specification limits
- Check Box D if a release that was large enough to require an emergency classification of NOUE or above to be declared had occurred but is now stopped.
- If EDCM dose assessment information has already been provided in lines 11, 12 and 13 in previous messages continue to provide this information.
- If a new estimate of projected off site dose is not available then do not provide data in lines 11 through 13, go directly to line 14.

Line 11

- For actual releases only, normally check A
- Mixed mode calculations are normally not done at FNP
- Fill in actual start/stop times for C, D, E and F as appropriate. It is not necessary to check the boxes when data is being entered into the blanks.

ATTACHMENT 1

Line 12

- The ARDA and MIDAS programs provide release rate in μCuries/Sec. Check box A.
- For all emergency releases check C or D to indicate if release is above or below Technical Specification limits. Any release that can be used to declare a NOUE or above is above Technical Specification Limits and if no emergency can be declared it is below Technical Specification limits
- The data for completing blanks E, F, G and H on this line is available from the MIDAS or the ARDA printout.
- It not necessary to check the boxes when data is being entered into the blanks.

Line 13

- For all new dose assessment calculations that are to be reported, then check new and fill in Blanks C through L. Note that line 14 meteorological data must be the same data that was used in performing the new dose assessment calculation.
- If there has been no new dose assessment calculation performed since the last message, then check box B and leave all other line 13 blanks empty.
- Box B can also be checked to if the intent of the message is to report rapid changes in emergency status, plant conditions, meteorological conditions or changes in PARS without performing a new dose projection.
- In blank C fill in the length of time in hours that the release is expected to continue from the current time. Four hours should always be assumed if the actual time is unknown.
- The data for completing blanks D through K on this line is available from the MIDAS or the ARDA printout.
- It not necessary to check the boxes when data is being entered into the blanks.
- For box L Obtain the accident type number from the MIDAS Program printout. It is displayed in the same location on the MIDAS printout. If dose assessment is being done using ARDA place NA in this block
- For ODCM calculations (for NOUE or Alert), do not enter dose or dose rate information here. Place the dose rate information in the comments at line 7, along with the notes that are specified in the NOUE and Alert Guidelines if declaration is based on the ODCM.

Line 14

- Fill in the meteorological data required in A through D. Note that line 14 meteorological data must be the same data that was used in performing the dose assessment calculation information provided on line 12 and 13.
- When possible use 15 minute average data, available from the NR ERDS MIDAS page

Line 15

- Refer to Step L of the appropriate FNP-0-EIP-9.0 Guideline to determine appropriate Protective action Recommendations (PARs).
- If there are no protective action recommendations, check box A and proceed to step 16.
- Specific protective actions must be made for a general emergency.

ATTACHMENT 1

Line 16

- "Approved by" should be the Emergency Director or EOF Manager, with the current date and time.

99. G2.4.9 001/3/4/SFP/C/A 3.9/NEW/FA011005/S/CVR/GTO

Spent Fuel Pool (SFP) cooling has been lost due to the loss of both Trains of CCW. EH1, SFP TEMP HI, alarm has just come in.

Which one of the following is the correct mitigation strategy in order of preference per AOP-36.0, Loss of Spent Fuel Pool Cooling?
(Assume the first method slows but does **NOT** stop the temperature rise of the SFP water.)

A. Feed and Bleed using the Recycle Holdup Tanks (RHTs). Evaporative loss while maintaining SFP level using the Demineralized water system.

B✓ Feed and Bleed using the Refueling Water Storage Tank (RWST). Evaporative loss while maintaining SFP level using the Demineralized water system.

C. Feed and Bleed using the RHTs. Evaporative loss while maintaining SFP level using the RWST.

D. Feed and Bleed using the RWST. Evaporative loss while maintaining SFP level using the RWST.

A. Incorrect. First part wrong, second part right. The RHT can be aligned to makeup to the charging pump suction, but no procedural guidance exists to make up to the SFP in this condition.

B. Correct. **Feed and Bleed using the Refueling Water Storage Tank (RWST). Evaporative loss while maintaining SFP level using the Demineralized water system.**

Note prior to 10 clarifies the intent of step 10 & 11. The RWST is used to feed and bleed using the cooler water of the RWST which is at the correct boron concentration to supply the SFP. Warmer water from the SFP is bled back to the RWST and mixed with the cooler water of the tank contents.

C. Incorrect. Both parts wrong. Under certain conditions the RHT would be a makeup source to the RCS via the charging pump suctions, but no procedural guidance exists for use of the RHTs to make up to the SFP in this condition. Evaporative loss in this condition per AOP-36.0 would be compensated for in the same way that it is during normal evaporation per SOP-54.0: Demineralized or Reactor Makeup water. The reason is that the boron stays in the pool when the water evaporates out. Making up for evaporation with RWST water would eventually cause the SFP boron concentration to be higher than allowed.

D. Incorrect. First part right, second part wrong. Evaporative loss in this condition per AOP-36.0 would be compensated for in the same way that it is per SOP-54.0 during normal evaporation: Demineralized or Reactor Makeup water. The reason is that the boron stays in the pool when the water evaporates out. Making up for evaporation with RWST water would eventually cause the SFP boron concentration to be higher than allowed.

G2.4.9 Emergency Procedures / Plan
2.4.9 Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.(CFR: 41.10 / 43.5 / 45.13)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. Evaluate abnormal plant or equipment conditions associated with the Spent Fuel Pool Cooling and Purification and Refueling Water Storage Tank Purification Systems and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52108L02).

A. **Purpose**

This procedure provides actions for response to a loss of spent fuel pool cooling.

This procedure is applicable at all times.

B. **Symptoms or Entry Conditions**

I. This procedure is entered when a loss of spent fuel cooling is indicated by any of the following:

a. Loss of on service spent fuel pool cooling train

b. Loss of CCW supply to spent fuel pool heat exchangers

NOTE: Entry into AOP may be delayed until actual SFP temperature is verified to be ≥ 130°F.

---

c. Actuation of SFP TEMP HI annunciator EH1 (Variable, not to exceed 130°F [normally set at 115°F])

Step	Action/Expected Response		Response NOT Obtained
___ 1	Check on service spent fuel pool cooling train - OPERATING.	1	Place standby spent fuel pool cooling train in operation using FNP-1-SOP-54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.
___ 2	Check CCW to on service spent fuel pool cooling train - AVAILABLE.	2	Place standby spent fuel pool cooling train in operation using FNP-1-SOP-54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.
___ 3	<u>WHEN</u> spent fuel pool cooling restored, <u>THEN</u> go to procedure and step in effect.		

NOTE:
- Step 4 applies to fuel assemblies in the spent fuel room.
- Fuel should be verified to be stored properly in accordance with Technical Specification 3.7.15.

Step	Action/Expected Response		Response NOT Obtained
___ 4	Place any fuel assembly in transit in spent fuel rack.		
___ 5	Secure fuel handling operations in spent fuel room.		
___ 6	Verify fuel transfer cart - IN SPENT FUEL PIT LOCATION.		
___ 7	Verify fuel transfer tube gate valve - CLOSED. (155 ft, AUX BLDG spent fuel room)		
___ 8	Secure spent fuel pool skimmer loop using FNP-1-SOP-54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.		
___ 9	Place fuel handling area HVAC in service using FNP-1-SOP-58.0, AUXILIARY BUILDING HVAC SYSTEM.		

___Page Completed

Step	Action/Expected Response		Response NOT Obtained

NOTE: The intent of steps 10 and 11 is to provide spent fuel pool cooling by pumping warm water to the RWST and refilling with cool water from the RWST.

Step	Action/Expected Response		Response NOT Obtained
___10	**Raise spent fuel pool level to 153'10" using FNP-1-SOP 54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.**	10	Proceed to step 13.
___11	**Reduce spent fuel pool level to 151'6" using FNP-1-SOP-54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.**	11	Proceed to step 13.
___12	**Check spent fuel pool temperature N1G31TIS501 - GREATER THAN 150°F. (155 ft, AUX BLDG spent fuel room)**	12	Return to step 1.
___13	**Continue efforts to restore spent fuel pool cooling.**		

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE:

- Without cooling, evaporative loss from the spent fuel pool will rise. The intent of step 14 is to make up to the spent fuel pool from the demineralized water system or the reactor makeup water system to replace this loss.
- The 139 ft level in the spent fuel pool is approximately one third of the height up the weir gate.

Step	Action/Expected Response	Response NOT Obtained
__14	**Maintain spent fuel pool level 153'4" to 153'10" using FNP-1-SOP-54.0, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM.**	14 IF spent fuel pool level less than 139 ft, THEN perform the following: 14.1 Align any available water source for makeup to spent fuel pool. 14.2 Have Chemistry sample the SFP water to ensure a boron concentration of 2000 ppm or greater.

-END-

100. WE01G2.1.20 001/1/2/ESP-1.1/MEM 4.2/NEW/FA011005/S/CVR

An inadvertent Safety injection has occurred on Unit 1. The team was expecting to terminate SI in ESP-1.1, SI Termination, but conditions have changed, and the team is not sure what event has occurred. ESP-0.0, Rediagnosis, has been entered to evaluate which procedure to transition to next.

The following conditions exist:

- R-2 and 7, CTMT rad monitors, are in alarm.
- A controlled cooldown has not been commenced.
- SG narrow range water levels are:
  1A SG - 85%   1B SG - 30%   1C SG - 32%
- All MSIVs are open.
- All SG pressures are dropping rapidly.

Which one of the following shows the correct procedural actions of ESP-0.0 and the procedural transition for the above conditions?

A. • Check containment radiation normal.
 • Enter EEP-1, Loss of Reactor or Secondary Coolant.

B. • Check any SG level rising in an uncontrolled manner.
 • Enter EEP-3, Steam Generator Tube Rupture.

C. • Check any SG pressure stable or rising.
 • Check for a controlled cooldown in progress.
 • Enter EEP-2, Faulted Steam Generator Isolation.

D. • Check any SG pressure stable or rising.
 • Close the MSIVs.
 • Check any SG level rising in an uncontrolled manner.
 • Enter EEP-3, Steam Generator Tube Rupture.

A. Incorrect. ESP-0 never looks at CTMT radiation. EEP-0 does look at this in diagnostics. Also ESP-0 looks at SG pressures first and would transition to that procedure.

B. Incorrect. This will be done at step 3 and would be correct if all SG pressure were not falling rapidly.

C. Correct.

- **Check any SG pressure stable or rising.**
- **Check for a controlled cooldown in progress.**
- **Enter EEP-2, Faulted Steam Generator Isolation.**

ESP-0.0 looks at the faulted SGs first. If no controlled cooldown is in progress, then the operator is directed to isolate MS lines in EEP-2.

D. Incorrect. The first step is correct, however, the MSIVs are closed in EEP-2.0 and the SGs are isolated in that procedure. While it is obvious that a SGTR is also going on here, it is the last step of ESP-0.

WE01G2.1.20 Rediagnosis Ability to execute procedure steps.
(CFR: 41.10 / 43.5 / 45.12)

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

8. Evaluate plant conditions and determine if transition to another section of ESP-1.1 or to another procedure is required. (OPS52531E08)

FARLEY NUCLEAR PLANT

EVENT SPECIFIC PROCEDURE

FNP-1-ESP-0.0

REDIAGNOSIS

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	**ALL**
Reference Use	
Information Use	

SAFETY RELATED

Approved:

D. J. WHITE
Operations Manager

Date Issued: 3-27-01

A. Purpose

This procedure provides a mechanism to allow the operator to determine or confirm the most appropriate post accident recovery procedure with SI in progress or required.

B. Symptoms or Entry Conditions

I. This procedure is entered based on operator judgement.

Step	Action/Expected Response	Response NOT Obtained

NOTE: This procedure should only be used if SI in progress or required.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Check if any SG is not faulted.**	
1.1	Check pressures in all SGs - ANY STABLE OR RISING.	1.1 IF a controlled cooldown is in progress, THEN proceed to Step 2. IF NOT, THEN the following applies. • IF main steam lines have NOT been isolated, THEN go to FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION. OR • IF main steam lines are isolated, THEN go to FNP-1-ECP-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.
___ 2	**Check if all SGs are not faulted.**	
2.1	Check no SG pressure - FALLING IN AN UNCONTROLLED MANNER OR LESS THAN 50 psig.	2.1 IF affected SG(s) NOT previously isolated, THEN go to FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION.

___ Page Completed

FNP-1-ESP-0.0	REDIAGNOSIS	Revision 12

Step	Action/Expected Response	Response NOT Obtained

NOTE: The LOCA ECPs are FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION and FNP-1-ECP-1.2, LOCA OUTSIDE CONTAINMENT.

___ 3 **Check any SG ruptured.**

- Check secondary radiation indication - HIGH

[] R-15 SJAE EXH
[] R-19 SGBD SAMPLE
[] R-23A SGBD HX OUTLET
[] R-23B SGBD TO DILUTION
[] R-15B TURB BLDG VNTL (BOP)
[] R-15C TURB BLDG VNTL (BOP)
[] R-60A MS ATMOS REL (BOP)
[] R-60B MS ATMOS REL (BOP)
[] R-60C MS ATMOS REL (BOP)
[] R-60D TDAFWP EXH (BOP)

<u>OR</u>

- **Check any SG level - RISING IN AN UNCONTROLLED MANNER.**

3 <u>IF</u> any LOCA ECP in effect <u>AND</u> conditions have <u>NOT</u> changed, <u>THEN</u> go to procedure in effect,
<u>IF NOT</u>, go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

NOTE: The SGTR ECPs are FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED, FNP-1-ECP-3.2, SGTR WITH LOSS OF REACTOR COOLANT SATURATED RECOVERY DESIRED and FNP-1-ECP-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.

___ 4 **<u>IF</u> any SGTR ECP in effect <u>AND</u> conditions have <u>NOT</u> changed, <u>THEN</u> go to procedure in effect.**

4 **Go to FNP-1-EEP-3, STEAM GENERATOR TUBE RUPTURE.**

-END-