QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

1. 2009A NRC RO 001

Given the following plant conditions:
- A reactor trip has just occurred from full power due to a loss of Offsite Power
- The crew has entered EPP-004, Reactor Trip Response

The following plant conditions currently exist:
- All SG levels are 21% and lowering
- Total AFW flow to the Steam Generators is 211 KPPH
- Loop Low Tavg Bistable lights are lit on TSLB 3
- Loop Low-Low Tavg Bistable lights are lit on TSLB 3
- Group 1 Condenser Steam Dumps red AND green lights are lit on SLB 1
- The BOP is directed to control temperature in accordance with EPP-004 Table 1

Which ONE of the following describes the action required in accordance with Table 1?

A.✓ Close all Main Steam Isolation Valves

B. Place Steam Dumps in the Steam Pressure Mode

C. Decrease AFW flow to the Steam Generators to suspend the RCS Cooldown

D. Increase AFW flow to the Steam Generators to raise Steam Generator Water Levels

Plausibility and Answer Analysis

A Correct. With Loop Low-Low Tavg Bistable lights illuminated on TSLB 3 (P-12), the Steam Dumps should be closed and are not. EPP-004 states if the cooldown continues, close the MSIV and bypass valves.

B Incorrect. EPP-004 has guidance to place the Steam Dumps in the Steam Pressure Mode in several locations. The candidate may believe this will assist with present plant conditions, but P-12 affects Steam Dumps in both the Tavg and Steam Pressure Modes.

C Incorrect. Because RCS Temperature is less than 557°F, the candidate may believe decreasing AFW flow to be correct because EPP-004 has guidance to control RCS temperature at 555 to 559°F but greater than 210 KPPH is required for heatsink.

D Incorrect. Steam Generator Water Levels are 21% and decreasing so the candidate may believe increasing AFW flow to be correct because EPP-004 has guidance to maintain SGWL 25 to 50%. However, with RCS Temperature less than 557°F, AFW flow should be greater than 210 KPPH but minimized to prevent further cooldown.

KA Statement - Reactor Trip/Stabilization/Recovery - Knowledge of the interrelations between (EMERGENCY PLANT EVOLUTION) and the following: (CFR: 41.7 / 45.7 / 45.8): Reactor trip status panel

Importance Rating:	3.5 3.6
Technical Reference:	EPP-004 Rev. 18, Page 6
References to be provided:	None
Learning Objective:	MCB, Obj. 2a
Question Origin:	NEW
Comments:	K/A match because individual must analyze conditions from TSLB and plant to come to correct answer for stabilization.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	EPP-004
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	007EK2.03	K/A 2:	

Instructions	Response Not Obtained
3. Check RCS Temperature:	
a. Check SG blowdown isolation valves - SHUT	a. Shut SG blowdown FCVs: 1BD-18 (FCV-8405A) 1BD-37 (FCV-8405B) 1BD-56 (FCV-8405C)

SG	(MLB-1A-SA)	(MLB-1B-SB)
A	1BD-11	1BD-1
B	1BD-30	1BD-20
C	1BD-49	1BD-39

b. Stabilize AND maintain temperature between 555°F AND 559°F using Table 1.

TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP

o Guidance is applicable until another procedure directs otherwise.

o IF no RCPs running, THEN use wide range cold leg temperature.

	RCS TEMPERATURE TREND		
	LESS THAN 557°F AND DECREASING	GREATER THAN 557°F AND INCREASING	STABLE AT OR TRENDING TO 557°F
OPERATOR ACTION	o Stop dumping steam o Control feed flow o Maintain total feed flow greater than 210 KPPH until level greater than 25% at least one intact SG o IF cooldown continues, THEN, shut MSIVs AND BYPASS valves	o IF condenser available THEN transfer steam dump to STEAM PRESSURE mode using OP-126, Section 5.3 AND dump steam to condenser - OR - o Dump steam using intact SG PORVs o Control feed flow to maintain SG levels	o Control feed flow and steam dump to establish and maintain RCS temperature between 555°F AND 559°F

2. 2009A NRC RO 002

Given the following plant conditions:
- The plant was operating at 100% when a small break LOCA occurred
- The crew tripped the Reactor and performed a manual SI
- The crew is performing actions of PATH-1

The RO reports the following indications:
- FI-943, SI Flow: 0 gpm
- 'A' CSIP ammeter: 49 amps
- 'B' CSIP ammeter: 50 amps
- RCS pressure: 1390 psig
- SI Valves are properly aligned per PATH-1 Guide Attachment 1, Emergency Alignment

Which ONE of the following describes the required operator actions for these conditions?

	Trip ALL RCPs	Verify Alternate Miniflow Valves are:
A.✓	Immediately	SHUT
B.	Immediately	OPEN
C.	When FI-943 exceeds 200 gpm	SHUT
D.	When FI-943 exceeds 200 gpm	OPEN

Plausibility and Answer Analysis

RCP TRIP CRITERIA - IF both of the following occur, THEN stop all RCPs: SI flow - GREATER THAN 200 GPM, RCS pressure - LESS THAN 1400 PSIG.

ALTERNATE MINIFLOW OPEN/SHUT CRITERIA - IF RCS pressure decreases to less than 1800 PSIG, THEN verify alternate miniflow isolation OR miniflow block valves - SHUT. IF RCS pressure increases to greater than 2200 PSIG, THEN verify alternate miniflow isolation AND miniflow block valves - OPEN

A Correct. PATH-1 Foldout A items:
EOP User's Guide section 6.29 explains that the SI Flow Meter is not safety related and in its absence, CSIP performance and valve alignment should be used.

B Incorrect. Tripping ALL RCP's is correct based on PATH-1 foldout A and EOP User's Guide but the CSIP alternate miniflow isolation valves would be verified SHUT, not OPEN.

C Incorrect. Plausible if candidate knows the Foldout but is unaware that the SI Flow Meter is not safety related and in its absence, CSIP performance and valve alignment should be used. The second part of the response is correct.

D Incorrect. CSIP performance and valve alignment should be used to determine RCP Trip Criteria is met. Also, the CSIP alternate miniflow isolation valves would be verified SHUT not OPEN.

KA Statement - Small Break LOCA - Ability to (a) predict the impacts of the following on the (SYSTEM) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:(CFR: 41.5 / 43.5 / 45.3 / 45.13): RCS parameters

Importance Rating:	3.3 3.4
Technical Reference:	PATH-1 Foldout A Rev. 23, page 8 EOP-Users Guide Rev. 26, page 53
References to be provided:	None
Learning Objective:	EOP-LP-3.1, Obj. 3c
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	2	Reference:	PATH-1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	009EA2.15	K/A 2:	

FOLDOUT A

- RCP TRIP CRITERIA

  IF both of the following occur, THEN stop all RCPs:

  - SI flow - GREATER THAN 200 GPM
  - RCS pressure - LESS THAN 1400 PSIG

- AFW SUPPLY SWITCHOVER CRITERIA

  IF CST level decreases to less than 10%, THEN switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

- RHR RESTART CRITERIA

  IF RCS pressure decreases to less than 230 PSIG in an uncontrolled manner, THEN restart RHR pumps to supply water to the RCS.

- ALTERNATE MINIFLOW OPEN/SHUT CRITERIA

  - IF RCS pressure decreases to less than 1800 PSIG, THEN verify alternate miniflow isolation OR miniflow block valves - SHUT
  - IF RCS pressure increases to greater than 2200 PSIG, THEN verify alternate miniflow isolation AND miniflow block valves - OPEN

6.29 Loss of SI Flow Indication

Numerous EOPs check for the presence of high head SI flow which is indicated on FI-943. It is possible; however, this instrument may fail. Lack of flow indication on FI-943 should not be interpreted as a failure of the high head SI piping and should not be used as the basis to establish an alternate injection flow path. In the absence of flow indication, the operator should rely on the valve position indication and CSIP performance to determine the status of SI flow. If a flow path through the BIT can be verified, the operator should assume the high head SI system is intact. The same approach should be taken when determining the status of SI flow for the RCP trip criteria.

Establishing an alternate flow path in this situation is not appropriate for the following reasons:

(1) Failure of the high head SI piping is not considered plausible by the EOPs and is not a significant contributor to the core damage frequency (CDF) in the SHNPP Individual Plant Evaluation (IPE).

(2) FI-943 is not safety related. It is a non-conservative action to realign the SI system in a configuration different from that assumed in the FSAR based on a non-safety related instrument that potentially has failed.

(3) In the unlikely case of a failure of the high head SI piping, absence of flow on FI-943 would indicate a failure upstream of the associated flow transmitter. Shutting the BIT outlet valves would not isolate the failure. Opening an alternate injection valve with the failure unisolated would force the CSIPs to discharge through two flow paths simultaneously, potentially causing the CSIPs to experience runout.

Appropriate diagnostics for failures of the SI system are included in the EOPs with associated contingency actions. In the unlikely event the BIT valves do not open on SI, the CSIPs are not expected to function for any significant period of time. The normal miniflow and charging isolation valves will be shut from the SI signal. Also, the alternate miniflow isolation valves will likely be shut in response to decreasing RCS pressure. The only flowpath remaining will be seal injection. In the event a CSIP does continue to operate, guidance for establishing an alternate injection path is given in PATH-1.

In the event no high head SI flow can be established, the cooldown and depressurization strategy employed by the EOPs should be successful in maintaining adequate core cooling. If adequate core cooling can not be maintained, FRP-C.1 and FRP-C.2 provide guidance to preclude core damage. This guidance includes aligning alternate high head injection flow paths. In this case it may be necessary to restore the "swing" CSIP to service to provide high head SI flow (Reference 2.2.2.4).

3. 2009A NRC RO 003

Given the following plant conditions:
- A large break LOCA has occurred
- ALB-004-2-4, Refueling Water Storage Tank 2/4 Low Low Level, annunciator was received several minutes ago
- RWST level indicates 20%
- All required automatic actions have occurred as designed

Which ONE of the following describes the action(s) that must be taken to properly align the 'A' RHR pump for continued core cooling?

A.✓ SHUT 1SI-322, RWST to RHR pump suction valve

B. OPEN 1SI-340, Low Head SI to Cold Leg injection valve

C. SHUT 1RH-25 and 1RH-63, RHR Discharge to CSIP suction valves

D. OPEN 1SI-300 and 1SI-310, CNMT Sump to RHR pump suction valves

Plausibility and Answer Analysis

Continued Core Cooling will be aligned IAW EPP-010.

A *Correct. 1SI-322 must be shut after verifying 1SI-300 and 1SI-310 have automatically opened.*

B *Incorrect. 1SI-340 must be operated in EPP-010 but the direction is to SHUT 1SI-340.*

C *Incorrect. These valves would be opened by EPP-010 to establish the CSIP Recirculation alignment. Plausible if candidate believes this flow path is secured for long term core cooling.*

D *Incorrect. 1SI-300 and 1SI-310 must be verified open to align 'A' RHR but these valves will open automatically on the RWST Low Low Level.*

KA Statement - Large Break LOCA - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

Importance Rating:	4.2 4.4
Technical Reference:	EPP-010 Rev. 24, pages 4 & 6
	APP-ALB-004 Rev. 16, page 9
References to be provided:	None
Learning Objective:	EOP-LP-3.3, Obj. 5b
Question Origin:	BANK LOR NRC Bank RHRS-2.0-R11 001
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	3	Reference:	EPP-010
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	011EG2.2.44	K/A 2:	

TRANSFER TO COLD LEG RECIRCULATION

Instructions	Response Not Obtained

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CAUTION

o Perform Steps 1 through 8 without delay. Do NOT implement Function Restoration Procedures prior to completion of these steps.

o SI recirculation flow to RCS must be maintained at all times.

o Switchover to recirculation may cause high radiation levels in the reactor auxiliary building. Radiation levels must be assessed prior to performance of local actions in the affected areas.

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NOTE:

o Foldout applies.

o A minimum of 142 INCHES CNMT wide range sump level ensures the recirculation sump strainers are completely submerged AND assures a long term recirculation suction source.

o The following sequence of steps to transfer to cold leg recirculation assumes operability of at least one train of safeguards equipment.

1. Establish RHR Pump Recirculation Alignment:

   a. Verify both RHR pumps - RUNNING

   b. Verify CNMT sump to RHR pump suction valves - OPEN:

      o Train A RHR pump:

        1SI-300 AND 1SI-310

      o Train B RHR pump:

        1SI-301 AND 1SI-311

   c. Shut RWST to RHR pump suction valves:

      1SI-322 (Train A)
      1SI-323 (Train B)

   d. Shut low head SI Train A to cold leg valve:

      1SI-340

      RNO: d. Shut low head SI Train B to cold leg valve:

      1SI-341

   e. Check RHR pump recirculation alignment - AT LEAST ONE TRAIN ESTABLISHED

      RNO: e. GO TO EPP-012, "LOSS OF EMERGENCY COOLANT RECIRCULATION", Step 1.

Instructions	Response Not Obtained
2. Establish CSIP Recirculation Alignment:	
a. Shut CSIP alternate miniflow isolation valves: 1CS-746 (Train A CSIP) 1CS-752 (Train B CSIP)	a. Shut the associated block valve: 1CS-745 (Train A CSIP) 1CS-753 (Train B CSIP)
b. Verify normal miniflow isolation valves - SHUT 1CS-182 1CS-196 1CS-210 1CS-214	
c. Open RHR discharge to CSIP suction valves: 1RH-25 1RH-63	
d. Reset SI.	d. IF any train of SI will NOT reset at MCB, THEN reset at SSPS using PATH-1 GUIDE, Attachment 12.
e. Manually realign safeguards equipment following a loss of offsite power. (Refer to PATH-1 GUIDE, Attachment 2.)	
f. Shut RWST to CSIP suction valves AND place in pull-to-lock position: 1CS-291 (LCV-115B) 1CS-292 (LCV-115D)	

2-4

DEVICES:	LS-0990CW	**SETPOINT**: 23.4% (144.6 in WC)	**REFUELING WATER STORAGE TANK 2/4 LOW LOW LEVEL**
	LS-0991CW	23.4% (144.6 in WC)	
	LS-0992CW	23.4% (144.6 in WC)	
	LS-0993CW	23.4% (144.6 in WC)	

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using RWST level indicators LI-990, LI-991, LI-992, and/or LI-993.

2. **VERIFY** Automatic Functions:
   a. With an SI signal and 2/4 low-low levels present, CNMT Sump to RHR Pump suction valves 1SI-300, 1SI-301, 1SI-310, and 1SI-311 open.
   b. With a CNMT Spray pump running and 2/4 low-low levels present, the following valves reposition for the running pump:
      (1) 1CT-105 or 1CT-102, CNMT Sump to CT Pump, opens.
      (2) 1CT-26 or 1CT-71, RWST to CT Pump, shuts.

3. **PERFORM** Corrective Actions:
   a. **IF** alarm is due to SI,
      THEN GO TO EOP-PATH-1. **[Reference 7]**
   b. **DISPATCH** an operator to verify RWST level using LI-7110 (ACP) or LI-7116 (RWST Pit).
   c. **IF** RWST low level is due to a tank or supply line rupture,
      THEN GO TO AOP-008, Accidental Release of Liquid Waste.
   d. **IF** RWST low level is due to makeup to the RCS
      AND RWST level is approaching 12%,
      THEN:
      (1) **REFER TO** OP-107.01, CVCS Boration, Dilution, and Chemistry Control
      (2) **INITIATE** action to fill the RWST.
   e. **IF** neither SI nor CNMT Spray are actuated,
      THEN VERIFY SHUT ALL Recirculation Sump suction valves.
   f. **MONITOR** RWST, RCS, and Fuel Pool levels as necessary.
   g. **IF** maintenance is to be performed,
      THEN GO TO OWP-ESF.

CAUSES:

1. Safety Injection or Containment Spray actuation
2. Tank or supply line rupture below low-low level setpoint
3. RHR or CSS Recirculation Sump suction valves open
4. Use of water to fill Refueling Cavity during refueling
5. 1CT-23, Fuel Pool Cooling System Supply Valve, open
6. Improper valve alignment
7. Instrument or alarm circuit malfunction

REFERENCES:

1. AOP-008, Accidental Release of Liquid Waste
2. EOP-PATH-1
3. OP-107.01, CVCS Boration, Dilution, and Chemistry Control
4. OWP-ESF
5. 6-B-401 0046C, 0416-0419, 0636, 1039-1042
6. Technical Specifications 3.1.2.5, 3.1.2.6, 3.3.2, 3.3.3.6, and 3.5.4
7. FSAR Section 6.3.5

4. 2009A NRC RO 004

Given the following plant conditions:
- The plant is operating at 100% power
- The PRZ Level Control Selector Switch is selected to 459/460

The following conditions are observed:
- Actual Pressurizer level is trending DOWN
- VCT level is trending UP
- Auto Makeup is NOT in progress
- RCS temperature and pressure are stable

Which ONE of the following describes the event in progress?

A. Charging Line leak outside CNMT

B.✓ Charging Flow Control Valve failure

C. LT-460, Pressurizer Level, is failing LOW

D. LT-460, Pressurizer Level, is failing HIGH

Plausibility and Answer Analysis

A *Incorrect. This would cause pressurizer level to lower, but VCT level would not be trending up.*

B *Correct. Failure of the charging flow control valve in the shut direction results in charging being less then letdown therefore RCS inventory decreases, lowering pressurizer level. Letdown is returning to VCT but outflow from VCT decreases causing VCT level to increase.*

C *Incorrect. Plausible since pressurizer level is trending down, but if pressurizer level were trending down due to a failed channel then CVCS would respond by increasing charging flow to raise actual level and VCT level would then be trending down.*

D *Incorrect. Plausible since the indicated conditions are correct if 459 were failing high.*

KA Statement - Loss of Rx Coolant Makeup - Knowledge of the operational implications of the following concepts as they apply to the (ABNORMAL PLANT EVOLUTION): (CFR: 41.8 to 41.10 / 45.3): Relationship between charging flow and Pressurizer level

Importance Rating:	3.0 3.4
Technical Reference:	APP-ALB-009 2-2 Rev. 11, page 9
References to be provided:	None
Learning Objective:	CVCS, Obj. 12
Question Origin:	OIT Development Bank
Comments:	KA is matched since a Loss of Reactor Coolant Makeup is in progress with the charging flow control valve failed. APP-ALB-009 2-2 for Pressurizer control low level deviation will alarm when actual level is 5% below programmed level. The operator is directed to verify that the Pressurizer level control system is returning level to normal caused by letdown flow greater than charging flow. In this case the Pressurizer level is trending down and VCT level is trending up. Both are indications that there is more letdown than charging. This is an indication of a possible instrumentation malfunction per the APP.

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	APP-ALB-009
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	022AK1.03	K/A 2:	

DEVICES: LS-01RC-0459EW

SETPOINT: 5% below programmed level

PRESSURIZER CONTROL LOW LEVEL DEVIATION

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. Pressurizer level LI-459A1, LI-460, LI-461.1
   b. LRC-0459B, Pressurizer Level Control Setpoint

2. **VERIFY** Automatic Functions: None

3. **PERFORM** Corrective Actions:
   a. **IF** level deviation is due to load change,
   THEN VERIFY that the Prz Level Control system is returning level to normal.
   b. **IF** Tavg is stable,
   THEN ADJUST Charging and Letdown flow to bring Prz level to normal, using OP-107, Chemical and Volume Control.
   c. **IF** RCS leakage is indicated,
   THEN GO TO AOP-016, Excessive Primary Plant Leakage.
   d. **IF** maintenance is to be performed,
   THEN REFER TO OWP-RP, Reactor Protection.

CAUSES:

1. Letdown flow greater than charging flow
2. Sudden load increase
3. Sudden decrease in Tavg
4. RCS leakage
5. Alarm circuit or instrumentation malfunction

REFERENCES:

1. Tech Specs 3.3.1, 3.3.3.6, and 3.4.6.2
2. AOP-016, Excessive Primary Plant Leakage
3. OP-107, Chemical and Volume Control
4. OWP-RP, Reactor Protection
5. 2166-B-401 0939

5. 2009A NRC RO 005

Given the following plant conditions:
- The plant is in Mode 5
- 'A' RHR pump is in operation for Shutdown Cooling
- 'B' RHR pump is secured while packing is adjusted on 1RH-40, RHR Pump 'B' Hot Leg Suction Valve
- 1RH-40 is SHUT for packing adjustment
- PT-402, RCS Wide Range Pressure, fails HIGH

The following occur:
- 'A' RHR Pump trips on overcurrent
- The crew has entered AOP-020, Loss of RCS Inventory or RHR While Shutdown

Which ONE of the following is the action to be taken outside of the MCR to recover 'B' RHR and what is the effect of this action?

A. The Test Switch for PT-402 must be placed in TEST
Applies 'B' train interlocks to 1RH-40

B.✓ The Test Switch for PT-402 must be placed in TEST
Defeats the pressure interlock for 1RH-40

C. 1RH-40 must be connected to its alternate power supply
Applies 'B' train interlocks to 1RH-40

D. 1RH-40 must be connected to its alternate power supply
Defeats the pressure interlock of 1RH-40

Plausibility and Answer Analysis

AOP-020 Attachment 2 directs recovery of remote operation of non functional valves. 1RH-40 is interlocked with PT-402 and will not open if RCS Pressure is high.

A Incorrect. Attachment 2 does direct placing PT-402 Test Switch to TEST but does not transfer interlocks to 'B' Train as connecting to the alternate power supply does.

B Correct. Attachment 2 directs placing PT-402 Test Switch to TEST to defeat the RCS Pressure High Interlock.

C Incorrect. Attachment 2 directs connecting to an alternate power supply but only if 1RH-40 will not open due to loss of power.

D Incorrect. Attachment 2 directs connecting to an alternate power supply but only if 1RH-40 will not open due to loss of power. Effect is correct, but does not occur when connecting alternate power supply to valve.

KA Statement - Loss of RHR System - Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects

Importance Rating:	4.2 4.1
Technical Reference:	AOP-020 Rev. 31, page 71
References to be provided:	None
Learning Objective:	RHR, Obj. 7d
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	4	Reference:	AOP-020
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	025AG2.4.34	K/A 2:	

Attachment 2 - Remote Operation of Non-Functional RHR Loop Suction Valves

Sheet 1 of 1

To establish remote operating capability an RHR loop suction valve that will not open due to a failed wide range RCS pressure transmitter, place the indicated switch in TEST: **[A.1]**

NOTE

Placing Switch BS1 in TEST on PIC-4 will make B Train RVLIS inoperable and cause the pressure indicator associated with the pressure transmitter to fail low.

Failed Transmitter	Cabinet	Card	Switch
PT-402	PIC-1	846	BS1
PT-403	PIC-4	824	BS1

To establish remote operating capability for an RHR loop suction valve powered from the opposite train that will not open due to a loss of power, direct Maintenance to connect alternate power to the valve using the applicable procedure listed below: **[C.4]**

NOTE

- If 1RH-40 is connected to alternate power (SB from 1RH-1), 1RH-1 becomes inoperable, and all its interlocks will apply to 1RH-40.
- If 1RH-1 is connected to alternate power (SA from 1RH-40), 1RH-40 becomes inoperable, and all its interlocks will apply to 1RH-1.
- The necessary changes will take approximately 4 hours. The CWDs applicable to complete the changes are 326, 327, 336, and 337.

1RH-1:

- MST-I0277, Electrical Power Feed Switchover for RHR Inlet Isolation Valve 1RH-1

1RH-40:

- MST-I0276, Electrical Power Feed Switchover for RHR Inlet Isolation Valve 1RH-40

--END OF ATTACHMENT 2--

OPERATOR PREREQUISITE SUMMARY SHEET

3. No testing shall be allowed in Channel P-0403 during procedure performance; unless the following conditions exist:

   a. Safety Injection Alternate Miniflow valves are **NOT REQUIRED** to be OPERABLE **AND** ONE of the following conditions exist:

      (1) RHR Suction valve Interlock is **NOT REQUIRED** to be OPERABLE.

      OR

      (2) RHR is **ALREADY** in service.

4. **RHR INTERLOCK** that prevents OPENING valves 1RH-2 and 1RH-40 until RCS Wide Range Pressure is below 363 PSIG will be **DEFEATED**, when this loop is placed in **TEST**.

5. This procedure will place channel P-0402 in a TEST condition. IF plant **IS** in Midloop Operation **WITH** fuel in Rx Vessel, THEN this procedure **CAN NOT** be performed until Unit SCO has signed off that he has reviewed OPERATOR PREREQUISITE SUMMARY SHEET and grants permission to perform procedure when in Midloop **WITH** fuel in Rx Vessel.

6. Procedure performance will cause the following to occur in the circuit for CVCS Miniflow valve 2CS-V757SA-1 (1CS-746):

   a. CYCLING of CONTACTS that provide AUTOMATIC OPENING on high RCS pressure coincident with an SI (Control switch in NORMAL).

   b. CYCLING of CONTACTS that provide AUTOMATIC CLOSING on low RCS pressure coincident with an SI (Control switch in NORMAL). If the contacts are CLOSED during an SI, 1CS-746 can NOT be OPENED until the RWST SI RESET switch has been taken to RESET OR contacts are CYCLED OPEN.

7. This procedure affects equipment covered by the following L.C.O.s:

a.	3.1.2.3	g.	3.4.1.4.2
b.	3.1.2.4	h.	3.5.2 (Action Item Assignment 92H0910)
c.	3.3.3.5	i.	3.5.3 (Action Item Assignment 92H0910)
d.	3.3.3.6	j.	3.9.8.1
e.	3.4.1.3	k.	3.9.8.2
f.	3.4.1.4.1		

6. 2009A NRC RO 006

Given the following plant conditions:

- The plant is operating at 100% power
- At 1530, the crew entered AOP-018, RCP Abnormal Conditions, due to increasing RCP temperatures
- The following RCP temperatures are observed:

Time	A Mtr Brg Temp	B Mtr Brg Temp	C Mtr Brg Temp
1531	168°F	172°F	171°F
1533	170°F	184°F	171°F
1535	172°F	187°F	172°F
1537	172°F	192°F	172°F
1539	173°F	206°F	172°F
1541	173°F	231°F	173°F

Which ONE of the following is the time the reactor is required to be tripped and the additional operator action(s) required after the reactor trip?

	Time	Additional Action(s)
A.	1537	Stop ALL RCPs
B.✓	1537	Stop ONLY the 'B' RCP
C.	1541	Stop ALL RCPs
D.	1541	Stop ONLY the 'B' RCP

Plausibility and Answer Analysis

AOP-018 addresses Loss of CCW to the RCPs and Attachment 1 lists the limits for RCPs.

A *Incorrect. The time is correct but the action is incorrect, directions are to stop the affected pump ONLY. This is plausible since AOP-014 has all RCPs secured if a complete loss of CCW has occurred.*

B *Correct. At 1537, the Motor Bearing Temperatures of the B RCP exceed their limit of 190°F and directions are to stop the affected RCP.*

C *Incorrect. Time is incorrect, plausible if candidate confuses setpoint for Motor Bearings and Pump bearings. Action is to stop ONLY the affected RCP. This is plausible since AOP-014 has all RCPs secured if a complete loss of CCW has occurred.*

D *Incorrect. The action is correct but the time is incorrect. At 1541, the Motor Bearing Temperature exceeds 230°F. This is the limit for the Pump Bearing Temperature but Motor Bearings are the concern as Seal Injection is maintained.*

KA Statement - Loss of Component Cooling Water - Ability to determine and interpret the following as they apply to (ABNORMAL PLANT EVOLUTION): The normal values and upper limits for the temperatures of the components cooled by CCW
(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	2.5 2.9
Technical Reference:	AOP-014 Rev. 31 Loss of CCW Pump and Attachment 1 RCP Trip limits, pages 35-36
References to be provided:	None
Learning Objective:	Lp-AOP-3.18, Obj. 3
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AOP-014
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	026AA2.04	K/A 2:	

Attachment 1
Sheet 1 of 2
Reactor Coolant Pump Trip Limits

NOTE

False indications such as step changes or spikes on both the upper and lower thrust bearings are signs that the instrumentation transient may not be valid.

Validation of the temperatures should be performed by observing positive indications of any of the following:

- Simultaneous temperature increases in upper and lower thrust bearing and upper guide bearing (may indicate loss of CCW cooling or oil viscosity problems common to the upper reservoir)
- Vibration levels increasing along with increasing bearing temperatures.
- High or low RCP oil level alarms along with increasing bearing temperatures.
- ERFIS points TRC0417A and TRC0439 have been defeated from processing during Cycle 13.
- Reference HNP POM Group Trends for AOP-018.

☐**1.** ANY of the following Motor Bearing temperatures exceeding 190°F: **[A.1, 2]**

	ERFIS Points		
	RCP A	RCP B	RCP C
Mtr Upper Thrust Brg Temp	TRC0417A	TRC0427A	TRC0437A
Mtr Lower Thrust Brg Temp	TRC0417B	TRC0427B	TRC0437B
Mtr Upper Radial Brg Temp	TRC0418A	TRC0428A	TRC0438A
Mtr Lower Radial Brg Temp	TRC0419	TRC0429	TRC0439

☐**2.** ANY of the following Pump temperatures exceeding 230°F: **[A.1, 2]**

	ERFIS Points		
	RCP A	RCP B	RCP C
Pump Radial Brg Temp	TRC0131	TRC0128	TRC0125
Seal Water Inlet Temp	TRC0132	TRC0129	TRC0126

☐**3.** RCP Stator Winding temperature exceeding 300°F:

	ERFIS Points		
	RCP A	RCP B	RCP C
Motor Stator Windg Temp	TRC0418B	TRC0428B	TRC0438B

Attachment 1
Sheet 2 of 2

Reactor Coolant Pump Trip Limits

NOTE

ALB-5-1-2B, RCP THERM BAR HDR LOW FLOW, indicates loss of CCW to all RCP thermal barriers.

☐**4.** Loss of RCP seal injection when ANY of the following conditions exist:

- CCW flow is lost to the associated RCP Thermal Barrier HX
- RCS temperature is greater than or equal to 400°F AND CCW HX outlet temperature is greater than 105°F
- RCS temperature is less than 400°F AND CCW HX outlet temperature is greater than 125°F

☐**5.** RCP vibration in excess of the following: **[A.1]**

- 20 mils shaft
- 15 mils shaft and increasing greater than 1 mil/hr
- 5 mils frame
- For A and C RCPs ONLY: 3 mils frame and increasing greater than 0.2 mil/hr
- For B RCP ONLY: 3.5 mils frame and increasing greater than 0.2 mils/hr

☐**6.** RCP Motor current fluctuations of 40 amps peak-to-peak:

	ERFIS Points		
	RCP A	RCP B	RCP C
Motor Current	IRC0160	IRC0161	IRC0162

☐**7.** Loss of CCW to an RCP or RCP Motor when:

- An RCP has operated for 10 minutes without CCW flow to either motor oil cooler **[A.1, 2]**
- Isolation of CCW to an RCP is necessary to stop excessive CCW System leakage

-- END OF ATTACHMENT 1 --

7. 2009A NRC RO 007

Given the following plant conditions:
- The plant is operating at 100% power
- Pressurizer Pressure is 2050 psig and stable
- The crew enters AOP-019, Malfunction of RCS Pressure Control
- PK-444A is placed in MANUAL

Which ONE of the following describes the action required to return pressure to 2235 psig?

A.✓ Lower the controller output

B. Raise the controller output

C. Lower the pressure setpoint adjustment

D. Raise the pressure setpoint adjustment

Plausibility and Answer Analysis

A Correct. Lowering controller output will energize heaters and raise pressure.

B Incorrect. Raising controller output will de-energize heaters/open spray valves lowering pressure. Plausible if candidate doesn't understand the inverse relationship between controller output and actual pressure, ie., believes raising controller output raises pressure.

C Incorrect. Once in manual adjusting the setpoint will have no effect. Plausible if applicant believes setpoint is still in the control circuitry while in manual.

D Incorrect. Once in manual adjusting the setpoint will have no effect. Plausible if applicant believes setpoint is still in the control circuitry while in manual.

KA Statement - Pressurizer Pressure Control System Malfunction - Ability to operate and / or monitor the following as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.7 / 45.5 / 45.6) Pressure control when on a steam bubble

Importance Rating:	3.6 3.5
Technical Reference:	AOP-019 Rev. 21, pages 4, 5, & 20 MST-I0080 Rev. 10, page 62 PZRPC Student Text
References to be provided:	None
Learning Objective:	LP-AOP-3.19, Obj. 4
Question Origin:	2008 NRC RO Exam (BANK OIT Development Bank 027 AA1.03 027)
~~Comments:~~	~~K/A Match from 2008 NRC RO Exam~~

Origin:	PREVIOUS NRC	Cog Level:	F
Difficulty:	2	Reference:	AOP-019
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	027AA1.03	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

Steps 1 through 3 are immediate actions.

CAUTION

A pressure transmitter or indicator malfunction may exist. When referred to throughout this procedure, actual RCS pressure should be obtained by cross-checking of diverse instrumentation, such as PI-455.1, PI-456, and PI-457.

☐ 1. **CHECK** that a bubble exists in the PRZ.

1. **PERFORM** the following:

☐ a. **IF** PRZ pressure is GREATER THAN 360 PSIG,
THEN STOP the running charging pump.

☐ b. **GO TO** Section 3.2, Pressure Control Malfunctions During Solid Plant Operation.

NOTE

Failure of a PRZ PORV to fully reseat after operation may require initiation of the Emergency Plan.

☐ 2. **VERIFY** ALL PRZ PORVs **AND** associated block valves properly positioned for current PRZ pressure and plant conditions.

☐ 2. **IF** ANY PRZ PORV will NOT shut when required,
THEN SHUT its associated block valve.

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	
☐ 3. **CHECK** BOTH PRZ Spray Valves properly positioned for current PRZ pressure and plant conditions.	3. **CONTROL** PRZ Spray Valves using ONE of the following methods (listed in order of preference): ☐ a. AFFECTED Spray Valve controller in MANUAL (if only one is obviously malfunctioning) **OR** ☐ b. PK-444A, Master Pressure Controller, in MANUAL **OR** ☐ c. BOTH individual Spray Valve controllers in MANUAL
☐ 4. **GO TO** Section 3.1, Pressure Control Malfunctions While Operating With a Pressurizer Bubble.	

—END OF SECTION 3.0—

Attachment 2

Sheet 1 of 1

Pressurizer Pressure Controller PK-444A Operation

1. **SETPOINT**

Set at 66.88% (corresponding to 2235 psig) during normal operation.

In AUTO, a 1% change in setpoint will result in an initial 5% change in output signal and an equilibrium pressure change of 8 psig.

2. **OUTPUT**

The following will occur at the given outputs in AUTO or MAN:

87.5%	PCV-444B SB opens
75%	PCV-444B SB shuts
71.87%	Spray valves full open
40.62%	Spray valves full shut
34.56%	Proportional heaters full off
25%	Nominal output at 2235 psig
15.62%	Proportional heaters full on
14.38%	Backup heaters off
9.4%	Backup heaters on

—END OF ATTACHMENT 2—

8. 2009A NRC RO 008

Given the following plant conditions:

- The plant is operating at 51% power and ramping up to 100% following an outage
- High vibrations are observed on 'B' RCP
- The crew enters AOP-018, RCP Abnormal Conditions
- It is determined that a manual Reactor Trip and a trip of 'B' RCP is required by AOP-018
- The Reactor fails to trip either automatically or manually, but the Turbine Trip is successful

Which ONE of the following actions should the Reactor Operator be directed to take in accordance with FRP-S.1, Response to Nuclear Power Generation / ATWS?

A. Immediately trip the 'B' RCP

B. Trip the 'B' RCP immediately after FRP-S.1 immediate actions are completed

C. Continue operating ALL RCPs until power is <48% and then trip 'B' RCP

D✓ Continue operating ALL RCPs until power is <5% and then trip 'B' RCP

Plausibility and Answer Analysis

A *Incorrect. This would be correct if FRP-S.1 were not in effect and plant power was less than P-8 (49%) but FRP-S.1 is in effect and power is at P-8.*

B *Incorrect. With power greater than P-8 (49%), the RCP is tripped after immediate actions of Path-1 are complete but FRP-S.1 is in effect.*

C *Incorrect. Rod insertion is directed in FRP-S.1 as an Immediate Action which will reduce power to less than P-8 (49%) but with FRP-S.1 in effect, all RCPs should be left running.*

D *Correct. To maximize core cooling, RCPs should NOT be tripped with reactor power GREATER THAN 5%. (Normal support conditions for running RCPs are NOT required for these circumstances. The RCP TRIP CRITERIA for small break LOCA conditions is NOT applicable to this procedure.)*

KA Statement - ATWS - Ability to execute procedure steps.

Importance Rating:	4.6 4.6
Technical Reference:	FRP-S.1 Rev. 15, page 3
References to be provided:	None
Learning Objective:	EOP-LP-3.15, Obj. 2
Question Origin:	NEW
Comments:	Must analyze the requirements AOP-018 but apply a caution from FRP-S.1

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	FRP-S.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	029EG2.1.20	K/A 2:	

Instructions	Response Not Obtained

****

CAUTION

To maximize core cooling, RCPs should NOT be tripped with reactor power GREATER THAN 5%. (Normal support conditions for running RCPs are NOT required for these circumstances. The RCP TRIP CRITERIA for small break LOCA conditions is NOT applicable to this procedure.)

****

NOTE: Steps 1 through 4 are immediate action steps.

---

1. Verify Reactor Trip:

Instructions	Response Not Obtained
a. Check for all of the following: o Check for any of the following: o Trip breakers RTA AND BYA - OPEN o Trip breakers RTB AND BYB - OPEN o Rod bottom lights - LIT o Neutron flux - DECREASING	a. IF the reactor will NOT trip (automatically OR using either manual trip switch), THEN verify negative reactivity inserted by any of the following while continuing with this procedure: o Manually insert control rods. o Verify control rods inserting in automatic.

9. 2009A NRC RO 009

Given the following plant conditions:

- 'B' Steam Generator has been identified as Ruptured and Isolated in accordance with PATH-2
- 'A' and 'C' Steam Generators are intact and were used for cooldown
- All RCPs are tripped
- The crew is evaluating SI Termination Criteria in PATH-2
- B Steam Generator Pressure reads 1050 psig
- 'A' Loop Wide Range Thot (TI-413) reads 505°F
- 'B' Loop Wide Range Thot (TI-423) reads 525°F
- 'C' Loop Wide Range Thot (TI-433) reads 500°F
- RCS Wide Range Pressure (PI-402) reads 1435 psig
- RCS Wide Range Pressure (PI-403) reads 1385 psig

Which ONE of the following is RCS subcooling?

A. 62°F

B. 67°F

C.✓ 82°F

D. 87°F

Plausibility and Answer Analysis

The EOP User's Guide Section 6.2 provides instructions for Calculating Subcooling.

A *Incorrect. This number is derived using the lowest RCS Pressure (correct) and the Highest Thot (wrong). 'B' Loop is NOT active as it is isolated and not steaming.*

B *Incorrect. This number is derived using the highest RCS Pressure (wrong) and the Highest Thot (wrong) but 'B' Loop is NOT active as it is isolated and not steaming.*

C *Correct. This number is derived using the Highest Active Loop Thot and the lowest RCS Pressure.*

D *Incorrect. This number is derived using the highest RCS Pressure (wrong) and the Highest Active Loop Thot (correct).*

KA Statement - Steam Gen. Tube Rupture - Knowledge of the physical connections and/or cause-effect relationships between (SYSTEM) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Use of steam tables

Importance Rating:	3.1 3.4
Technical Reference:	EOP User's Guide Rev. 26, page 33-34 Steam tables
References to be provided:	Steam tables
Learning Objective:	EOP-LP-3.19, Obj. 4a
Question Origin:	NEW
Comments:	(K/A match) Candidate must understand that the loop with the SGTR is not active and properly use steam tables to calculate subcooling.

Origin:	NEW	Cog Level:	H
Difficulty:	4	Reference:	USERS GUIDE
Ref. Provided?:	STEAM TABLES	Key Words:	2009A NRC RO
K/A 1:	038EK1.01	K/A 2:	

6.0 GENERAL INFORMATION

6.1 Background and Basis for EOP Network

During the verification and validation of the EOP network, and during Operator simulator training, many questions are frequently asked concerning the EOP network. The background and basis for the SHNPP EOP network is found in detail in the background documents of the WOG ERGs and the EOP Step Deviation Documents. The following sections contain clarification of certain procedural requirements that are frequently questioned by the operators.

6.2 RCS Subcooling

The ERFIS plant computer functions as the plant "subcooling monitor". RCS subcooling will normally be obtained from the top level Safety Parameter Display System (SPDS) screen (Turn-on-code: "SPTOP", Parameter: "SUBCOOL (DEGF)", ERFIS point: TRC9400). If for some reason the subcooling monitor is not available, the operators will manually determine subcooling using one of the following (Reference 2.2.2.2):

- o Graph provided on the CSFSTs
- o "Subcooling Margin Calc. Program" - Version 1.0
- o Steam Tables

Subcooling values are generally presented in the following format:

10°F [40°F] - C
20°F [50°F] - M

The top set of values is normally used when the subcooling monitor is available (designated by C). The bottom set of numbers is used only when the subcooling monitor is not available (designated by M for manual). The subcooling values used in the procedure were determined based on specific instrument inaccuracies. Should it be necessary to manually determine subcooling, the following conventions apply:

1. Primary temperature is obtained using one of the following based on availability of the indications (listed in order of preference):

   - o Core exit TC reading on SPDS (Turn-on-code: "SPTOP", Parameter: "T EXIT (DEGF)", ERFIS point: TRC9300). This reading is the average of the five hottest core exit TCs and is the input used for the Subcooling Monitor.

   - o Highest core exit TC reading from the Inadequate Core Cooling Monitor (ICCM).

6.2 RCS Subcooling (continued)

- o Highest active loop wide range T-hot (TI-413, 423, 433). An active loop is defined as one that has forced or natural circulation flow. If any RCPs are running, all loops will be active (backflow is available in loops where RCPs are not running). A classic example of a non-active loop would be a loop that has a SGTR since it is isolated and natural circulation flow in this loop would not be available.

2. Primary pressure is obtained using one of the following based on the range and availability of RCS and PRZ pressure indication:

- o If ERFIS is available, then use the RCS pressure reading on SPDS (Turn-on-code: "SPTOP", Parameter: "PRZ PRES (PSIG)", ERFIS point PRC9455L). If PRZ pressure is above 1700 PSIG, this reading is the lowest of the three PRZ pressure channels (PT-457, PT-456, and PT-455). IF PRZ pressure is below 1700 PSIG, this reading is the lowest of the two RCS wide range pressure channels: PT-402 (ERFIS point PRC0402) and PT-403 (ERFIS point PRC0403).

- o If PRZ pressure is greater than 1700 PSIG and CNMT conditions are normal, then use the lowest PRZ pressure indication (PI-457, PI-456, or PI-455.1).

- o If PRZ pressure is off scale low or adverse CNMT conditions exist, then use the lowest of the two RCS wide-range pressure indications PI-402.1 or PI-403. Only PT-402 and PT-403 are used since these transmitters are located outside containment.

- o When RCS pressure is less than 700 PSIG, PI-402A should be used. PI-402A receives input from qualified instrument PT-402 and its narrow range scale provides a more precise indication of pressure.

10. 2009A NRC RO 010

Given the following plant conditions:
- The plant is operating at 100% power
- OWP-ESF-01 is in place for CNMT Pressure Channel III

Subsequently, Instrument Bus S-II is lost.

Which ONE of the following describes the status of the Channel II CNMT HI-3 Pressure Bistable light on TSLB-4 and the CNMT Spray Actuation Signal (CSAS)?

	TSLB-4 CNMT HI-3	CSAS
A.	illuminated	actuated
B.	illuminated	NOT actuated
C.	extinguished	actuated
D.✓	extinguished	NOT actuated

Plausibility and Answer Analysis

A *Incorrect. The TSLB Light for Channel II HI-3 will be extinquished because this is an energize to actuate function and the Instrument Bus is lost, however most functions are deenergize to actuate. The CSAS will NOT be actuated as this is an energize to actuate function, however most functions are deenergize to actuate.*

B *Incorrect. The TSLB Light for Channel II HI-3 will be extinquished because this is an energize to actuate function and the Instrument Bus is lost, however most functions are deenergize to actuate. The CSAS will NOT be actuated as this is an energize to actuate function.*

C *Incorrect. The TSLB Light for Channel II HI-3 will be extinquished because this is an energize to actuate function and the Instrument Bus is lost. The CSAS will NOT be actuated as this is an energize to actuate function, however most functions are deenergize to actuate.*

D *Correct. The TSLB Light for Channel II HI-3 will be extinquished because this is an energize to actuate function and the Instrument Bus is lost. The CSAS will NOT be actuated as this is an energize to actuate function.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Loss of Vital AC Inst. Bus - Ability to determine and interpret the following as they apply to ABNORMAL PLANT EVOLUTION): (CFR: 41.10 / 43.5 / 45.13) ESF system panel alarm annunciators and channel status indicators.

Importance Rating:	3.7 4.0
Technical Reference:	OWP-ESF-01 Rev. 17 DWG 1364-871 (Westinghouse Logic 108D831 Seet 8)
References to be provided:	None
Learning Objective:	CSS, Obj. 7
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	OWP-ESF-01
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	057AA2.04	K/A 2:	

EIR Number: ____________
W/O Number: ____________
Clearance Number: ____________

1. OWP-ESF-01

2 System: ESFAS

3. Component: Containment Pressure

4. Scope: LCO Action required due to inoperable Containment Pressure Circuits (PT-950, PT-951, PT-952, or PT-953)

5. Applicable Requirements: 3.3.2 and 3.3.3.6

6. Precautions: (1) Ensure only one channel is in TEST at a time. (2) Ensure ESF Equipment is OPERABLE on the ESF Logic Train not being tested.

7. Component lineups completed per attached sheet (s) ____________ / ______
Signature Date

8. Testing required on redundant equipment while component is inoperable:
None

9. Testing/Action required to restore operability. (N/A if tracked on EIR)

MST-I0003 for Channel I		/
MST-I0004 for Channel II		/
MST-I0005 for Channel III		/
MST-I0006 for Channel IV		/
	Signature	Date

10. Component lineups restored per attached sheet (s) ____________ / ______
Signature Date

11. Remarks:

12. Reviewed by: ____________________ ____________
Superintendent- Shift Operations Date

After receiving the final review signature, this OWP becomes a QA Record and should be submitted to Document Services.

Bistable /Status Light Lineup

Component ID or Number	Position for Maintenance	Initial/Verified	Restored Position	Initial/Verified

Containment Pressure Channel III

PIC Cabinet 3 on Card C3-862

> **NOTE:** This switch may be re-positioned for troubleshooting. It is not required to be in TEST to meet Tech Specs. Operating this switch first aids in troubleshooting by maintaining system conditions the same as they were when the trouble occurred.

SW2 (P-0952 Master Test Switch)	TEST	/	NORMAL	/

PIC Cabinet 3 on Card C3-832

> **NOTE:** Concurrent verification is preferred while tripping bistables.

BS3 (PB/952A Cont Press HI 3 Spray Actuation	TEST	/	NORMAL	/
BS2 (PB/952B2 Cont Press HI 2 MS Line Isol	TEST	/	NORMAL	/
BS1 (PB/952B1 Cont Press HI 1 SI Actuation	TEST	/	NORMAL	/

Trip Status Light Box-4

CNMT HI-1 PRESS PB 952B1 (Window 12-3)	ENERGIZED	/	DE-ENERGIZED	/
CNMT HI-2 PRESS PB 952B2 (Window 13-3)	ENERGIZED	/	DE-ENERGIZED	/

Bypass Permissive Panel

CNMT SPRAY ACUTATION BYPASS CHAN III TEST (Window 5-7)	ENERGIZED	/	DE-ENERGIZED	/

On ERFIS Computer (Using DR Functional)

PCT0952	Deleted from Processing	/	Restored to Processing	/

11. 2009A NRC RO 011

Given the following plant conditions:
- The plant is operating at 100% power
- The 1B-SA Battery Charger is under clearance

The following has just occurred:
- ALB-015-4-4,125 VDC Emergency Bus 'A' Trouble is received
- The field operator reports that the 1A-SA Battery Charger AC Input and DC Output Breakers have tripped with smoke coming from the charger

Which ONE of the following describes the power supplies to the 'A' Train Safety Related Instrument Buses Inverters immediately after this event?

The Instrument Bus Inverters continue to supply the Instrument Buses with the:

A.✓ NORMAL 480 VAC power supplying the Inverter and 125 VDC available as a BACKUP as long as the DC Input Breaker remains closed.

B. NORMAL 480 VAC power supplying the Inverter but 125 VDC is NOT available as a BACKUP due to low battery voltage caused by the loss of chargers.

C. BACKUP 480 VAC power available and NORMAL 125 VDC power supplying the Inverter as long as the DC Input Breaker remains closed.

D. BACKUP 480 VAC power supplying the Inverter because the NORMAL 125 VDC is NOT available due to low battery voltage caused by the loss of chargers.

Plausibility and Answer Analysis

A *Correct. The normal power source to the Instrument Bus Inverters is 480VAC with safety related DC power available as a backup. The 125 VDC will still be available from the batteries even with both battery chargers offline.*

B *Incorrect. Normal source of power is correct, but loss of both battery chargers will not prevent the battery from supplying the Instrument Bus Inverters. Loss of Battery Chargers will cause voltage to drop and continue to decay however the voltage will not decay to the point that it can no longer supply the inverters for a lengthy time. HNP Safety Related Batteries are rated for 4 hours.*

C *Incorrect. The inverters have two supplies; a normal and a backup. 480 VAC is the normal source of power and DC is the backup. It is possible for the candidate to confuse these.*

D *Incorrect. The inverters have two supplies; a normal and a backup. 480 VAC is the normal source of power and DC is the backup. It is possible for the candidate to confuse these. Loss of Battery Chargers will cause voltage to drop and continue to decay however the voltage will not decay to the point that it can no longer supply the inverters for a lengthy time. HNP Safety Related Batteries are rated for 4 hours.*

KA Statement - Loss of DC Power - Ability to operate and / or monitor the following as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.7 / 45.5 / 45.6) Static inverter dc input breaker, frequency meter, ac output breaker and ground fault detector.

Importance Rating:	3.1 3.1
Technical Reference:	OP-156.02 Rev. 75 attachment 4, page 287-290 120 VUPS Student Text Rev. 3, page 5
References to be provided:	None
Learning Objective:	120VUPS, Obj. 5
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	OP-156.02
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	058AA1.02	K/A 2:	

Attachment 4 - Normal AC Electrical Lineup Checklist for Emergency Bus 1A-SA
Sheet 4 of 8

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	480V EMERGENCY 1A3-SA (continued)			
1A3-SA-5A	208/120 VAC Heater Power	ON		
1A3-SA-6B	Main Feeder Breaker	CLOSED		
	DP-1A-SA (RAB 286)			
DP-1A-SA-28	DC Bkr to Inverter Channel I	ON		
DP-1A-SA-29	DC Bkr to Inverter Channel III	ON		
	MCC-1A31-SA (RAB 286)			
1A31-SA-2B	Transformer PP-1A311-SA Feeder Bkr	ON		

NOTE: Loss of power to PP-1A312-SA, from Appendix R Inverter SA, will cause a loss of Sample flow to Plant Vent Stack WRGM, REM-3509. This is due to loss of power to Rad Mon Cab Train A Bay 1 which powers the grab sample controller (Reference 2.6.6).

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
1A31-SA-6AR	7.5 KVA Appendix R Inverter SA	ON		
1A31-SA-6AL	7.5 KVA Inverter Channel III	ON		
1A31-SA-1D	Power Panel Fdr to IDP-1A-SIII (PP-1A311 CKT 20)	ON		
	PP-1A311-SA (RAB 286)			
PP-1A311-SA-1	ARP-1A-SA	ON		
PP-1A311-SA-4	ARP-19A-SA	ON		
PP-1A311-SA-5	MCC-1A36-SA Space Heaters	ON		
PP-1A311-SA-7	MCC-1A34-SA Space Heaters	ON		
PP-1A311-SA-8	480V Bus 1A3-SA Space Heaters and Motors	ON		
PP-1A311-SA-9	MCC-1A31-SA Space Heaters	ON		
PP-1A311-SA-11	MCC-1A35-SA Space Heaters	ON		
PP-1A311-SA-13	ARP-3A	ON		
PP-1A311-SA-14	ARP-4A-SA	ON		
PP-1A311-SA-15	ARP-3A	ON		
PP-1A311-SA-16	Transfer Panel 1A Sect SA-TB1	ON		

Attachment 4 - Normal AC Electrical Lineup Checklist for Emergency Bus 1A-SA

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	7.5 KVA Appendix R Inverter SA (RAB 286)			

NOTE: Loss of power to PP-1A312-SA, from Appendix R Inverter SA, will cause a loss of sample flow to Plant Vent Stack WRGM, REM-3509. This is due to loss of power to Rad Mon Cab Train A Bay 1 which powers the grab sample controller (Reference 2.6.6).

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
4CB	Inverter AC Output Breaker	ON		
	PP-1A312-SA (RAB 286)			
PP-1A312-SA-6	PIC-CAB 17-SA	ON		
PP-1A312-SA-11	Aux Transfer Panel SA	ON		
PP-1A312-SA-18	Connected to line side of CKT 16	ON		
	Channel III Inverter (RAB 305)			
CKT 2CB	Battery Input Breaker	ON		
CKT 1CB	Rectifier AC Input Breaker	ON		
CKT 3CB	Inverter DC Input Breaker	ON		
CKT 4CB	Inverter AC Output Breaker	ON		
	IDP-1A-SIII (RAB 305)			
IDP-1A-SIII-3	Process Prot Set-PIII PIC-P3 (C3)	ON		
IDP-1A-SIII-7	Solid State Protection (A) Input (III)	ON		
IDP-1A-SIII-8	Solid State Protection (B) Input (III)	ON		
IDP-1A-SIII-9	CB Monitor Lights (MTC-10A-SA Pnl-1)	ON		
IDP-1A-SIII-12	PIC Cabinet C-13	ON		
IDP-1A-SIII-14	PIC Cab 9-SA	ON		
IDP-1A-SIII-15	I/O Cab CIO-1TH0052 ASA	ON		
IDP-1A-SIII-16	I/O Cab CIO-1TH0053 ASA	ON		
IDP-1A-SIII-17	I/O Cab CIO-1TH0057 ASA	ON		

Attachment 4 - Normal AC Electrical Lineup Checklist for Emergency Bus 1A-SA

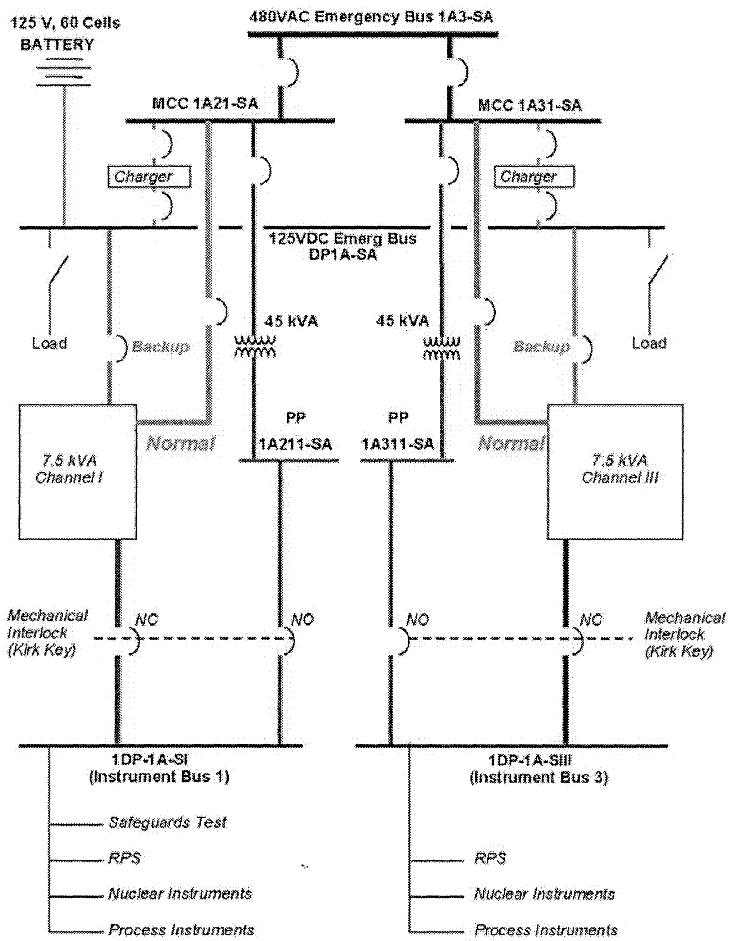
Sheet 6 of 8

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	Panel 1A-SI & 1A-SIII			
Cubicle 2A	Inst Channel III UPS (Inverter)	ON		
Cubicle 2B	PP-1A311-SA (Alt Pwr Sup)	OFF		
	MCC-1A21-SA (RAB 286)			
1A21-SA-1B	Instrument Power Panel Feeder to IDP-1A-SI	ON		
1A21-SA-2A	Power Panel 1A211-SA Feeder Breaker	ON		
1A21-SA-3A	Lighting Panel 533 Feeder Breaker (both breakers)	ON		
		ON		
1A21-SA-3BR	Channel I 7.5 KVA Inverter	ON		
1A21-SA-14AL	Fdr Bkr to Xfmr Fdr for LP-114 (Secondary)	ON		
1A21-SA-14AR	Fdr Bkr to Xfmr Fdr for LP-114 (Primary)	ON		
1A21-SA-14BL	Fdr Bkr to Xfmr Fdr for LP-118 (Secondary)	ON		
1A21-SA-14BR	Fdr Bkr to Xfmr Fdr for LP-118 (Primary)	ON		
1A21-SA-14CL	Fdr Bkr to Xfmr Fdr for LP-112 (Secondary)	ON		
1A21-SA-14CR	Fdr Bkr to Xfmr Fdr for LP-112 (Primary)	ON		
1A21-SA-14DL	Fdr Bkr to Xfmr Fdr for LP-134 (Secondary)	ON		
1A21-SA-14DR	Fdr Bkr to Xfmr Fdr for LP-134 (Primary)	ON		
1A21-SA-14EL	Fdr Bkr to Xfmr Fdr for LP-106 (Secondary)	ON		
1A21-SA-14ER	Fdr Bkr to Xfmr Fdr for LP-106 (Primary)	ON		
1A21-SA-14FL	Fdr Bkr to Xfmr Fdr for LP-128 (Secondary)	ON		

Attachment 4 - Normal AC Electrical Lineup Checklist for Emergency Bus 1A-SA
Sheet 7 of 8

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	MCC-1A21-SA (RAB 286) (continued)			
1A21-SA-14FR	Fdr Bkr to Xfmr Fdr for LP-128 (Primary)	ON		
1A21-SA-15A	Power Panel 1A212-SA Feeder Breaker	ON		
	PP-1A211-SA (RAB 286)			
PP-1A211-SA-5	MCC-1A21-SA Space Heater	ON		
PP-1A211-SA-7	ARP-4A-SA	ON		
PP-1A211-SA-13	ARP-2A-SA	ON		
PP-1A211-SA-14	ARP-2A-SA	ON		
PP-1A211-SA-21	480V Bus 1A2-SA and Motors Space Heater	ON		
PP-1A211-SA-22	Inst Cab A1-C7	ON		
	Channel I Inverter (RAB 306)			
CKT 2CB	Battery Input Breaker	ON		
CKT 1CB	Rectifier AC Input Breaker	ON		
CKT 3CB	Inverter DC Input Breaker	ON		
CKT 4CB	Inverter AC Output Breaker	ON		
	IDP-1A-SI (RAB 305)			
IDP-1A-SI-3	STC A Pnl-1	ON		
IDP-1A-SI-4	Process Protection Set 1 (PIC-P1-C1)	ON		
IDP-1A-SI-5	Transfer Pnl-1A (Sect SA)	ON		
IDP-1A-SI-7	Solid State Protection (A) Input (I)	ON		
IDP-1A-SI-8	Solid State Protection (B) Input (I)	ON		
IDP-1A-SI-13	PIC-CAB 17-SA	ON		
IDP-1A-SI-15	I/O Cabinet CIO-1TH0051 1ASA	ON		
IDP-1A-SI-16	RVLIS Panel	ON		
IDP-1A-SI-18	RVLIS Plasma Display Console	ON		
	Panel 1A-SI & 1A-SIII			
Cubicle 1B	Instr Channel I UPS (Inverter)	ON		
Cubicle 1A	PP-1A211-SA (Alt Pwr Sup)	OFF		

Figure 2: Safety Related 120V



APPENDIX R 120V INSTRUMENT POWER PANELS & TRANSFORMERS

The other two safety related 120v power panels are the Appendix R 120v power panels, PP-1A312-SA and PP-1B312-SB (Figure 3). These are supplied by the Appendix R transformers. Some of the significant loads include the auxiliary transfer panels, backup power to PIC-17 and PIC-18, the A and B SG PORVs, the RM23 trains of radiation monitoring, and backup power for wide range flux monitor A.

12. 2009A NRC RO 012

Given the following plant conditions:
- The plant is operating at 100% power
- 'A' Train equipment is in service
- A rupture occurs in the Normal Service Water System
- The location of the rupture has NOT been identified
- BOTH ESW Trains received automatic start signals
- The crew is performing actions of AOP-022, Loss of Service Water

The following conditions are observed:
- 1A-SA ESW Pump trips upon starting
- 'A' ESW Header pressure is 0 psig

Which ONE of the following describes the required actions in accordance with AOP-022?

A. Immediately START the 'B' CSIP then STOP the 'A' CSIP

B. Immediately STOP the 'A' CSIP and then isolate letdown

C. If 'A' ESW flow is lost for > ONE minute, START the 'B' CSIP and then STOP the 'A' CSIP

D✓ If 'A' ESW flow is lost for > ONE minute, STOP the 'A' CSIP and then isolate letdown

Plausibility and Answer Analysis

A *Incorrect. Starting 'B' CSIP is incorrect. Plausible since starting the standby CSIP and then stopping the running pump would not require letdown to be isolated. This is the normal manner that pumps would be switched. The action to start the standby CSIP will be performed eventually to address the loss of ESW to the 'A' Train CSIP, but not before securing the 'A' CSIP.*

B *Incorrect. Plausible if candidate does not implement the immediate action correctly. Isolating letdown is correct response to the loss of charging.*

C *Incorrect. Starting the 'B' CSIP and stopping the 'A' CSIP after loss of flow for 1 minute is incorrect for the same reasons as answer A. However starting 'B' CSIP is incorrect. Plausible since this will be performed eventually to address the loss of charging but it is not directed by AOP-022. Immediately starting a non-running CSIP is directed by multiple procedures, but only when the CSIP is serving its safety injection function, not just for charging.*

D *Correct. If ESW flow is lost to a running CSIP for >1 minute then the CSIP should be stopped and letdown isolated in accordance with AOP-022.*

KA Statement - Loss of Nuclear Svc Water - Knowledge of the reasons for the following responses as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.5 / 41.10 / 45.6 / 45.13) Guidance actions contained in EOP for Loss of nuclear service water

Importance Rating:	RO 4.0
Technical Reference:	AOP-022 Rev. 28, page 4
References to be provided:	None
Learning Objective:	LP-AOP-3.22, Obj. 3
Question Origin:	New
Comments:	(KA Match) Loss of Service Water is addressed in the AOP network at Harris, not by the EOP network.

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	AOP-022
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	062AK3.03	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

Steps 1 and 2 are immediate actions.

INSTRUCTIONS	RESPONSE NOT OBTAINED
✱1. **CHECK** ESW flow lost to ANY RUNNING CSIP - MORE THAN 1-minute:	☐ 1. **GO TO** Step 2.
☐ a. **STOP** affected CSIP.	
☐ b. **CHECK** ANY CSIP RUNNING.	☐ b. **ISOLATE** Letdown by verifying the following valves SHUT: • 1CS-7, 45 GPM Letdown Orifice A • 1CS-8, 60 GPM Letdown Orifice B • 1CS-9, 60 GPM Letdown Orifice C
✱2. **CHECK** ESW flow lost to ANY RUNNING EDG - MORE THAN 1-minute:	☐ 2. **GO TO** Step 3.
☐ a. **PLACE** EMERGENCY STOP switch for affected EDG in EMER STOP.	

3. **GO TO** the appropriate step as indicated by the parameter LOST:

IF...LOST:	DUE TO ANY	GO TO
☐ ESW Header	• ESW Pump failure • ESW Pump loss of flow	3.0/ Step 4 (page 5)
☐ ESW Header	• Loss of NSW flow	3.0/ Step 5 (page 5)
☐ NSW Header	• NSW Pump failure • NSW Pump loss of flow	3.0/ Step 6 (page 6)
☐ ESW or NSW Header	• Any ESW or NSW Header Rupture	3.0/ Step 7 (page 7)
☐ Ultimate Heat Sink	• Main Reservoir level less than 215 feet • Aux Reservoir level less than 250 feet	3.3/ Step 1 (page 39)

13. 2009A NRC RO 013

Given the following plant conditions:
- The plant is operating at 100% power
- The Compressed Air System (CAS) Control Panel is in Sequence 1
- A loss of Auxiliary Bus 1D has occurred
- 'A' EDG is carrying Bus 1A-SA
- A leak is in progress on the Instrument Air system that is causing pressure to lower
- The crew enters AOP-017, Loss of Instrument Air
- No operator actions have been taken

As Instrument Air pressure continues to lower, which ONE of the following describes the order that Air Compressors will start and what determines this start order?

	Compressor Start Order	Reason
A.	B then C	CAS Sequence 1 controlling
B.	B then C	Local Pressure switches controlling
C.	C then B	CAS Sequence 1 controlling
D.✓	C then B	Local Pressure switches controlling

Plausibility and Answer Analysis

'A' A/C is supplied by bus 1A1 off of 1A-SA and has been lost until operator action is taken to restore, so 'A' A/C is left off the answers.

A Incorrect. The loss of Auxiliary Bus 1D will disable CAS. With CAS disabled compressors will start off of their local low pressure switches (see attachment 7 of AOP-017). Plausible since this would be correct if 1D had not lost power.

B Incorrect. Compressors will start off of their local pressure switches but those setpoints actually correlate to the start order of CAS Sequence 1. Plausible if candidate believes CAS Sequence 1 and Local Pressure switch start orders are the same.

C Incorrect. This is the order that Compressors will start, but CAS has lost power and compressors will start on internal pressure switches. Plausible since this is the start order if CAS was in sequence 3.

D Correct. The lost of Aux Bus 1D will cause CAS to lose power. With the loss of power auto starts will be generated according to the local low pressure switches. C will start at 101 psig, A at 96 psig, and B at 95 psig.

KA Statement - Loss of Instrument Air - Knowledge of the reasons for the following responses as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.5 / 41.10 / 45.6 / 45.13) Cross-over to backup air supplies

Importance Rating:	3.0 3.2
Technical Reference:	AOP-017 attachment 7 Rev. 28, page 57 OP-151.01 Rev. 53, page 8
References to be provided:	None
Learning Objective:	ISA, Obj. 9
Question Origin:	NEW
Comments:	(K/A Match) The plant does not have backup air supplies other than the backup/standby compressors. This question is written to the auto start function of the backup air compressors.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-017
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	065AK3.04	K/A 2:	

Attachment 7
Sheet 1 of 1

Significant Instrument Air Pressure Setpoints

NOTE

- Each compressor start listed below applies only when the applicable compressor is isolated from the CAS Control Panel Sequencer, or when power has been lost to the CAS Panel. If connected to the CAS Panel with the panel operating normally, the lead compressor will load at 98 psig and unload at 114 psig.
- For the compressor starts listed below, MCB pressure indication may be 7 psig below the given setpoint before action occurs, due to air dryer ΔP.
- Normal air receiver pressure is between 100 and 115 psig.

101 psig	--	Air Compressor 1C starts (if in STANDBY and isolated from CAS Panel)
96 psig	--	Air Compressor 1A starts (if in STANDBY and isolated from CAS Panel)
95 psig	--	Air Compressor 1B starts (if in STANDBY and isolated from CAS Panel)
90 psig	--	1SA-506 shuts to isolate Service Air from Instrument Air. (All stored air is available to Instrument Air System)
85 psig	--	RCS letdown flowpath valves begin to fail to mid-position
75 psig	--	Low air header pressure alarms on MCB for Instrument and Service Air.
60 psig	--	FW Flow Control valves auto shut (1FW-133, 1FW-191, 1FW-249)
35 psig	--	Remaining air-operated valves no longer considered reliable

--END OF ATTACHMENT 7--

4.0 PRECAUTIONS AND LIMITATIONS (continued)

7. If the Valve Shift failure is present on Air Dryer 1C-NNS, then Air Dryers 1A-NNS and 1B-NNS should be placed in service and the CAS Controller Sequence changed to Sequence 1or Sequence 2 and Air Dryer 1C-NNS removed from service until corrective actions may be performed. If a valve shift failure is present on Air Dryer 1A-NNS or 1B-NNS, then Air Dryers 1C-NNS should be placed in service and the CAS Controller Sequence changed to Sequence 3 with Air Dryer 1A-NNS and 1B-NNS removed from service until corrective actions may be performed.

8. If communication is lost between the CAS Control Panel and any Air Compressor, that Air Compressor will be prevented from operating by the CAS Control Panel (although local control may allow use of the compressor). For example, if Bus 1A1 was lost, then Air Compressor 1A-NNS would not be started by the CAS Control Panel even when Bus 1A1 is restored until manual actions are taken to integrate Air Compressor 1A-NNS back into the sequence at the CAS Control Panel.

9. If Power is lost to 1D344-11 feeding the CAS Control Panel, each Air Compressor will revert to it's local controller with Air Compressors 1A-NNS and 1B-NNS sensing their internal pressure switches and Air Compressor 1C-NNS sensing Service Air Receiver pressure.

10. Normally, when CAS Control Panel Sequence 1 or Sequence 2 is selected, Air Dryers 1A-NNS and 1B-NNS are in service with Air Dryer 1C-NNS OFF and bypassed. When CAS Sequencer program 3 is selected, Air Dryer 1C-NNS is in service with Air Dryers 1A-NNS and 1B-NNS OFF and bypassed.

11. If purge flow from an air dryer is secured, the dryer must be turned off to prevent tower overheating.

12. 1IA-650, IA to SA Cross-tie, shall remain open to allow Service Air to support the Instrument Air demands.

13. Air Compressor Air Dryers 1A-NNS and 1B-NNS have two modes of operation. The two modes control what signal causes the dyers to swap from drying to the regeneration cycle. Normally, the moisture control mode is used to shift the towers. Once the tower is placed in service and the dew point reaches the set-point, the towers will swap to the regeneration cycle and the other tower will be placed in service. The other mode of operation is called the timed mode. In this mode the tower will swap to regeneration based on the amount of time it has been in service. This mode may be used when operating conditions consist of high load (constant flow rates greater than 1300 SCFM) or shifting failures have occurred. The minimum time for regeneration takes 5 hours which is the time that this mode is currently set at. Engineering should be contacted when a dryer mode change is required."

14. 2009A NRC RO 014

Given the following plant conditions:
- The plant is operating at 50% power
- 'A' Train equipment is in service except for Charging
- 'B' CSIP is in service for routine testing, 'A' CSIP is in standby
- The crew has entered AOP-028, Grid Instability, due to switchyard voltage oscillations

(Breaker 105 SA, Emergency Bus A-SA to Aux Bus D Tie)
(Breaker 125 SB, Emergency Bus B-SB to Aux Bus E Tie)

Which ONE of the following lists the action(s) required to be taken FIRST to transfer the 6.9kV emergency buses in accordance with AOP-028?

A. Start the A-SA EDG, parallel with Offsite Power to power the 1A-SA Safety Bus, then open breaker 105

B. Start the B-SB EDG, parallel with Offsite Power to power the 1B-SB Safety Bus, then open breaker 125

C.✓ Open breaker 105 to initiate automatic starting and loading of the A-SA EDG

D. Open breaker 125 to initiate automatic starting and loading of the B-SB EDG

Plausibility and Answer Analysis

A *Incorrect. Plausible since 'A' Safety Train equipment is in service and this would be the normal method for a normal transfer of buses to the EDG, but the procedure specifically prohibits this action when voltages are degraded due to possible damage that could occur to safety equipment.*

B *Incorrect. Plausible since 'B' CSIP is in service and this is a priority in AOP-028 and this would be the normal method for a normal transfer of buses to the EDG, but the procedure specifically prohibits this action when voltages are degraded due to possible damage that could occur to safety equipment.*

C *Correct. Breaker 105 will be opened first because 'B' CSIP is in service and it will be performed first so that RCP Seal Injection Flow is maintained.*

D *Incorrect. Breaker 125 will not be opened but not first because 'B' CSIP is in service. Plausible since 'A' Safety Train is in service.*

KA Statement - Generator Voltage and Electric Grid Disturbances - Ability to operate and / or monitor the following as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.7 / 45.5 / 45.6) Engineered Safety Features

Importance Rating:	3.9 4.0
Technical Reference:	AOP-028-BD Rev. 9, pages 10 and 14
References to be provided:	None
Learning Objective:	LP-AOP-3.28, Obj. 4
Question Origin:	NEW
Comments:	(K/A Match) Individual must determine which safety bus (which power ESF Equipment) to transfer first as directed in AOP-028, Grid Instability.

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AOP-028
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	077AA1.05	K/A 2:	

Section 3.0—Operator Actions

Step	Description
10	**I: Check any EDG operating paralleled to Grid.**
	RNO: Go to Step 12.
	If an EDG is operating paralleled to grid when grid instability occurs, the EDG could be subjected to conditions which could result in the inoperability of the EDG.
11	**I: Open the associated EDG output breaker**
	This step protects the EDG from tripping due to the grid instability and provides the associated Emergency Bus with a stable power source.
12	**I: Check all of the following parameters within the limits of the indicated range:** • **6.9 kV buses - 6550 to 7250 volts** • **Frequency - 59.5 to 60.5 Hz**
	RNO: Energize the Emergency Buses with the associated EDG per Attachment 2. [F.1]
	Operation outside of these conditions is severe enough to cause degradation of vital plant equipment. The RNO step will power the emergency buses from the EDGs to ensure proper voltage and frequency are supplied to plant equipment to ensure damage to equipment is minimized while grid voltage and/or frequency is outside required limits. This step partially satisfies SOER 99-1 Addendum 1 Recommendation 2A.
N13	**I: Over-excitation may have caused voltage regulator to trip to manual.**
	The voltage regulator normally operates in auto (ON). Generator output voltage is sensed and fed back to the voltage regulator which compares this to the reference and adjusts excitation as needed to maintain generator output. The voltage regulator will trip to manual if excitation limit is exceeded.

Attachment 2— Energizing Emergency Buses From EDGs

Step	Description
1	**I: Determine which Emergency Bus is supplying power to the operating CSIP.** The intent is not to interrupt power to the CSIP to ensure that RCP seal injection flow is not interrupted. The first emergency bus to be transferred to the EDG will be the one that is not powering the operating CSIP.
N2	• **MDAFW FCVs will get an auto open signal (unless an AFW isolation signal is present) when either breaker 105 or 125 opens.** • **On a loss of power to an emergency bus the associated steam supply valve to the Turbine Driven AFW Pump will open.** • **This step will cause CVIS isolation and render both Containment Vacuum Reliefs inoperable (Tech Spec 3.0.3).** This note serves to remind operator of automatic operations that will occur when power to the emergency bus is interrupted in the following step. CVIS isolation renders both Containment Vacuum Reliefs inoperable and results in Tech Spec 3.0.3 entry.
2	**I: Open supply breaker to the emergency bus NOT supplying power to the operating CSIP:** Opening the supply breaker will cause automatic load shedding of the bus and starting of the EDG to power the bus. The CSIP powered from the affected emergency bus will sequence on by the load sequencer after the EDG is started.
3	**I: Verify the associated EDG starts and energizes the associated Emergency Bus.** The EDGs are aligned for auto startup when normal power to the emergency buses is interrupted and should do so without any operator actions regarding verifying prerequisites.
4	**I: Verify proper load sequencing for the affected emergency bus per OMM-004, Post Trip/Safeguards Review, Attachment 12.** Because the Emergency bus is de-energized to start the EDG and load the bus, this step checks for proper load sequencing.

15. 2009A NRC RO 015

Given the following plant conditions:

- Reactor Trip and Safety Injection have occurred from 100% power
- PZR level is off scale low
- Containment Radiation Monitors are at normal values
- Auxiliary Building Radiation Monitors are in alarm
- The crew is implementing EPP-013, LOCA Outside Containment
- The leak has been isolated by shutting 1SI-340, Low Head SI Train 'A' to Cold Leg Valve

Which ONE of the following describes the parameter used to determine that the break was isolated and subsequent operation of the RHR Pumps as a result of these actions in accordance with EPP-013?

	Parameter Used	RHR Pump Operation
A.✓	RCS Pressure Increasing	Stop 'A' RHR Pump
B.	RCS Pressure Increasing	Stop 'A' and 'B' RHR Pumps
C.	RCS Subcooling Increasing	Stop 'A' RHR Pump
D.	RCS Subcooling Increasing	Stop 'A' and 'B' RHR Pumps

Plausibility and Answer Analysis

A *Correct. EPP-013 directs use of RCS Pressure. Stopping 'A' RHR Pump is correct as its flow path is isolated to support break isolation.*

B *Incorrect. EPP-013 directs use of RCS Pressure. Stopping 'A' and 'B' RHR Pump is incorrect. Only 'A' RHR Pump is secured because its flow path is isolated to support break isolation. Other procedures direct stopping 'A' and 'B' RHR Pump if RCS Pressure is greater than 230 psig, but the candidate is not given this information and must evaluate why EPP-013 directs stopping 'A' RHR.*

C *Incorrect. EPP-013 directs use of RCS Pressure. RCS Subcooling could be an indication of RCS Pressure increasing if RCS Temperature is stable but the candidate is not given a temperature trend and EPP-013 directs use of RCS Pressure. Stopping 'A' RHR Pump is correct as its flow path is isolated to support break isolation.*

D *Incorrect. EPP-013 directs use of RCS Pressure. RCS Subcooling could be an indication of RCS Pressure increasing if RCS Temperature is stable but the candidate is not given a temperature trend and EPP-013 directs use of RCS Pressure. Stopping 'A' and 'B' RHR Pump is incorrect. Only 'A' RHR Pump is secured because its flow path is isolated to support break isolation. Other procedures direct stopping 'A' and 'B' RHR Pump if RCS Pressure is greater than 230# but the candidate is not given this information and must evaluate why EPP-013 directs stopping 'A' RHR.*

KA Statement - LOCA Outside Containment - Knowledge of the interrelations between (EMERGENCY PLANT EVOLUTION) and the following: (CFR: 41.7 / 45.7 / 45.8) Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.

Importance Rating:	3.8 4.0
Technical Reference:	EPP-013 Rev 8, page 5
References to be provided:	None
Learning Objective:	LP-EOP-3.3, Obj. 2d
Question Origin:	New
Comments:	(K/A Match) Operation of RHR/Decay Heat Removal during EPP-013.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	EPP-013
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE04EK2.2	K/A 2:	

Instructions	Response Not Obtained
5. Identify AND Isolate Break In RHR System.	
a. Verify low head SI to hot leg crossover valves - SHUT: 1SI-326 1SI-327	
b. Check RCS pressure - INCREASING	b. GO TO Step 5d.
c. GO TO Step 7.	
d. Check RHR train A:	
1) Shut low head SI train A to cold leg valve: 1SI-340	
2) Check RCS pressure - INCREASING	2) Open 1SI-340. GO TO Step 5e.
3) Stop RHR pump A.	
4) Shut RWST to RHR pump A suction valve: 1SI-322	
5) GO TO Step 7.	
e. Check RHR train B status:	
1) Shut low head SI Train B to cold leg valve: 1SI-341	
2) Check RCS pressure - INCREASING	2) Open the following valves: 1SI-341 1SI-326 1SI-327 GO TO Step 6.
3) Stop RHR pump B.	
4) Shut RWST to RHR pump B suction valve: 1SI-323	
5) GO TO Step 7.	

16. 2009A NRC RO 016

Given the following plant conditions:
- A Reactor Trip and Safety Injection have occurred
- AFW flow cannot be established
- All SG NR levels are off-scale low
- The crew enters FRP-H.1, Response to Loss of Secondary Heat Sink, from PATH-1
- RCS Pressure is 175 psig and stable
- Intact SG pressures are 475 psig and trending down slowly

Which ONE of the following describes the Secondary Heat Sink requirements in this condition and the required action?

	Steam Generators are	Required Action
A.	required to provide secondary heat sink	Return to PATH-1
B.	required to provide secondary heat sink	Remain in FRP-H.1
C.✓	NOT required to provide secondary heat sink	Return to PATH-1
D.	NOT required to provide secondary heat sink	Remain in FRP-H.1

Plausibility and Answer Analysis

A *Incorrect. In most cases, the Steam Generators will be needed as a heat sink. The case of a Large Break LOCA is unique though. Steam Generators are not needed as a Secondary Heat Sink. FRP-H.1 checks if Secondary heat sink is required; if Steam Generators are at a higher pressure than the RCS, they are not required and act as a heat source. A return to the procedure in effect is directed.*

B *Incorrect. In most cases, the Steam Generators will be needed as a heat sink. The case of a Large Break LOCA is unique though. Steam Generators are not needed as a Secondary Heat Sink. FRP-H.1 checks if Secondary heat sink is required; if Steam Generators are at a higher pressure than the RCS, they are not required and act as a heat source. Remaining in FRP-H.1 in not appropriate and a return to the procedure in effect is directed.*

C *Correct. Steam Generators are not needed as a Secondary Heat Sink. FRP-H.1 checks if Secondary heat sink is required; if Steam Generators are at a higher pressure than the RCS, they are not required and act as a heat source. A return to the procedure in effect is directed.*

D *Incorrect. Steam Generators are not needed as a Secondary Heat Sink. FRP-H.1 checks if Secondary heat sink is required; if Steam Generators are at a higher pressure than the RCS, they are not required and act as a heat source. Remaining in FRP-H.1 in not appropriate and a return to the procedure in effect is directed.*

KA Statement - Inadequate Heat Transfer - Loss of Secondary Heat Sink - Knowledge of the reasons for the following responses as they apply to (EMERGENCY PLANT EVOLUTION): (CFR: 41.5 / 41.10 / 45.6 / 45.13) Normal, abnormal and emergency operating procedures associated with (Loss of Secondary Heat Sink).

Importance Rating:	3.7 4.1
Technical Reference:	FRP-H.1 Rev. 23, page 4
References to be provided:	None
Learning Objective:	LP-EOP-3.11, Obj. 4e
Question Origin:	BANK OIT DEVELOPMENT Bank E05 G2.1.20
Comments:	

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	FRP-H.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE05EK3.2	K/A 2:	

Instructions	Response Not Obtained
2. Check Secondary Heat Sink Requirements:	
a. RCS pressure - GREATER THAN ANY NON-FAULTED SG PRESSURE	a. RETURN TO procedure and step in effect.
b. RCS temperature - GREATER THAN 350°F [330°F]	b. Place RHR system in service using GP-007, "NORMAL PLANT COOLDOWN" AND OP-111, "RESIDUAL HEAT REMOVAL SYSTEM", Section 5.1 while continuing with this procedure. <u>IF</u> RHR cooling is subsequently established, <u>THEN</u> RETURN TO procedure and step in effect. Observe <u>NOTE</u> prior to Step 3 AND GO TO Step 3.
c. Stop any running RHR pumps.	

17. 2009A NRC RO 017

Given the following plant conditions:
- A LOCA has occurred
- Multiple failures have resulted in a transition to EPP-012, Loss of Emergency Coolant Recirculation
- RWST level is 31%
- CNMT pressure is 12 psig

Which ONE of the following describes how plant operation is affected when the SI Suction Auto Switchover is reset in accordance with EPP-012?

A. Allows securing one CSIP and realigning CSIP flow to the Charging Line

B. Allows securing CNMT Spray Pumps and realigning CNMT Spray Valves

C. Prevents RHR Pump Miniflow Isolation Valves cycling open/shut automatically

D.✓ Prevents CSIP Alternate Miniflow Isolation Valves cycling open/shut automatically

Plausibility and Answer Analysis

SI reset is performed independently. CS discharge valves may be operated after resetting Phase B and Spray. RHR miniflow valves are not part of this circuit.

A Incorrect. SI will be terminated in EPP-012 to minimize RWST depletion when criteria are met but SI must be reset to allow this alignment not the SI Auto Suction Switchover reset.

B Incorrect. With present RWST level and CNMT pressure, CNMT Spray would be secured and realigned in EPP-012 to minimize RWST depletion but CNMT Spray must be reset to allow this alignment not the SI Auto Suction Switchover reset.

C Incorrect. The RHR Pump Miniflow Isolation valves provide pump protection by cycling automatically based on pump flow but these particular valves are not interlocked with the RWST-S Signal as the CSIP Alternate Miniflow valves are.

D Correct. The CSIP Alternate Miniflow Isolation Valves cycle automatically based on RCS Wide Range Pressure when an RWST-S Signal is present. Resetting the RWST-S Signal defeats this cycling.

KA Statement - Loss of Emergency Coolant Recirc. - Knowledge of the interrelations between (EMERGENCY PLANT EVOLUTION) and the following: (CFR: 41.7 / 45.7 / 45.8) Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Importance Rating:	3.6 3.9
Technical Reference:	EPP-012 Rev. 20, Note prior to step 4
References to be provided:	None
Learning Objective:	SIS, Obj. 8
Question Origin:	BANK OIT Development Bank E11 EK2.1 001
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	3	Reference:	EPP-O12
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE11EK2.1	K/A 2:	

Instructions	Response Not Obtained
NOTE: Foldout applies.	
1. Restore Emergency Coolant Recirculation Equipment.	
2. Reset SI.	IF any train of SI will NOT reset at MCB, THEN reset at SSPS using PATH-1 GUIDE, Attachment 12.
3. Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Refer to PATH-1 GUIDE, Attachment 2.)	
NOTE: Resetting the SI suction auto switchover signal also defeats the automatic open and shut signals to the CSIP alternate miniflow isolation valves.	
4. Reset SI Suction Auto Switchover.	
5. Add Makeup To RWST Using OP-107.01, "CVCS BORATION, DILUTION, AND CHEMISTRY CONTROL", Section 8.4.	Consult plant operations staff for alternate makeup sources.

18. 2009A NRC RO 018

Given the following plant conditions:

- The crew is performing EPP-015, Uncontrolled Depressurization of All Steam Generators
- Containment pressure is 4.1 psig and rising
- 'A' SG level is 10% and lowering
- 'B' SG level is 52% and stable
- 'C' SG level is 30% and stable
- RCS Hot Leg temperatures are lowering
- RCS Cooldown Rate is 90°F/hr

Which ONE of the following describes the MINIMUM required AFW flow rate for these conditions?

A. 12.5 KPPH to each SG

B.✓ 12.5 KPPH to 'A' and 'C' SGs, no minimum required to 'B' SG

C. 12.5 KPPH to 'A' SG, no minimum required to 'B' and 'C' SGs

D. No minimum required to any individual SG, minimum of 210 KPPH total AFW to all SGs is required

Plausibility and Answer Analysis

EOP-015 Foldout - Minimum Feed Flow
IF level in any SG is less than 25% [40%]. THEN maintain a minimum of 12.5 KPPH feed flow to that SG.

A *Incorrect. EPP-015 step 3.b says to control feed flow to maintain level less than 50%. 'B' SG is greater than 50% so a minimum feed flow is not required to the 'B' SG. Plausible though as a Foldout exists for minimum feed flow but 'B' SG does not meet this requirement either.*

B *Correct. EPP-015 step 3.b says to control feed flow to maintain level less than 50%. 'B' SG is greater than 50% so a minimum feed flow is not required to the 'B' SG. The Minimum Feed Flow Foldout exists but 'B' SG does not meet this requirement either. 'A' and 'C' SGs do meet the Minimum Feed Flow requirement so 12.5 KPPH is required to them.*

C *Incorrect. EPP-015 step 3.b says to control feed flow to maintain level less than 50%. 'B' SG is greater than 50% so a minimum feed flow is not required to the 'B' SG. The Minimum Feed Flow Foldout exists but B SG does not meet this requirement either. 'A' and 'C' SGs do meet the Minimum Feed Flow so 12.5 KPPH is required to them. This is plausible though because if CNMT was not adverse 'C' SG level would not meet the Foldout.*

D *Incorrect. Plausible since normally in the EOP network there are no minimums to each SG only a general requirement to maintain >210 KPPH total or NR level > 25% [40%] EPP-015 is an exception to this requirement.*

KA Statement - Steam Line Rupture - Excessive Heat Transfer - Knowledge of the operational implications of the following concepts as they apply to the EMERGENCY PLANT EVOLUTION): (CFR: 41.8 to 41.10 / 45.3) Normal, abnormal and emergency operating procedures associated with (Uncontrolled Depressurization of all Steam Generators).

Importance Rating:	3.5 3.8
Technical Reference:	EPP-015 Rev. 19, pages 7 and 8
References to be provided:	None
Learning Objective:	LP-EOP-3.9, Obj. 9b
Question Origin:	Modified BANK OIT Exam Bank EOP-3.9(08) 003
Comments:	

Origin:	MODIFIED	Cog Level:	H
Difficulty:	4	Reference:	EPP-015
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE12EK1.2	K/A 2:	

FOLDOUT

- <u>SI REINITIATION CRITERIA</u>

  <u>IF</u> any of the following occurs:

  - RCS subcooling - LESS THAN 10°F [40°F]
20°F [50°F]
  - PRZ Level - CAN <u>NOT</u> BE MAINTAINED GREATER THAN 10% [30%]

  <u>THEN</u> perform the following:

  a. <u>IF</u> CSIP suction aligned to VCT, <u>THEN</u> realign to RWST.

  b. Shut charging line isolation valves AND open BIT valves.

  c. Verify normal miniflow isolation valves - SHUT

  d. <u>IF</u> necessary to restore conditions, <u>THEN</u> restart standby CSIP.

- <u>EPP-014 TRANSITION CRITERIA</u>

  <u>IF</u> any SG pressure increases at any time, <u>THEN</u> GO TO EPP-014, "FAULTED STEAM GENERATOR ISOLATION", Step 1.

- <u>PATH-2 TRANSITION CRITERIA</u>

  <u>IF</u> any SG level increases in an uncontrolled manner <u>OR</u> any SG has abnormal radiation levels, <u>THEN</u> GO TO PATH-2, entry point J.

- <u>COLD LEG RECIRCULATION SWITCHOVER CRITERIA</u>

  <u>IF</u> RWST level decreases to less than 23.4% (2/4 Low-Low alarm), <u>THEN</u> GO TO EPP-010, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.

- <u>AFW SUPPLY SWITCHOVER CRITERIA</u>

  <u>IF</u> CST level decreases to less than 10%, <u>THEN</u> switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

- <u>RHR RESTART CRITERIA</u>

  <u>IF</u> RCS pressure decreases to less than 230 PSIG in an uncontrolled manner, <u>THEN</u> restart RHR pumps to supply water to the RCS.

- <u>MINIMUM FEED FLOW</u>

  <u>IF</u> level in any SG is less than 25% [40%], <u>THEN</u> maintain a minimum of 12.5 KPPH feed flow to that SG.

Instructions	Response Not Obtained
NOTE: As SG pressure and steam flow decrease, RCS hot leg temperatures will eventually stabilize and may increase. Adjusting feed flow and steam dump will control RCS hot leg temperatures.	
3. Control RCS Temperature:	
a. Check RCS cooldown rate - LESS THAN 100°F/HR	a. Decrease feed flow to 12.5 KPPH to each SG. GO TO Step 3c.
b. Check all SG levels - LESS THAN 50%	b. Control feed flow to maintain levels less than 50% in all SGs.
c. Check RCS hot leg temperatures - STABLE <u>OR</u> DECREASING	c. Control feed flow OR steam dump to stabilize RCS hot leg temperatures.
4. Maintain RCP Seal Injection Flow Between 8 GPM And 13 GPM.	
5. Check RCP Trip Criteria:	
a. Check RCPs - AT LEAST ONE RUNNING	a. GO TO Step 6.
b. Check all of the following: o SI flow - GREATER THAN 200 GPM o Check RCS pressure - LESS THAN 1400 PSIG o Check RCS hot leg temperatures - STABLE <u>OR</u> INCREASING	b. GO TO Step 6.
c. Stop all RCPs.	

QUESTIONS REPORT
for OIT Exam Bank

1. EOP-3.9 (08) 003

Given the following plant conditions:
- EPP-015, Uncontrolled Depressurization of All Steam Generators, is being implemented
- Containment pressure is normal
- 'A' SG level is 10%
- 'B' SG level is 40%
- 'C' SG level is 30%
- RCS Hot Leg temperatures are lowering
- RCS Cooldown Rate is 110°F/Hr

Which ONE of the following is correct?

A.✔ Feed flow to all 3 SGs should be decreased to 12.5 KPPH each.

B. Feed flow to all 3 SGs should be maintained as is and controlled to maintain less than 50% NR.

C. Feed flow to 'A' & 'C' SGs should be decreased to 12.5 KPPH.
Feed flow to 'B' should be maintained as is.

D. Feed flow to 'A' SG should be increased to return level to greater than 25%.
Feed flow to 'B' & 'C' should be maintained as is.

05SESS03
07PROC07

Program:	RS	Cog Level:	H
Difficulty:	3	Reference:	EPP-015
Ref. Provided?:	N	Key Words:	
K/A 1:	E12EK1.2	K/A 2:	

19. 2009A NRC RO 019

Given the following plant conditions:
- The plant is operating at 75% power
- Rod Control is in Manual for NI Gain Adjustment

The following conditions are observed:
- Tavg is increasing
- Tref remains constant
- Pressurizer pressure and level are increasing
- Control Rods are stepping

Which ONE of the following describes the event in progress and the required operator action?

	Event	Required Operator Action
A.	Inadvertant dilution	Manually trip the Reactor
B.	Inadvertant dilution	Perform a Rapid Addition of Boric Acid to the RCS
C.✓	Continuous rod withdrawal	Manually trip the Reactor
D.	Continuous rod withdrawal	Perform a Rapid Addition of Boric Acid to the RCS

Plausibility and Answer Analysis

A *Incorrect. RCS Temperature/Pressurizer response are correct for an inadvertant dilution but an inadvertant dilution would not cause rods to step while rods are in Manual. OMM-001 would direct a manual reactor trip if RPS setpoints are challenged but none are presently given.*

B *Incorrect. RCS Temperature/Pressurizer response are correct for an inadvertant dilution but an inadvertant dilution would not cause rods to step while rods are in Manual. AOP-003 would direct* a Rapid Addition of Boric Acid to the RCS *for an over dilution.*

C *Correct. A continuous rod withdrawal event is in progress with rods stepping in Manual and AOP-001 RNO directs trip of the reactor.*

D *Incorrect. A continuous rod withdrawal event is in progress with rods stepping in Manual but a Rapid Addition of Boric Acid to the RCS will only offset the reactivity effects and not correct the situation.*

KA Statement - Continuous Rod Withdrawal - Ability to operate and / or monitor the following as they apply to (ABNORMAL PLANT EVOLUTION): (CFR: 41.7 / 45.5 / 45.6) Reactor trip switches

Importance Rating:	4.3 4.2
Technical Reference:	AOP-001 Rev. 32, page 4
References to be provided:	None
Learning Objective:	RODCS, Obj. 4d
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	2	Reference:	AOP-001
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	001AA1.05	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

Steps 1 through 3 are immediate actions.

☐1. **CHECK** that LESS THAN TWO control rods are dropped.	☐1. **TRIP** the Reactor **AND GO TO** EOP Path-1.
☐2. **POSITION** Rod Bank Selector Switch to MAN.	
☐3. **CHECK** Control Bank motion STOPPED.	☐3. **TRIP** the Reactor **AND GO TO** EOP Path-1.

NOTE

Throughout this procedure, "Westinghouse Rod Control System Troubleshooting Guidelines" refers to Section 6.0 of EPRI document TR-108152, Rod Control System Maintenance – Westinghouse PWRs.

4. **GO TO** the appropriate section:

☐ • Section 3.1, Dropped Control Rod

☐ • Section 3.2, Continuous Spurious Control Bank Motion

☐ • Section 3.3, Failure of a Control Bank To Move

☐ • Section 3.4, Misaligned Control Rod (One or more rods misaligned from associated Group)

☐ • Section 3.5, Misaligned Control Rod Group (Group misaligned from Bank by more than one step)

☐ • Section 3.6, Malfunctioning Rod Position Indicator

--END OF SECTION 3.0--

20. 2009A NRC RO 020

Given the following plant conditions:
- Control bank 'D' group 2 rod F-10 dropped partially into the core due to a blown movable gripper fuse
- The condition has been repaired and the fuse replaced
- Actions per AOP-001, Malfunction of Rod Control and Indication System, are in progress to recover the misaligned rod
- The lift coil disconnect switches for all control bank 'D' rods except for F-10 are open
- ALB-013-7-1, Rod Control Urgent Alarm, is received when the operator starts to withdraw rod F-10

Which ONE of the following describes the Rod Control Power Cabinet that is producing the Rod Control Urgent Alarm and the cause of this alarm?

	Power Cabinet	Cause
A.	1BD	Phase Failure
B.✓	1BD	Regulation Failure
C.	2BD	Phase Failure
D.	2BD	Regulation Failure

Plausibility and Answer Analysis

AOP-001 alerts the operator that the Rod Control Urgent Alarm will occur.

A *Incorrect. Even though the misaligned rod is in Group 2, all lift coil disconnects in Group 1 will also be opened. This will result in NO rod motion in the Group 1 rods which yields a regulation failure in the Group 1 Power Cabinet that produces the Rod Control Urgent Alarm. Phase failure is plausible since it is produced on a loss of power to a phase and none of the lift coils are energizing. However, this is a regulation failure.*

B *Correct. The misaligned rod is in Group 2 so all lift coil disconnects in Group 1 and all but one of Group 2 will be opened. This will result in NO rod motion in the Group 1 rods which yields a regulation failure in the Group 1 Power Cabinet that produces the Rod Control Urgent Alarm*

C *Incorrect. This is plausible since 3 of the 4 lift coil disconnects are open for the Group 2 Cabinet and one rod is moving in this cabinet when the other three are not but the regulation failure and alarm is only generated if all rods are effected. A phase failure is plausible though as a phase failure is produce due to a loss of power on a phase and three of the lift coils are energizing but this is a regulation failure.*

D *Incorrect. This is plausible since 3 of the 4 lift coil disconnects are open for the Group 2 Cabinet and one rod is moving in this cabinet when the other three are not but the regulation failure and alarm is only generated if all rods are effected.*

KA Statement - Inoperable/Stuck Control Rod - Knowledge of annunciators alarms, indications or response procedures

Importance Rating:	4.2 4.1
Technical Reference:	AOP-001 Rev. 32, page 23 APP-ALB-013 Rev. 28, window 7-1, page 28 RODCS Student Text
References to be provided:	None
Learning Objective:	RODCS, Obj. 7c
Question Origin:	BANK OIT EXAM BANK RODCS (07C) 1
Comments:	Changed Distractors C and D to aid in plausibility and formatted question.

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	AOP-001
~~Ref. Provided?:~~	~~N~~	~~Key Words:~~	~~2009A NRC RO~~
K/A 1:	005AG2.4.31	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.4 Misaligned Control Rod

NOTE

- When rod movement is started, ALB-13-7-1, ROD CONTROL URGENT ALARM will be received, and movement for the other group of rods in the affected bank will be locked out. ALB-13-8-5, COMPUTER ALARM ROD DEV/SEQ NIS PWR RANGE TILTS may be received during rod motion due to a rod-to-bank deviation or bank sequence error.
- Surveillance requirement 4.1.1.1.1.a requires performing a shutdown margin calculation upon detecting an inoperable control rod. **[C.1]**

☐ 21. **POSITION** the misaligned rod to the step counter position recorded in Step 17 **AND MONITOR** the following for the misaligned rod to verify movement:
- Step counter
- DRPI

RESPONSE NOT OBTAINED:

21. **IF** the misaligned rod can NOT be repositioned,
THEN:

☐ a. **REFER TO** Tech Spec Section 3.1.3.1, Movable Control Assemblies – Group Height.

b. **NOTIFY** the following:

☐ • Manager – Operations

☐ • Reactor Engineering

7-1

DEVICES: Rod Control System Circuitry

SETPOINT: Malfunction

ROD CONTROL URGENT ALARM

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. Digital Rod Position Indication
   b. Axial Flux Distribution
      (1) NI-41C
      (2) NI-42C
      (3) NI-43C
      (4) NI-44C

2. **VERIFY** Automatic Functions:
   a. Automatic and manual rod motion blocked for the affected group(s)
   b. Stationary grippers energized at reduced current
   c. Movable grippers for selected group energized at reduced current

3. **PERFORM** Corrective Actions:

CAUTION

If the Rod Bank Selector switch is selected to an individual bank, moving the switch to another position could potentially cause rods to drop.

   a. **IF** the Rod Bank Selector Switch is in AUTO,
      THEN POSITION the Rod Bank Selector Switch to any position other than AUTO
      AND DO NOT move rods.
   b. **IF** there is an indication of a control rod malfunction (MCB or AEP-1),
      THEN GO TO AOP-001, Malfunction of Rod Control and Indication System.
   c. **IF** the alarm is due to a logic error,
      THEN DO NOT RESET the alarm until the cause of the failure has been corrected.
   d. **IF** the alarm is due to a group being on DC Hold or otherwise inoperable,
      THEN REFER TO Technical Specification surveillance requirement 4.1.1.1.1a
      AND PERFORM Shutdown Margin as required. (Reference 2)

CAUSES:

1. Regulation failure (Power Cabinet)
2. Phase failure (Power Cabinet)
3. Logic error (Power Cabinet)
4. Multiplexing error (Power Cabinet)
5. Loose card (Power Cabinet or Logic Cabinet)
6. Slave cycler failure (Logic Cabinet)
7. Oscillator failure (Logic Cabinet)
8. Alarm circuit or instrument malfunction

REFERENCES:

1. Technical Specifications 3.1.1.1, 3.1.1.2, 3.1.3.1, 3.1.3.5, 3.1.3.6, 4.1.1.1.1a
2. CR 98-00340 (LER 98-003)
3. ~~AOP-001, Malfunction of Rod Control and Indication System~~
4. 6-B-401 Sheet 0046K, 0483
5. VM-PKO-V01, Rod Control System

21. 2009A NRC RO 021

Given the following plant conditions:
- 1CS-278, Emergency Boric Acid Addition is under clearance
- Emergency Boration is required

Which ONE of the following alignments is the preferred method resulting in the FASTEST power reduction in accordance with AOP-002, Emergency Boration?

A. OPEN 1CS-291, Suction From RWST LCV-115B then SHUT1CS-165, VCT Outlet LCV-115C

B. OPEN 1CS-283, Boric Acid To Boric Acid Blender FCV-113A and 1CS-155, Makeup To VCT FCV-114A

C.✓ OPEN 1CS-283, Boric Acid To Boric Acid Blender FCV-113A and 1CS-156, Makeup To CSIP Suction FCV-113B

D. OPEN 1CS-283, Boric Acid To Boric Acid Blender FCV-113A, 1CS-156, Makeup To CSIP Suction FCV-113B, and 1CS-155, Makeup To VCT FCV-114A

Plausibility and Answer Analysis

A *Incorrect. This is plausible because it is listed in AOP-002 as an alternate flow path, but is only performed if the previous two using a Boric Acid Transfer Pump were unsuccessful.*

B *Incorrect. This is plausible because it is listed in AOP-002 as an alternate flow path, but is only performed if the previous flowpaths using a Boric Acid Transfer Pump or the RWST were unsuccessful.*

C *Correct. This flow path is an alternate flow and is used if 1CS-278 (preferred) is not available.*

D *Incorrect. This is plausible because the system configuration is physically possible and it combines elements of two approved flowpaths, but it is not listed in AOP-002.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Emergency Boration - Knowledge of the operational implications of the following concepts as they apply to the (ABNORMAL PLANT EVOLUTION): (CFR: 41.8 to 41.10 / 45.3) Relationship between boron addition and reactor power

Importance Rating:	3.6 3.9
Technical Reference:	AOP-002 Rev. 23, page 4
References to be provided:	None
Learning Objective:	LP-AOP-3.2, Obj. 2
Question Origin:	BANK OIT Development Bank 024 AK1.02 001
Comments:	K/A match

Origin:	BANK	Cog Level:	F
Difficulty:	2	Reference:	AOP-002
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	024AK1.02	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

This procedure contains no immediate actions.

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐1. **VERIFY** a Boric Acid Pump RUNNING.	☐1. **GO TO** Step 6.
2. **ESTABLISH** boration flowpath using 1CS-278 as follows:	
☐ a. **OPEN** 1CS-278, Emergency Boric Acid Addition.	☐ a. **GO TO** Step 3.
☐ b. **VERIFY** at least 30 gpm boric acid flow to CSIP suction on FI-110.	☐ b. **GO TO** Step 3.
☐ c. **GO TO** Step 4.	
3. **ESTABLISH** boration flowpath using FCV-113A/B as follows:	
a. **OPEN** the following valves:	☐ a. **GO TO** Step 6.
☐ • 1CS-283, Boric Acid To Boric Acid Blender FCV-113A	
☐ • 1CS-156, Make Up To CSIP Suction FCV-113B	
☐ b. **VERIFY** at least 30 gpm boric acid flow to CSIP suction on recorder panel or ERFIS point FCS0113A.	☐ b. **GO TO** Step 6.
☐4. **VERIFY** and **MAINTAIN** at least 30 gpm charging flow to RCS (FI-122A.1) until required boration is completed.	
☐5. **GO TO** Step 8.	

22. 2009A NRC RO 022

Given the following plant conditions:
- The compensating voltage on Intermediate Range (IR) channel N-35 is set too low
- A plant shutdown is in progress in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 to Mode 3)

Which ONE of the following describes the effect on N-35 indication and how SR NIs will be operated when N-36 decreases to 4.5E-11 amps?

A. N-35 will indicate LOWER than N-36;
Both SR NIs will automatically energize

B. N-35 will indicate LOWER than N-36;
Both SR NIs must be manually energized

C. N-35 will indicate HIGHER than N-36;
Both SR NIs will automatically energize

D.✓ N-35 will indicate HIGHER than N-36;
Both SR NIs must be manually energized

Plausibility and Answer Analysis

A Incorrect. N35 would indicate lower than N36 if the channel was over compensated and therefore both SR would energize automatically as P-6 is a 2/2 coincidence.

B Incorrect. N35 would indicate lower than N36 if the channel was over compensated but both SR would energize automatically as P-6 is a 2/2 coincidence.

C Incorrect. With low compensating volts, N35 will indicate higher than N36 but both SR must be manually energized since P-6 is a 2/2 coincidence. (1/2 increasing)

D Correct. With low compensating volts, N35 will indicate higher than N36 and therefore both SR must be manually energized as P-6 is a 2/2 coincidence.

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Loss of Intermediate Range NI - Knowledge of the operational implications of the following concepts as they apply to the (ABNORMAL PLANT EVOLUTION): Effects of voltage changes on performance

Importance Rating:	2.7 3.0
Technical Reference:	OP-105 Rev 24, page 9 (Caution in section 7.1) NIS Student Text
References to be provided:	None
Learning Objective:	NIS, Obj. 8c
Question origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	OP-105
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	033AK1.01	K/A 2:	

REFERENCE USE

6.0 NORMAL OPERATION

NR-45 is normally selected to record the highest Power Range channel on one pen. The other pen may be selected to an Intermediate Range channel or a ΔF channel.

7.0 SHUTDOWN

7.1. Normal Plant Shutdown (MODE 1 to MODE 3)

7.1.1. Initial Conditions

1. The Plant is in MODE 1 and a shutdown is in progress. _____

7.1.2. Procedure Steps

CAUTION
Both Intermediate Range channels must be below the P-6 reset of 5 x 10^{-11} amps for Source Range high voltage to automatically re-energize. If one Intermediate Range channel fails to drop below the P-6 reset when core conditions indicate that the channel should have, the Source Range high voltage will have to be manually energized.

1. **WHEN** Intermediate Range Power is less than 5 x 10^{-11} amps on both channels,
THEN CHECK the following:

- At the Source Range Drawers, both Source Range Detectors are energized. _____
- ALB-13-2-3, SOURCE RANGE LOSS OF DETECTOR VOLTAGE, not lit. _____
- At the Bypass Permissive Light Panel, the following lights are de-energized:
  - SOURCE RANGE TRAIN A TRIP BLOCKED HI VOLT OFF (Window 1-2) _____
  - SOURCE RANGE TRAIN B TRIP BLOCKED HI VOLT OFF (Window 2-2) _____
  - IR PWR > P-6 SOURCE RANGE BLOCK PERMISSIVE (Window 1-5) _____

23. 2009A NRC RO 023

Given the following plant conditions:
- The plant is operating at 100% power
- Rad monitor REM-01TV-3534, Condenser Vacuum Pump is under clearance
- 'B' SG has a known 4 GPD tube leak
- AOP-016, Excessive Primary Plant Leakage, is in progress

Chemistry reports 'B' SG tube leakage has increased and provides the following data:

0200	48 gpd
0300	76 gpd
0400	77 gpd
0500	78 gpd
0600	85 gpd
0700	126 gpd
0800	158 gpd

Which ONE of the following describes the time that the plant must be in Mode 3 in accordance with AOP-016? (**Reference provided**)

A.✓ Today at 0900

B. Today at 1000

C. Today at 1400

D. Tomorrow at 0400

Plausibility and Answer Analysis

All 3 distractors have valid required action times IAW AOP-016 Attachment 1 Primary-To-Secondary Leakage. The operator must use the provided leakrate and determine the required action then determine which of the actions need to be performed first based on the time that the leakage was quantified.

A Correct. At 0300 tube leakage has exceeded 75 gpd with REM-01TV-3534 inoperable the required action is to be in Mode 3 in less than 6 hours (0300 + 6 hours = 0900).

B Incorrect. At 0700 the leak rate is ≥ 75 gpd with the rate of increase > 30 gpd/hr (actual is 41 gpd increase). The action is to reduce power to 50% within 1 hour and be in Mode 3 within the next 2 this is a total of 3 hours. (0700 + 3 hours = 1000). While the unit must be in Mode 3, there are more restrictive requirements.

C Incorrect. At 0800 the leakage exceeded 150 gpd requiring the unit to be placed in Mode 3 in less than 6 hours (0800 + 6 hours = 1400). While the unit must be in Mode 3, there are more restrictive requirements.

D Incorrect. At 0400 the leak rate is ≥ 75 gpd sustained for 1 hour and rate of increase is < 30 gpd/hr making the required action Be in Mode 3 within 24 hours. (0400 + 24 hours = 0400 the following day). While the unit must be in Mode 3, there are more restrictive requirements.

KA Statement - Steam Generator Tube Leak - Ability to determine and interpret the following as they apply to ABNORMAL PLANT EVOLUTION): (CFR: 41.10 / 43.5 / 45.13) Past history of leakage with current problem

Importance Rating:	2.8 3.3
Technical Reference:	AOP-016 Rev. 39, Page 19
References to be provided:	AOP-016 Rev. 39, Page 19
Learning Objective:	LP-AOP-3.16, Obj. 3
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-016
Ref. Provided?:	YES	Key Words:	2009A NRC RO
K/A 1:	037AA2.05	K/A 2:	

Attachment 1
Sheet 6 of 7

Primary-To-Secondary Leak

INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- For initial leakage reports, where no previous leakage existed, leakage should be assumed to have changed from zero to the current value in the last hour.
- The monitoring requirements of Step 3 become optional if Step 10 directs performance of Attachment 9, 10, or 11.

✱☐ 10. **PERFORM** the required actions based on the following: **[C.5, 7]**

Leak Rate (gpd) in any SG	+	Rate of Increase (gpd/hr) in any SG	=	Required Action
Leakage Detected				
less than 5	+	N/A	=	• Notify Chemistry to refer to CRC-804
Increased Monitoring				
5 to less than 30	+	N/A	=	• Perform Attachment 9
Action Level 1				
30 to less than 75	+	N/A	=	• Perform Attachment 10
Action Level 2				
Greater than or equal to 75 sustained for 1 hour	+	Less than 30	=	• Perform Attachment 11 • Be in Mode 3 within 24 hours
Action Level 3				
Greater than or equal to 75	+	Greater than or equal to 30	=	• Verify sustained rate of change above 30 gpd/hr (not followed by a decrease - spike) • Perform Attachment 11 • Reduce power to 50% within 1 hour • Be in Mode 3 within the next 2 hours (3 hours total time)
Greater than or equal to 75 **AND** LOSS of REM-01TV-3534, Condenser Vacuum Pump Rad Monitor (Grid 2)	+	N/A	=	• Perform Attachment 11 • Be in Mode 3 in less than 6 hours
Greater than or equal to 150	+	Less than 30	=	• Perform Attachment 11 • Be in Mode 3 in less than 6 hours • Be in Mode 5 within the next 30 hours (36 hours total)

24. 2009A NRC RO 024

Given the following plant conditions:

- The crew is performing FRP-C.1, Response To Inadequate Core Cooling
- Both CSIPs are running
- SI flow indication is 0 gpm
- 1SI-3 and 1SI-4, BIT Outlet Valves will not OPEN from the MCB
- The RO has been directed to establish another high head injection flow path

Which ONE of the following describes which valve will be operated preferentially to establish this flow path in accordance with FRP-C.1 and the actions required to supply control power to that valve?

A. 1SI-52, Alt High Head SI To Cold Leg;
Direct operators locally to shut control power breakers in the field

B.✓ 1SI-52, Alt High Head SI To Cold Leg;
Turn on valve control power at the Main Control Board

C. 1SI-86, High Head SI To Hot Leg;
Direct operators locally to shut control power breakers in the field

D. 1SI-86, High Head SI To Hot Leg;
Turn on valve control power at the Main Control Board

Plausibility and Answer Analysis

High Head Injection Valves are equipped with two control switches. One is Control Power and is normally OFF. The other is the valve itself and is normally in SHUT/PULL TO LOCK. FRP-C.1 pg 6 lists the order of priority.

A *Incorrect. This is the correct valve but incorrect manner of supplying control power. Plausible since many valves in this system and the EOP Network do require local operator action to supply control power to the MCB switches (SI Cold Leg Accumulators, CSIP Discharge and Suction Cross-connects, RHR Hot Leg Suctions) .*

B *Correct. 1SI-52 is listed first and this is action required to supply control power so the valve can be operated.*

C *Incorrect. This is plausible because 1SI-86 is listed second. Control power is incorrect though plausible since many valves in this system and the EOP Network do require local operator action to supply control power to the MCB switches (SI Cold Leg Accumulators, CSIP Discharge and Suction Cross-connects, RHR Hot Leg Suctions)*

D *Incorrect. This is plausible because 1SI-86 is listed second. Control is supplied to this valve by a switch at the MCB as well.*

KA Statement - Inad. Core Cooling - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: (CFR: 41.5 / 45.5) ECCS valve control switches and indicators

Importance Rating:	4.2 4.2
Technical Reference:	EOP-FRP-C.1 Rev. 12 step 2 RNO, page 6
References to be provided:	None
Learning Objective:	SIS, Obj 4a
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	FRP-C.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	074EA1.27	K/A 2:	

Instructions	Response Not Obtained
2. Verify SI Flow In All Trains:	
a. Check for all of the following: o SI flow - GREATER THAN 200 GPM o RHR HX Train A header flows - GREATER THAN 1000 GPM o RHR HX Train B header flows - GREATER THAN 1000 GPM	a. Perform the following: 1) Verify CSIPs - RUNNING 2) Verify RHR pumps - RUNNING 3) IF SI flow can NOT be established, THEN align valves to establish any other high head injection flowpath (listed in order of preference): (a) Alternate high head SI to cold legs (1SI-52 SA) (b) High head SI to hot legs (1SI-86 SB) (c) Alternate high head SI to hot legs (1SI-107 SA) (d) Charging flow path

25. 2009A NRC RO 025

Of the following, which are part of the basic strategy for AOP-032, High RCS Activity?

1. Initiate actions to minimize further fission product barrier challenges
2. Increase secondary sampling to identify Steam Generator Tube Leakage
3. Direct actions to ensure compliance with T.S. 3.4.8, RCS Specific Activity
4. Verify a CVCS Ion Exchanger is in service for RCS cleanup

A. 1 and 2

B. 2 and 3

C.✓ 3 and 4

D. 1 and 4

Plausibility and Answer Analysis

AOP-032-BD lists the basic strategies as:

- *Confirming the activity level.*
- *Making an initial announcement of potential high radiation levels, conducting area radiation surveys for personnel protection, and advising personnel of the survey results using the PA.*
- *Initiating the necessary actions under TS 3.4.8 and the site Emergency Plan. This may require shutting down the plant and cooling down to below 500°F.* ***(Number 3 in stem)***
- *Verifying that a CVCS ion exchanger is placed in service.* ***(Number 4 in stem)***
- *Initiating a shutdown if fuel fission product boundaries are breached or jeopardized.*

A: Incorrect. 1 and 2 are both incorrect. ***(1)*** *While operations always strives to preserve FPBs and with the fuel already challenged this becomes more important, these actions are not initiated by AOP-032.* ***(2)*** *Incorrect but plausible since the basis for the RCS Activity limits in Tech Specs is a SGTR and the cooldown to < 500°F is required because of this.*

B Incorrect. 3 is correct but 2 is incorrect. ***(2)*** *Incorrect but plausible since the basis for the RCS Activity limits in Tech Specs is a SGTR and the cooldown to < 500°F is required because of this.*

C Correct. Initiating the necessary actions under TS 3.4.8 and the site Emergency Plan. This may require shutting down the plant and cooling down to below 500°F. Verifying that a CVCS ion exchanger is placed in service is required for cleanup of the RCS.

D Incorrect. 4 is correct and 1 is incorrect. ***(1)*** *While operations always strives to preserve FPBs and with the fuel already challenged this becomes more important, these actions are not initiated by AOP-032.*

KA Statement - High Reactor Coolant Activity - Knowledge of the effect that a loss or malfunction of the (SYSTEM) will have on the following: (CFR: 41.7 / 45.6) Corrective actions as a result of high fission-product radioactivity level in the RCS

Importance Rating:	2.9 3.6
Technical Reference:	AOP-032-BD Rev. 3, page 3
References to be provided:	None
Learning Objective:	LP-AOP-3.32, Obj. 1
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	AOP-032
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	076AK3.05	K/A 2:	

FSAR Chapter 15 Accident Analyses for Steam Generator Tube Ruptures are based on a normal letdown purification flow rate of 105 gpm. Radiological dose consequence evaluations conservatively consider maximum letdown flow to be approximately 120 gpm. A Technical Specification limit for reactor coolant Dose Equivalent Iodine I-131 (DEI I-131) of 1.0 μCi/gm is established based on these analyses.

Previously, interim administrative limits were imposed on specific activity and letdown flowrate by ESR 0000412 and ESR 9800326 on Tech Spec Section 3.4.8 Action a and FSAR Sections 9.3.4 and 15.6.3.4 were amended. Subsequently, SGR/PUR ESR 9400001 provided a new source term analysis that replaced the conventional source term method and limits with new DEI-131 limits evaluated by Alternate Source Term calculations in accordance with Reg Guide 1.183. This resulted in cancellation of the interim administrative limits as implemented by Tech Spec Section 3.4.8 Amendment 107 and FSAR Sections 9.3.4 and 15.6.3.4 Amendment 51.

Basic AOP strategy for high RCS activity consists of the following:

- Confirming the activity level.
- Making an initial announcement of potential high radiation levels, conducting area radiation surveys for personnel protection, and advising personnel of the survey results using the PA.
- Initiating the necessary actions under TS 3.4.8 and the site Emergency Plan. This may require shutting down the plant and cooling down to below 500°F.
- Verifying that a CVCS ion exchanger is placed in service.
- Initiating a shutdown if fuel fission product boundaries are breached or jeopardized.

26. 2009A NRC RO 026

Given the following plant conditions:
- A LOCA has occurred
- RCS pressure is 1100 psig and stable
- Containment pressure is 5 psig and stable
- The crew is performing actions contained in EPP-009, Post LOCA Cooldown and Depressurization

Which ONE of the following describes the method that will be used to perform the cooldown of the RCS?

Perform the cooldown using. . .

A.✓ S/G PORVs at less than 100°F per hour.

B. S/G PORVs at the maximum achievable rate.

C. Condenser steam dumps at less than 100°F per hour.

D. Condenser steam dumps at maximum achievable rate.

Plausibility and Answer Analysis

A Correct. SG PORVs will be used for the cooldown because the condenser is not available and EPP-009 limits cooldown to 100°F/hour.

B Incorrect. SG PORVs will be used for the cooldown because the condenser is not available. The Cooldown Rate is incorrect. Other EOPs perform a max rate cooldown but EPP-009 limits cooldown to 100°F/hour.

C Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Rate is correct.

D Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Credible because it is the normal method of cooldown. The Cooldown Rate is incorrect. Other EOPs perform a max rate cooldown but EPP-009 limits cooldown to 100°F/hour.

KA Statement - LOCA Cooldown - Depress. - Knowledge of the interrelations between (EMERGENCY PLANT EVOLUTION) and the following: (CFR: 41.7 / 45.7 / 45.8) Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.

Importance Rating:	3.7 4.0
Technical Reference:	EPP-009 Rev 14, page 12
References to be provided:	None
Learning Objective:	LP-EOP-3.5, Obj. 5c
Question Origin:	BANK OIT Development 039 A2.01
Comments:	

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	EPP-009
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE03EK2.2	K/A 2:	

Instructions	Response Not Obtained
10. Initiate RCS Cooldown To Cold Shutdown:	
a. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR	
b. Check RHR system - OPERATING IN SHUTDOWN COOLING MODE	b. GO TO Step 10f.
c. Cooldown using RHR.	
d. Dump steam from intact SG(s) to supplement cooldown and cool SG(s).	
e. GO TO Step 11.	
f. Check all of the following to determine if steam can be dumped to condenser: o Check any intact SG MSIV - OPEN o Check condenser available (C-9) light (BPLB 3-3) - LIT o Steam dump control system - AVAILABLE	f. Dump steam from intact SGs at using any of the following (listed in order of preference): 1) SG PORVs 2) Locally operate SG PORVs using OP-126, "MAIN STEAM, EXTRACTION STEAM, AND STEAM DUMP SYSTEMS", Section 8.2. 3) TDAFW pump GO TO Step 11.
g. Transfer steam dump to steam pressure mode using OP-126, "MAIN STEAM, EXTRACTION STEAM AND STEAM DUMP SYSTEM", Section 5.3.	
h. Dump steam from intact SGs to condenser using condenser steam dumps.	

27. 2009A NRC RO 027

Given the following plant conditions:
- The crew is performing EPP-007, Natural circulation cooldown with steam void in vessel without RVLIS

Current plant conditions are:
- RCS pressure is 1600 psig
- All hot leg temperatures are 449°F and slowly lowering
- Pressurizer level is 30%
- Letdown flow is 60 GPM
- Total RCP Seal Return Flow is 8 GPM
- Total RCP Seal Injection Flow is 27 GPM
- An RCS depressurization to 800 psig is about to be performed

Which ONE of the following describes what charging flow should be adjusted to prior to starting the depressurization and the reason for making this adjustment?

	Charging Flow	Reason
A.	41 GPM	to accommodate void growth
B.✓	41 GPM	to allow accurate monitoring of void growth
C.	60 GPM	to accommodate void growth
D.	60 GPM	to allow accurate monitoring of void growth

Plausibility and Answer Analysis

A *Incorrect. With the present flows provided in the stem, charging flow should be set to 41 GPM. This will balance inventory but this is not performed to accommodate void growth although with inventory balanced, PZR Level will be stable in the required band to accommodate void growth.*

B *Correct. With the present flows provided in the stem, charging flow should be set to 41 GPM. This will balance inventory and allow PZR Level to be an accurate measurement of void growth.*

D *Incorrect. 60 GPM will balance charging and letdown which is the high level action of EPP-007 but subsequent steps direct balancing all flows. Balanced inventory will allow PZR Level to be an accurate measurement of void growth.*

C *Incorrect. 60 GPM will balance charging and letdown which is the high level action of EPP-007 but subsequent steps direct balancing all flows. Balanced inventory will maintain PZR Level stable in the required band to accommodate void growth but this is not performed to accommodate void growth, it is performed to allow PZR Level to be an accurate measurement of void growth .*

KA Statement - Natural Circ. - Knowledge of the reasons for the following responses as they apply to (EMERGENCY PLANT EVOLUTION): (CFR: 41.5 / 41.10 / 45.6 / 45.13) Normal, abnormal and emergency operating procedures associated with (Natural Circulation Operations).

Importance Rating:	3.2 3.6
Technical Reference:	EPP-007 Rev.11, page 16
References to be provided:	None
Learning Objective:	LP-EOP-3.8, Obj. 2c
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	EPP-007
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	WE09EK3.2	K/A 2:	

Instructions	Response Not Obtained

NOTE: Balancing inventory gains and losses during RCS depressurization allows potential void growth to be accurately monitored.

7. Equalize Charging AND Letdown Flows:

   a. Place charging flow controller in manual:

      FK-122.1

   b. Control charging AND seal injection flows to equal letdown AND seal return flows.

8. Maintain RCP Seal Injection Flow Between 8 GPM And 13 GPM.

28. 2009A NRC RO 028

Given the following plant conditions:

- The plant is operating in Mode 5
- A cooldown is in progress on RHR
- 'B' RCP is in service with 'A' and 'C' RCPs secured
- ALB-008-3-3, RCP B Seal #1 Leakoff High Low Flow Alarm, has just been received

The RO reports the following indications on 'B' RCP:

- #1 Seal Leakoff Flow	7.5 gpm
- Seal Injection Flow	10 gpm
- Seal Inlet Temperature	193°F and rising
- Pump Radial Bearing Temperature	155°F and stable
- #1 Seal Delta P	175 psid

Which ONE of the following is the response that should be taken for these conditions?

A. RCP 'B' should remain running

B.✓ Stop RCP 'B' immediately

C. Stop RCP 'B' within 10 minutes

D. Stop RCP 'B' within 8 hours

Plausibility and Answer Analysis

A *Incorrect. Plausible if the candidate does not recognize conditions of the RCP as the P&L of OP-100 states "Whenever RCS temperature is greater than 160°F, at least one RCP must be in operation" but seal D/P does not allow continued operation.*

B *Correct. GP-007 P&L #22, When the #1 RCP Seal differential pressure is below 200 psid or when the VCT pressure is below 15 psig, the RCP must not be operated. With the Reactor already shutdown the RCP should be immediately secured.*

C *Incorrect. Plausible as AOP-014 does have RCPs secured if CCW is expected to be lost for 10 minutes but seal D/P does not allow continued operation.*

D *Incorrect. Plausible as AOP-018 does have the RCP secured within 8 hours if the #1 seal is degraded but seal D/P does not allow continued operation.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Reactor Coolant Pump - Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) RCP seal differential pressure instrumentation

Importance Rating:	3.1 3.0
Technical Reference:	GP-007 Rev. 48, P&L #22, page 12 OP-100 Rev 28, P&L #20, page 7 AOP-014 Rev 31, page 19 AOP-018 Rev 34, page 15
References to be provided:	None
Learning Objective:	RCS, Obj 6a
Question Origin:	BANK LOR NRC B04 054
Comments:	

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	GP-007
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	003A4.04	K/A 2:	

4.0 PRECAUTIONS AND LIMITATIONS (continued)

21. Prior to reducing RCS temperature to less than 325°F, at least one train of RHR suction relief valves must be aligned to the RCS to insure that there is a flow path from the RCS to the RHR Suction Reliefs. (References 2.3.3 and 2.6.3)

22. When the #1 RCP Seal differential pressure is below 200 psid or when the VCT pressure is below 15 psig, the Reactor Coolant Pumps must not be operated.

R 23. Xenon free shutdown margin should be achieved prior to securing all RCPs. (References 2.6.13 and 2.7.11)

24. When making a determination of RCS temperature, use all available temperature detectors to prevent reaching saturation conditions in the RCS when reducing RCS pressure.

- Wide range T_{hot}
- Wide range T_{cold}
- Core thermocouples
- RHR HX inlet & outlet temperatures

R 25. If all RCPs have been stopped for more than 5 minutes, and the reactor coolant temperature is greater than the charging and seal injection water temperature, do **NOT** restart a RCP until a steam bubble has been formed in the Pressurizer. This will minimize the pressure transient when the first pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. (Reference 2.6.3)

R 26. If the RCPs are secured and RHR is cooling down the RCS, the temperature in the RCS loops may not be equivalent throughout the system. A RCP restart should not be attempted unless a bubble is in the pressurizer. This will minimize the pressure transient when the first pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. (Reference 2.6.3)

R 27. The greatest improvement in reducing the risk of pressurized thermal shock is by procedurally verifying that all required evolutions for which RCP□s would be needed are complete prior to securing the final RCP. (Reference 2.6.10)

R 28. Attempting to increase RCS boron concentration with no RCP's in operation can result in pockets of reduced boron concentration. OPS-NGGC-1306 provides guidance concerning suspected boron pockets. (Reference 2.6.13 and 2.7.11)

29. If the hot legs or cold legs temperature is greater than or equal to 140°F, the S-2 and S-4 fans need to be in service. Before placing these fans under clearance below 140°F, the pressurizer cooldown should be complete and there should be no immediate need to heat up above 140°F.

4.0 PRECAUTIONS AND LIMITATIONS (continued)

19. During RCS fill and vent, the # 1 Seal bypass valve should be opened briefly to vent the lines. The bypass valve should not be opened until RCS pressure is at least 100 psig and the # 1 Seal leakoff line is opened. After venting the lines, the # 1 Seal bypass valve should be closed and remain closed unless the conditions described in 18 above occur.

20. Whenever RCS temperature is greater than 160°F, at least one RCP must be in operation. [R - Reference 85H0635]

21. RCP seal flow has a normal operating range of between 8 and 13 gpm per pump. However, at no time should the total to all three RCPs exceed 31 gpm with RCS pressure at 2215 to 2255 psig. This is the maximum controlled leakage allowed by Tech Spec 3.4.6.2e. If plant conditions dictate, seal flow to each RCP may be reduced to a minimum of 6 gpm.

22. RCPs shall not be started with one or more of the RCS cold leg temperatures less than or equal to 325°F unless the secondary water temperature is less than 50°F above each of the RCS cold leg temperatures. (Reference 2.2.2)

23. When starting an RCP while the plant is solid, pressure surges large enough to lift an LTOP relief should be anticipated. Prompt action will be required to prevent LTOP actuation.

24. When RCS pressure is being maintained by the low pressure letdown control valve, changes to the RHR flow rate by throttling valves or starting and stopping the RHR pumps will result in changes in RCS pressure.

25. Maximize purification system use during periods when high coolant activity levels are anticipated:

    - Following hydrogen peroxide addition before cooldown
    - Following RCP jog
    - Reactor Startup

26. Pressurizer Relief Tank (PRT) temperature should be maintained less than 120°F.

27. A nitrogen cover pressure of 2 to 4 psig should be maintained in the pressurizer relief tank to prevent the formation of an explosive hydrogen-oxygen mixture. Explosive mixtures of oxygen and hydrogen in the PRT must be avoided at all times. The oxygen content in the tank must not exceed 4% by volume with hydrogen present. (Reference 2.7.1)

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.2 Leakage From CCW System

CAUTION

Operation of RCPs for greater than 10 minutes without CCW cooling to the motor oil coolers may result in RCP bearing damage.

INSTRUCTIONS	RESPONSE NOT OBTAINED
✱☐11. **CHECK** CCW expected to be lost for greater than 10 minutes.	☐11. **GO TO** Step 12.
☐ a. **CHECK** the Reactor is TRIPPED.	☐ a. **TRIP** the Reactor **AND GO TO** EOP PATH-1. (Continue with this procedure as time permits.)
☐ b. **STOP** ALL running RCPs. **[A.1, 2]**	
☐ c. **SHUT** 1RC-107, PRZ Spray Loop A.	
☐ d. **SHUT** 1RC-103, PRZ Spray Loop B.	

CAUTION

Reactor Makeup Water Tank contains potentially tritiated water. Making up to the CCW System from the Reactor Makeup Water Tank could result in CCW System contamination. Operation of the system while it is contaminated requires an evaluation per 10CFR50.59. **[C.2, 3]**

INSTRUCTIONS	RESPONSE NOT OBTAINED
✱☐12. **CHECK** CCW Surge Tank level STABLE OR RISING.	12. **IF** Reactor Makeup Water is needed to maintain Surge Tank level greater than 4%, **THEN PERFORM** the following (while continuing with this procedure):
	☐ a. **VERIFY** one RMW pump RUNNING.

(Continued on Next Page)

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.3 Reactor Coolant Pump Seal Malfunctions

15. (continued)

☐ d. **MAINTAIN** seal injection flow greater than calculated total #1 seal flow.

☐ e. **INITIATE** a plant shutdown using ONE of the following:

- GP-006, Normal Plant Shutdown from Power Operation to Hot Standby
- AOP-038, Rapid Downpower

☐ f. **STOP** the affected RCP within 8 hours of exceeding 6.5 gpm #1 seal leakoff flow.

☐16. **CHECK** RCPs free of #1 Seal blockage.

16. PERFORM the following:

☐ a. **DISPATCH** an operator to CNMT to read the RCP #2 seal leakoff flow.

b. **VERIFY** the following valves are OPEN:

☐ • 1CS-470, RCP Seal Water Return

☐ • 1CS-472, RCP Seal Water Return

☐ • 1CS-355, RCP A #1 Seal Water Return

☐ • 1CS-396, RCP B #1 Seal Water Return

☐ • 1CS-437, RCP C #1 Seal Water Return

☐ c. **VERIFY** VCT pressure between 20 and 30 psig per OP-107, Chemical and Volume Control System.

(Continued on Next Page)

29. 2009A NRC RO 029

Given the following plant conditions:
- The plant is operating at 100% power
- A manual blend to the RWST is being performed in accordance with OP-107.01, CVCS Boration, Dilution, and Chemistry Control
- All other systems are in automatic

The following annunciators alarm:
- ALB-007-4-3, VCT High-Low Level
- ALB-007-5-5, Computer Alarm Chem & Vol Systems
- ALB-006-8-3, Boric Acid Auto Make-Up Signal Blocked

Plant Computer provides the following information:

ERFIS ID	Description	Value and Trend
LCS0115	VCT Level	14% and lowering
LCS0112	VCT Level	86% and rising
ZCS0241	FCV115A Letdown to VCT Divert	RHT

Which ONE of the following identifies the failing indicator and the status of Emergency Makeup from the RWST?

	Failing indicator	Emergency Makeup Status
A.	LT-115 failing LOW	Available and in progress
B.	LT-115 failing LOW	Available but NOT in progress
C.	LT-112 failing HIGH	Available but NOT in progress
D.✓	LT-112 failing HIGH	Unavailable until leads on LT-112 are lifted

Plausibility and Answer Analysis

A Incorrect. Plausible since indications are that either 115 is failing low or 112 is failing high. If LT-115 were failing low then at 20% it would have started an automatic makeup. At that point actual level would have begun increasing. (Auto makeup in this case isn't available due to manual make up to RWST so level would remain relatively constant.) Computer point for FCV-115A (1CS-120) indicates that a Divert to the RHT has occurred. This indicates that actual plant conditions are such that no makeup is going to the VCT.

B Incorrect. Conditions are not consistent with a LT-115 failure (see above). If LT-115 had failed low then Emergency Makeup would still be considered available since 2/2 at the low setpoint is all that's required. Only becomes unavailable when indicator fails high.

C Incorrect. LT-112 is the failing instrument, but emergency makeup requires 2 out 2 indicators at the low setpoint and will not be available if one indicator has failed high.

D Correct. LT-112 is failing high. At 70% this caused a Divert to the RHT to begin. Actual level began lowering. At 80% a Full Divert to the RHT occurred. When LT-115 reached 20% it attempted to start an auto makeup but that can't occur while a manual blend is in progress to the RWST. Therefore level continues to lower. Emergency Makeup is unavailable until the leads on LT-112 are lifted since the instrument is failing high.

KA Statement - Chemical and Volume Control - Ability to use plant computer to evaluate system or component status

Importance Rating:	3.9 3.8
Technical Reference:	AOP-003 Rev. 26, Attachment 6, page 18 AOP-003 Rev. 26, Section 3.1 APP-ALB-006 Window 8-3
References to be provided:	None
Learning Objective:	CVCS, Obj. 12e
Question Origin:	NEW
Comments:	Sacrifice of symmetry for plausibility. Chose NOT to use redundant second halves in Emergency Makeup Status to increase the plausibility of each individual answer, ie., it isn't plausible that with LT-115 failing LOW that emergency makeup would require leads to be lifted.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-003, PMS TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	004G2.1.19	K/A 2:	

Attachment 1

Sheet 1 of 1

VCT Level Control Channels Operation

* LT-112 Divert setpoints will be changed by GP-002 during heatup.

LEVEL CHANNEL LT-112		**LEVEL CHANNEL LT-115**
	100%	
	90%	
* Full Divert to RHT →	80%	← Full Divert to RHT
	75%	← High Level Alarm
* Modulate Divert to RHT →	70%	
	60%	
	50%	
	40%	← Auto Makeup Off
	30%	
	20%	← Auto Makeup On
	15%	← Low Level Alarm
Clear Emergency Makeup →	10%	← Clear Emergency Makeup
Emergency Makeup # →	5%	← Emergency Makeup #
	0%	

REQUIRES BOTH CHANNELS TO OPERATE

DEVICES: LB-115C

SETPOINT: 20% with RWMU System not in AUTO

BORIC ACID AUTO MAKE-UP SIGNAL BLOCKED

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using the following:
   - LI-115, VCT level indicator
   - Control switches RWMU/MODE/SELECTOR and RWMU/CONTROL
2. **VERIFY** Automatic Functions: None
3. **PERFORM** Corrective Actions:

NOTE

Automatic makeup can be established for boric acid flows of 1 to 30 gpm (with RCS boron concentrations per OP-107.01) under the following conditions:
- VCT level less than or equal to 80%
- VCT pressure between 20 and 30 psig
- BAT level greater than or equal to 23%

Makeup with conditions outside of these ranges may need to be performed manually.

   a. **REFER TO** OP-107, Chemical and Volume Control System, **AND PERFORM** the following:
      (1) **IF** needed,
      THEN FILL the VCT using manual blender operation.
      (2) **IF** RCS boron concentration is within the limits of OP-107.01,
      THEN VERIFY the Reactor Makeup System in AUTO.
      (3) **START** the Reactor Makeup System using the applicable section of OP-107.01.
   b. **IF** VCT nears the low level alarm,
      THEN GO TO AOP-003, Malfunction of Reactor Makeup Control.

CAUSES:

1. Low-low level in VCT with Reactor Makeup Control System NOT in AUTO
2. Failure of LT-115 with Reactor Makeup Control System NOT in AUTO
3. Malfunction of Reactor Makeup Control System
4. Alarm circuit malfunction

REFERENCES:

1. 6-B-401 0301, 0307
2. AOP-003, Malfunction of Reactor Makeup Control
3. Technical Specifications 3.1.2.1 and 3.1.2.2
4. OP-107.01, CVCS Boration, Dilution, and Chemistry Control

30. 2009A NRC RO 030

Given the following plant conditions:
- The plant is operating at 100% power near the end of core life
- An RCS normal dilution is in progress
- During the dilution, an inadvertent Safety Injection occurs

Following the SI actuation, the dilution will. . .

A. continue to fill the Volume Control Tank until the total makeup water batch counter reaches zero.

B. continue to deliver water to the CSIP suction until the total makeup water batch counter reaches zero.

C.✓ stop due to an automatic flow deviation as a result of the RMUW pumps deenergizing.

D. stop due to an automatic flow deviation as a result of the closure of FCV-114A (1CS-155, Makeup to VCT).

Plausibility and Answer Analysis

A *Incorrect. This is normal operation for a normal dilution but the system is equipped with a flow deviation alarm that actuates because of loss of running pumps due to sequencer operation and the flow deviation alarm shuts FCV-113B and FCV-114A.*

B *Incorrect. This is normal operation for an alternate dilution but the system is equipped with a flow deviation alarm that actuates because of loss of running pumps due to sequencer operation and the flow deviation alarm shuts FCV-113B and FCV-114A.*

C *Correct. RMUW Pumps are lost when the sequencer actuates due to shedding of 1A1 and 1B1 which results in a flow deviation. The total makeup water flow deviation occurs when the total makeup flow (FT-114) is not within ±5.0 gpm of the reactor makeup water flow controller (FK-114) setpoint. This deviation is blocked for the first 15 seconds of starting makeup to allow the flow control valve to throttle to the proper position to match flow rate demanded. If the demanded flow setpoint is not within the ±5.0 gpm actual flow a TOTAL MAKEUP WATER FLOW DEVIATION alarm actuates. Automatic actions are the closure of 1CS-155 (FCV-114A) if open and the closure of 1CS-156 (FCV-113B) if open.*

D *Incorrect. A flow deviation is generated but it is because RMUW Pumps are lost when the sequencer actuates due to shedding of 1A1 and 1B1. FCV-114A shuts as an auto action, it is not the cause.*

KA Statement - Chemical and Volume Control - Knowledge of the physical connections and/or cause-effect relationships between (SYSTEM) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Makeup system to VCT

Importance Rating:	3.1 3.1
Technical Reference:	APP-ALB-006 Window 7-3 Rev. 21, page 18
References to be provided:	None
Learning Objective:	PMS, Obj. 9a
Question Origin:	Modified OIT Exam Bank, PMS (09A) 002
Comments:	

Origin:	MODIFIED	Cog Level:	H
Difficulty:	3	Reference:	PMS STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	004K1.06	K/A 2:	

DEVICES: FB-114A

SETPOINT: > 5 gpm difference between actual makeup flow and flow set on FK-114, 15 sec after start

TOTAL MAKE UP WATER FLOW DEVIATION

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using both of the following:
   - Total makeup flow indication on FR-113 Pen 2
   - Total flow setpoint on FIS-114
2. **VERIFY** Automatic Functions:
   - 1CS-155 (FCV-114A) shuts, if OPEN **AND** in AUTO
   - 1CS-156 (FCV-113B) shuts, if OPEN **AND** in AUTO
3. **PERFORM** Corrective Actions:

NOTE

Automatic makeup can be established for boric acid flows of 1 to 30 gpm (with RCS boron concentrations per OP-107.01) under the following conditions:
- VCT level less than or equal to 80%
- VCT pressure between 20 and 30 psig
- BAT level greater than or equal to 23%

Makeup with conditions outside of these ranges may need to be performed manually.

   a. **VERIFY SHUT** 1CS-155 **AND** 1CS-156.
   b. **IF** manual blender operations are in progress, **THEN PLACE** the RMW Mode Selector Switch in OFF.
   c. **IF** VCT nears the low level alarm, **THEN GO TO** AOP-003, Malfunction of Reactor Makeup Control.
   d. **VERIFY** that the following are properly aligned for the section of OP-107.01 in use:
      (1) Control switches for 1CS-151, 1CS-155, 1CS-156, and 1CS-283
      (2) Controllers for 1CS-151 (FCV-114B) and 1CS-283 (FCV-113A)
      (3) Potentiometer settings for FCV-113A and FCV-114B
   e. **VERIFY** that a Reactor Makeup Water pump and a Boric Acid pump are available.
   f. **REFER TO** the applicable section of OP-107.01 **AND START** Reactor Makeup System.
   g. **IF** alarm comes in after restarting Reactor Makeup System, **THEN PERFORM** manual Reactor makeup.

CAUSES:

1. Valve misalignment
2. Malfunction of 1CS-151 (FCV-114B), 1CS-155 (FCV-114A), 1CS-156 (FCV-113B), or 1CS-283 (FCV-113A)
3. Inadequate flow from reactor makeup water pumps
4. Instrument or alarm circuit malfunction

REFERENCES:

1. 6-B-401 0239, 0301, 0302
2. AOP-003, Malfunction of Reactor Makeup Control
3. Technical Specifications 3.1.2.1 and 3.1.2.2
4. OP-107.01, CVCS Boration, Dilution, and Chemistry Control

QUESTIONS REPORT
for OIT Exam Bank

2. PMS (09A) 002/RS/H/3/PMS TEXT/N//004A3.09/004K1.06

Given:

- Plant is at full power near the end of core life
- RCS normal dilution is in progress
- During the dilution, an inadvertent Safety Injection (SI) actuation occurs

Following the SI actuation, the dilution will. . .

A. continue to fill the Volume Control Tank until total makeup water batch counter reaches zero.

B. continue to deliver water to the CSIP suction until total makeup water batch counter reaches zero.

C. stop due to an SI generated shutdown signal to the Reactor Makeup Water controller.

D✓ stop due to an automatic flow deviation as a result of the closure of FCV-114A (1CS-155, Makeup to VCT).

K/A1 Statement: Ability to monitor automatic operation of the CVCS, including: VCT level
K/A2 Statement: Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following systems: Makeup system to VCT

Importance Rating:	RO 3.3 / SRO 3.2 (RO 3.1 / SRO 3.1)
Technical Reference:	PMS Text
References to be provided:	N/A
Learning Objective(s):	N/A
Question origin:	Bank
Comments:	**DO NOT USE UNTIL QUESTION IS EDITED - TOM HUNT, 8/2/06;** The distrators have some wording issues. It needs editing.

01SYSWK3, 04SYSWK7, 06SYSTST3 (Obj 9a)

Program:	RS	Cog Level:	H
Difficulty:	3	Reference:	PMS TEXT
Ref. Provided?:	N	Key Words:	
K/A 1:	004A3.09	K/A 2:	004K1.06

31. 2009A NRC RO 031

Given the following plant conditions:
- The plant is in Mode 5
- The RCS is in solid plant operation
- 'A' Train equipment is in service
- 'A' Train RHR is aligned for Shutdown Cooling
- PCV-145 (1CS-38), Letdown Pressure Control Valve, is in AUTO

Which ONE of the following will result in an increase in RCS pressure?

A. Output fails to 100% on HC-142.1 (1CS-28), RHR Letdown Controller

B. Output fails to 100% on PK-145.1 (1CS-38), Letdown Pressure Controller

C.✓ Instrument Air is lost to HCV-142 (1CS-28), RHR Letdown Control Valve

D. Instrument Air is lost to PCV-145 (1CS-38), Letdown Pressure Control Valve

Plausibility and Answer Analysis

A *Incorrect. For this component, the output failing high will result in 1CS-28 opening which will reduce RCS Pressure. Plausible because for some components (like the Charging FCV) output failing high will result in an RCS pressure increase.*

B *Incorrect. For this component, the output failing high will result in 1CS-38 opening which will reduce RCS Pressure. Plausible because for some components (like the Charging FCV) output failing high will result in an RCS pressure increase. Additionally, if the setpoint were to fail high while in Auto, this would result in an RCS pressure increase.*

C *Correct. 1CS-28 fails shut which results in a loss of letdown and an RCS pressure increase.*

D *Incorrect. 1CS-38 fails open which results in increased letdown and an RCS pressure decrease but as identified by the correct answer other components will result in an RCS pressure increase.*

KA Statement - Residual Heat Removal - Knowledge of the operational implications of the following concepts as they apply to RHRS: Plant response during solid plant: pressure changes due to the relative incompressibility of water

Importance Rating:	2.7 3.1
Technical Reference:	AOP-017 Rev. 28 Attachment 1, page 40
References to be provided:	None
Learning Objective:	LP-AOP-3.17, Obj. 2
Question Origin:	NEW
Comments:	**K/A replaced on 11/17/08. New K/A provided by chief examiner through random selection.**

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-017
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	005K5.05	K/A 2:	

Attachment 1

Sheet 1 of 5

Fail Positions for Major Valves Controlled by Instrument Air

Chemical and Volume Control System		
1CS-38	Ltdn Pressure (PK-145.1)	Fail Open
1CS-98	BTRS Bypass	Fail Open
1CS-231	Charging Flow (FK-122.1)	Fail Open
1CS-243	RCP Seal Wtr Inj Flow (HC-186.1)	Fail Open
1CS-283	Boric Acid to Boric Acid Blender (FCV-113A)	Fail Open
1CS-480	Alternate Charging Line	Fail Open
1CS-492	Normal Charging Line	Fail Open
1CS-1	Letdown Isolation LCV-460	Fail Shut
1CS-2	Letdown Isolation LCV-459	Fail Shut
1CS-7	45 gpm Letdown Orifice A	Fail Shut
1CS-8	60 gpm Letdown Orifice B	Fail Shut
1CS-9	60 gpm Letdown Orifice C	Fail Shut
1CS-11	Letdown Isolation	Fail Shut
1CS-28	RHR Letdown (HC-142.1)	Fail Shut
1CS-151	RMW to Boric Acid Blender FCV-114B	Fail Shut
1CS-155	Make Up to VCT FCV-114A	Fail Shut
1CS-156	Make Up to CSIP Suction FCV-113B	Fail Shut
1CS-460	Excess Letdown Isolation	Fail Shut
1CS-461	Excess Letdown Isolation	Fail Shut
1CS-487	Pressurizer Aux Spray	Fail Shut
1CS-50	Letdown to VCT/Demin TCV-143	Fail to VCT
1CS-120	Letdown to VCT/Hold Up Tank LCV-115A	Fail to VCT
Chill Water System		
1CH-115	Chilled Water to NESSR Fan Clrs Isol	Fail Shut
1CH-116	Chilled Water to NESSR Fan Clrs Isol	Fail Shut
1CH-125	Chilled Water from NESSR Fan Clrs Isol	Fail Shut
1CH-126	Chilled Water from NESSR Fan Clrs Isol	Fail Shut
1CH-148	Chilled Water to NESSR Fan Clrs Isol	Fail Shut
1CH-149	Chilled Water to NESSR Fan Clrs Isol	Fail Shut
1CH-196	Chilled Water from NESSR Fan Clrs Isol	Fail Shut
1CH-197	Chilled Water from NESSR Fan Clrs Isol	Fail Shut

32. 2009A NRC RO 032

Given the following plant conditions:
- All RCPs are secured
- The crew is implementing FRP-P.1, Reponse to Imminent Pressurized Thermal Shock
- SI Termination Criteria is being evaluated

Which ONE of the following describes why SI Termination is desired in FRP-P.1?

	Allows RCS	Minimizes
A.✓	pressure reduction	further cooldown
B.	pressure reduction	further inventory increases
C.	temperature increase	further cooldown
D.	temperature increase	further inventory increases

Plausibility and Answer Analysis

A *Correct. FRP-P.1 uses less restrictive SI Termination Criteria because SI flow may be contributing to the cooldown and RCS pressure reduction is desired but SI flow may hold up RCS pressure.*

B *Incorrect. FRP-P.1 uses less restrictive SI Termination Criteria because SI flow may be contributing to the cooldown and RCS pressure reduction is desired but SI flow may hold up RCS pressure. Terminating SI will give the operator better control of inventory with the Charging FCV but the concern is PTS with cold SI water cooling the downcomer.*

C *Incorrect. SI flow may be adding to the cooldown and terminating SI may alleviate this but the strategy of FRP-P.1 is to stabilize RCS temperature, not to increase it.*

D *Incorrect. SI flow may be adding to the cooldown and terminating SI may alleviate this but the strategy of FRP-P.1 is to stablize RCS temperature not to increase it. Terminating SI will give the operator better control of inventory with the Charging FCV but the concern is PTS with cold SI water cooling the downcomer.*

QUESTIONS REPORT
for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Emergency Core Cooling - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: (CFR: 41.5 / 45.5) Avoidance of thermal and pressure stresses due to pump startup

Importance Rating:	3.1 3.4
Technical Reference:	WOG Background Document for FRP-P.1
References to be provided:	None
Learning Objective:	LP-EOP-3.14, Obj. 4c
Question Origin:	NEW
Comments:	(KA Match) With Safety Injection actuated (ECCS pump start up), the candidate must understand the RCS parameters that are a PTS concern following a large cooldown and understand the strategy in FRP-P.1 for avoiding further thermal and pressure stresses by terminating SI, stabilizing RCS temperature, and reducing RCS pressure.

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	FRP-P.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	006A1.01	K/A 2:	

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-P.1 is to limit (or prevent) any potential flaw growth which may have occurred as detected by the Integrity Status Tree.

The following subsections provide a summary of the major categories of operator actions and the key utility decision points for guideline FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-P.1 is given on the following page in the form of major action categories. These are described below in more detail.

o Stop RCS Cooldown

These steps try to identify any possible source of excessive cooldown and to terminate or limit that cooldown.

o Terminate SI If Criteria Satisfied

Safety injection flow, if present, is a significant contributor to any cold leg temperature decrease. It can also be a significant contributor to an overpressure condition if the RCS is intact. A check for SI termination is performed early in this guideline based on less restrictive criteria than in other SI termination steps in the ORGs to try to remove its unfavorable PTS effects.

o Depressurize RCS To Minimize Pressure Stress

System pressure is reduced to a minimum to decrease the pressure stress on the reactor vessel. Minimum subcooling or high PRZR level is used as the criterion for the pressure reduction termination.

33. 2009A NRC RO 033

Given the following plant conditions:
- An inadvertent SI has occurred
- PATH-1 is in progress
- The RO notes that a PRZ PORV lifts correctly to reduce RCS pressure

Subsequently, the RO notes the following:
- The PORV indicates shut but it is suspected of having failed to fully reseat
- PRT pressure indicates 54 psig

Which ONE of the following describes the expected PORV Tailpipe Temperature if the Pressurizer PORV has failed to fully reseat?

PORV Tailpipe Temperature will be approximately equal to. . .

A. Pressurizer temperature.

B. Containment temperature

C.✓ Saturation Temperature for PRT pressure

D. Saturation Temperature for Containment pressure

Plausibility and Answer Analysis

A *Incorrect. Due to design, the PORV Tailpipe temperature will indicate saturation temperature for the PRT pressure. Pressurizer Safety Valve Temperatures will indicate Pressurizer temperature.*

B *Incorrect. This indicator is effected by CNMT temperature and frequently alarms when CNMT Conditions are adverse but if a PORV has failed to reseat it will read much higher.*

C *Correct. Due to design, the PORV Tailpipe temperature will indicate saturation temperature for the PRT pressure.*

D *Incorrect. If the PRT Rupture Disc is blown, the PORV Tailpipe temperature will indicate saturation temperature for the CNMT pressure but with a PRT pressure of 54 psig the rupture disc is intact.*

QUESTIONS REPORT
for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Pressurizer Relief/Quench Tank - Ability to monitor automatic operations of the (SYSTEM) including: (CFR: 41.7 / 45.5) Components which discharge to the PRT

Importance Rating:	2.7 2.9
Technical Reference:	PZR Student Text Rev 2, page 20
References to be provided:	None
Learning Objective:	PZR, Obj. 3b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	2	Reference:	PZR STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	007A3.01	K/A 2:	

120°F and is inaccurate at higher temperatures. Graphs in curve book show actual vs. indicated level for hot and cold calibrated instruments at various saturation temperatures (H-X-22 & 23).

The PRT has temperature, level and pressure instrumentation on the MCB and ACP to monitor status of leakage into the tank. MCB and ACP alarms are also provided from this instrumentation:

- PRT Pressure, PT-472 (0-120 psig) high pressure alarm at 8 psig.
- PRT Level, LT-470 (0-100%) high/low level alarms at 83%/62%
- PRT Temperature, TT-471 (50-350°F) high temperature alarm at 120°F.

Other Pressurizer related instrumentation for indication and alarm only:

- PORV discharge temp TT-463, (50-400°F, 1 detector in the common discharge line down stream of the PORVs) helps identify one or more PORVs leaking or open. Provides a high temperature alarm at 140°F.
- Safety valve discharge temperatures TT-465, 467 and 469 (50-400°F, 1/valve located upstream of the valve) helps identify which safety is leaking/open. Provides a high temperature alarm at 165°F.
- Pressurizer Liquid and Vapor space temperatures TT-453 and 454 (100-700°F, 1 meter for each). Provides indication of saturation conditions. Each provides a high temperature alarm at 665°F.

Because of the difference in location for the PORV Discharge Temperature Transmitter compared to the Safety Valve Temperature Transmitters, the temperature response for a stuck open valve will not be the same. The Safety Valve indicator will indicate close to the temperature of the PZR steam space. The PORV indicator will indicate close to saturation temperature for the pressure in the PRT (containment if the PRT has ruptured).

- Surge line temp TT-450 (100-700°F) is used to monitor for insurge and outsurge during heatup and cooldown evolutions. Provides a low temperature alarm at 530°F. T.S. limits on heatup and cooldown of the Pressurizer do not apply to the surge line temperature indicator.

34. 2009A NRC RO 034

Given the following plant conditions:
- The plant is operating at 100% power
- ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press Or Temp, is in alarm
- PRT pressure indicates 9 psig
- PRT temperature indicates 101°F
- PRT level indicates 74%

Which ONE of the following describes the required action for this alarm in accordance with the Annunciator Panel Procedure and OP-100, Reactor Coolant System?

A. Drain the PRT to the Waste Hold Tank

B. Drain the PRT to the Reactor Coolant Drain Tank

C.✓ Vent the PRT to the Waste Gas Vent Header

D. Vent the PRT to the Reactor Coolant Drain Tank

Plausibility and Answer Analysis

A Incorrect. The APP will direct lowering level if level is high and the PRT will be processed to the WHT with the RCDT Pumps but level is normal and pressure is high.

B Incorrect. The APP will direct lowering level if level is high and the PRT will be processed to the WHT with the RCDT Pumps but level is normal and pressure is high.

C Correct. Pressure is high and the PRT should be vented to the WG Vent Header.

D Incorrect. Pressure is high and the PRT should be vented but it will be vented to WG, not the RCDT. However during WG outages the RCDT will be vented to the PRT.

KA Statement - Pressurizer Relief/Quench Tank - Knowledge of annunciators alarms, indications or response procedures

Importance Rating:	4.2 4.1
Technical Reference:	APP-ALB-009 Rev. 11 Window 8-1, page 29 OP-100 Rev. 28, page 18 & 26 OP-120.07
References to be provided:	None
Learning Objective:	PZR, Obj. 5
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	APP-ALB-009
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	007G2.4.31	K/A 2:	

PRESSURIZER RELIEF TANK HIGH-LOW LEVEL PRESS OR TEMP

DEVICES:	SETPOINT:
LS-01RC-0470AW (LB-470A)	83%
LS-01RC-0470BW (LB-470B)	62%
PS-01RC-0472W (PB-472)	8 PSIG
TS-01RC-0471W (TB-471)	120°F

REFLASH: YES

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PRT level LI-470.1
   b. PRT pressure PI-472.1
   c. PRT temperature TI-471.1

2. **VERIFY** Automatic Functions: None

3. **PERFORM** Corrective Actions:

   a. **IF** level is low,
   THEN FILL the PRT with primary water to normal level using OP-100.
      (1) **IF** level continues to fall,
      THEN DIRECT an Operator to inspect the PRT for leaks.

   b. **IF** level is high,
   THEN DRAIN the PRT to normal level using OP-100.
      (1) **IF** level continues to rise,
      THEN PERFORM Step 3.e.

   c. **IF** pressure is high, **AND** level and temperature is Normal
   THEN perform the following:
      (1) **VENT** the PRT using OP-100.
      (2) **DIRECT** an Operator to verify N_2 Regulator is set at 0 psig.
      IF regulator setting is correct,
      THEN PERFORM Step 3.e.

R d. **IF** temperature is high,
   THEN REDUCE PRT temperature by recirculation through the RCDT Heat exchanger, **AND PERFORM** Step 3.e.

   e. **DETERMINE** source of inleakage. **REFER** to CAUSES Section for potential sources.

CAUSES:

1. PRZ PORVs and/or Safeties lifting or leaking
2. Relief valve actuation from RHR or CVCS systems
3. Leak or rupture of PRT or associated piping
4. Alarm circuit or instrumentation malfunction

REFERENCES:

1. Tech Specs 3.4.2.1, 3.4.2.2, 3.4.4, and 3.4.6.2
2. FSAR 5.4.11
3. 6-B-401 0936
4. ESR 9700584

8.2. Venting the Pressurizer Relief Tank

8.2.1. Initial Conditions

1. Attachment 1 is complete. _____
2. Attachment 2 is complete. _____
3. Waste Gas system is available. _____

NOTE: Venting the PRT to the Waste Gas System generally will increase the in-service Gas Decay Tank Pressure 2 to 10 psig.

4. In-service Gas Decay Tank has the capacity to receive the PRT Vent (2 to 10 psig). _____
5. Gas Decay Tanks E and F are not in service. _____
6. Chemistry has been notified to sample for hydrogen, nitrogen, and oxygen concentration during the venting process. This requirement may be waived if the source of the pressure increase is known to be either Nitrogen or Hydrogen. _____

8.2.2. Procedure Steps

1. **MAINTAIN** PRT pressure less than RCS pressure to prevent PRT backleakage into the RCS or Containment atmosphere. _____
2. **ESTABLISH** communications between the Main Control Room, the Radwaste Control Room, and the Operator at the manual PRT vent valve. _____
3. **SHUT** 1NI-373, N_2 to PRT Regulator Isolation Valve. _____
4. At the MCB, **OPEN** 1RC-144, N_2 TO PRT. _____
5. At the MCB, **OPEN** 1RC-141, N_2 TO PRT. _____
6. **IF** the VCT is being degassed per OP-120.07,
   THEN PERFORM the following Substeps:
   a. **ADJUST** VCT purge as necessary during PRT venting to maintain Waste Gas Inlet Hydrogen at less than 9%. _____
   b. **GO TO** Step 8.2.2.8. _____

8.4. Draining The Pressurizer Relief Tank

8.4.1. Initial Conditions

1. Attachment 1 is complete. _____
2. Attachment 2 is complete. _____

8.4.2. Procedure Steps

CAUTION

Good communication between the MCR and the RWCR must be maintained to prevent damage to the RCDT Pumps while draining to the desired level.

1. **MAINTAIN** PRT pressure less than RCS pressure to prevent PRT backleakage into the RCS or Containment atmosphere. _____
2. **REQUEST** the Radwaste Control Room process the PRT with the RCDT pumps per OP-120.08, Section 8.7. _____
3. At the MCB, **VERIFY OPEN** 1RC-144, N_2 TO PRT. _____
4. At the MCB, **VERIFY OPEN** 1RC-141, N_2 TO PRT. _____
5. **WHEN** directed by the Radwaste Control Room,
   THEN OPEN 1RC-135, PRT DRAIN. _____

NOTE: The intent of the following Step is to have pressure within the normal band, and have ALB-010-8-5A, CMPTR ALARM RX COOLANT, clear.
If ALB-010-8-5A is in due to low pressure, it would be necessary to increase the pressure.

6. As level is lowered in the PRT, **ADJUST** 1NI-241, Nitrogen to PRT Regulator, to control PRT pressure at 2 to 4 psig. _____
7. **WHEN** desired level is reached,
   THEN NOTIFY the Radwaste Control Room to stop the RCDT Pump. _____
8. **WHEN** directed by Radwaste Control Room,
   THEN SHUT 1RC-135. _____

35. 2009A NRC RO 035

Given the following plant conditions:
- The plant is operating at 100% power
- Excess Letdown is in service in preparation for removing Normal Letdown from service

The RO observes the following indications:
- Reactor Power has increased
- Control Bank 'D' Rods begin to step

Which ONE of the following CCW leak locations would result in these indications and what direction of rod motion is expected?

	CCW Leak	Rod Direction
A.✓	Seal Water Heat Exchanger	IN
B.	Seal Water Heat Exchanger	OUT
C.	Excess Letdown Heat Exchanger	IN
D.	Excess Letdown Heat Exchanger	OUT

Plausibility and Answer Analysis

A Correct. A leak in the Seal Water HX will leak to the CVCS system which will result in a dilution and inward rod motion.

B Incorrect. Plausible as a leak in the Seal Water HX will leak to the CVCS system which will result in a dilution. Rods will move in not out but plausible if the candidate believes reactor power has increased due to rod motion out.

C Incorrect. Plausible as Excess Letdown has just been placed in service but with the pressure in this HX, the RCS will leak to the CCW system. Rods will move in.

D Incorrect. Plausible as Excess Letdown has just been placed in service but with the pressure in this HX, the RCS will leak to the CCW system. Rods will move in not out but plausible if the candidate believes reactor power has increased due to rod motion out.

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Component Cooling Water - Knowledge of the effect that a loss or malfunction of the (SYSTEM) will have on the following: (CFR: 41.7 / 45.6) CRDS

Importance Rating:	2.7 2.9
Technical Reference:	AOP-014-BD Rev. 13, page 4
References to be provided:	None
Learning Objective:	LP-AOP-3.14, Obj. 2
Question Origin:	NEW
Comments:	(KA Match) Due to pressure differential, the CCW System would leak into the Seal Water Heat Exchanger. A non-borated leak into this system will cause a reactivity effect by dilution of the RCS. The question asks for the Control Rod Drive system's response to this fault in the CCW system.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-014
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	008K3.02	K/A 2:	

1.0 DISCUSSION (continued)

13. Due to pressure differential, the CCW System could leak into these systems/components:
    - RHR Heat Exchanger (if RHR pressure is less than CCW pressure)
    - RHR Pump Seal Cooler (if RHR pressure is less than CCW pressure)
    - RCP Thermal Barrier Coolers (if RCS pressure is low)
    - Seal Water Heat Exchanger
    - RDCT Heat Exchangers
    - Spent Fuel Pool Cooling
    - RCP Oil Reservoirs
    - BRS Evaporator
    - ESW at the CCW Heat Exchangers

<u>OSI-PI Guidance</u>

When OSI-PI displays, mimics, trends, and/or spreadsheets are used, the following guidance applies (reference CSP-NGGC-0004 for details):

- Quality Level D – B is included in the file name and/or display of OSI-PI functions.

  <u>Example</u>: HNP_SWQL_D_POM_AOP-018.xls is Quality Level D.
- Qualified OSI-PI functions may be used alone to support specified requirements.
- Non-Qualified OSI-PI functions shall only be used along with diverse indication.
- Qualified OSI-PI functions should be accessed using Start → Progress Energy PI Displays → Harris Qualified button.
- If the desired OSI-PI function is not available using the Harris Qualified button, it can be accessed by the following methods (Quality Level indicated as described above):
  - Start → Programs → Business Apps → PI System → HNP QPIM
  - Start → Progress Energy PI Displays → Harris General
  - Start → Programs → Business Apps → PI System → HNP GPIM

36. 2009A NRC RO 036

Given the following plant conditions:
- The plant is operating at 100% power
- 'A' CCW pump is running
- 'B' CCW pump is in standby

The following occurs:
- 'A' CCW trips on overcurrent
- Breaker 105, Emergency Bus A-SA to Aux Bus D Tie, trips open on fault

Which ONE of the following describes the response of the 'B' CCW Pump?

A. Will NOT auto start. Must be started manually

B. Will auto start from a sequenced safeguards signal

C✓ Will auto start due to low pressure in the CCW header

D. Will auto start due to the electrical trip of the 'A' CCW pump

Plausibility and Answer Analysis

A Incorrect. Auto start is available. Plausible since other plant pumps are not equipped with auto start features (CBP).

B Incorrect. Plausible since the 'A' sequencer will run when Breaker 105 trips open but 'B' sequencer will not run in this situation.

C Correct. The 'B' CCW pump will auto start when system pressure reaches 52 psig.

D Incorrect. Other plant pumps are equipped with an auto start on electrical fault of the running pump (MFP, NSW) but the CCW pump does not.

KA Statement - Component Cooling Water - Knowledge of (SYSTEM) design feature(s) and or interlock(s) which provide for the following: (CFR: 41.7) The "standby" feature for the CCW pumps

Importance Rating:	2.7 2.9
Technical Reference:	APP-ALB-005 Rev. 18, page 51
References to be provided:	None
Learning Objective:	CCW, Obj. 7a
Question Origin:	BANK OIT Development Bank 008 K.4.09 002
Comments:	K/A match from bank question.

Origin:	BANK	Cog Level:	H
Difficulty:	2	Reference:	APP-ALB-005
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	008K4.09	K/A 2:	

8-2

DEVICES: PB-0650B

SETPOINT: 57 psig

CCW PUMPS B DISCH HEADER LOW PRESS

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PI-650 SB, CCW Header B Pressure

2. **VERIFY** Automatic Functions:
   a. CCW Pump 1B-SB will start at 52 psig discharge pressure

3. **PERFORM** Corrective Actions:
   a. **VERIFY** CCW Pump 1B-SB starts.
   b. **IF** alarm is due to leaks **OR** loss of CCW,
   THEN GO TO AOP-014, Loss of Component Cooling Water.

CAUSES:

1. Valve misalignment
2. CCW pipe break or leakage
3. Loss of CCW pumps
4. Alarm circuit or instrumentation malfunction

REFERENCES:

1. Tech Specs 3.7.3
2. AOP-014, Loss of Component Cooling Water
3. ESR 95-0264
4. 2165-S-1319
5. 2166-B-401 0931

37. 2009A NRC RO 037

Given the following plant conditions:
- The plant is operating at 100% power
- Multiple annunciators alarm in the MCR
- The BOP announces that all MSIVs have shut
- The USCO directs a manual reactor trip
- The RO announces that 1CS-11, Letdown Isolation valve and 1MS-72, MS Line 'C' to TDAFW have lost power

Which ONE of the following Pressurizer PORV indications is lost and what Technical Specification action is required within 1 hour due to this event?

A.✓ 1RC-114 (PCV-444B SB)
Close the associated block valve and remove power

B. 1RC-114 (PCV-444B SB)
Close the associated block valve with power maintained

C. 1RC-118 (PCV-445A SA)
Close the associated block valve and remove power

D. 1RC-118 (PCV-445A SA)
Close the associated block valve with power maintained

Plausibility and Answer Analysis

A Correct. Control power is supplied by 125 VDC DP-1B-SB circuit 20 for 1RC-114 (PCV-444B). T.S. 3.4.4 action b is for causes other than excessive seat leakage and requires the block valve to be closed with power removed.

B Incorrect. Control power is supplied by 125 VDC DP-1B-SB circuit 20 for 1RC-114 (PCV-444B) but T.S. 3.4.4 action b is for causes other than excessive seat leakage and requires the block valve to be closed with power removed. Power is only maintained if the cause is excessive seat leakage.

C Incorrect. PCV-445A SA receives DC power also but it is from 'A' Train when 'B' Train is lost. T.S. 3.4.4 action b is for causes other than excessive seat leakage and requires the block valve to be closed with power removed.

D Incorrect. PCV-445A SA receives DC power also but it is from 'A' Train when 'B' Train is lost. T.S. 3.4.4 action b is for causes other than excessive seat leakage and requires the block valve to be closed with power removed. Power is only maintained if the cause is excessive seat leakage.

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Pressurizer Pressure Control - Knowledge of electrical power supplies to the following: (CFR: 41.7) Indicator for PORV position

Importance Rating:	2.8 3.0
Technical Reference:	Tech Specs 3.4.4.b, page 3/4 4-11 (p. 236) PZRPC Student Text Rev. 2, page 11
References to be provided:	None
Learning Objective:	PZRPC, Obj. 8h
Question Origin:	NEW
Comments:	Control power to the Pressurizer PORV's is supplied by 125 VDC DP-1B-SB circuit 20 for 1RC-114 (PCV-444B), 125 VDC DP-1A-SA circuit 20 for 1RC-118 (PCV-445A), and 125 VDC DP-1A-1 circuit 22 for 1RC-116 (PCV-445B). The power supplied to the MSIVs is 125V DC through DP-1A-SA, circuit 20; and DP-1B-SB, circuit 20. Loss of power to either circuit causes the MSIV's to close but continue to have position indication.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	T.S. 3.4.4
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	010K2.03	K/A 2:	

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

   1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b., above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

primary source, and it has a setpoint of 98 psig. The two parallel air supply regulators (1IA-999-R1 and 1IA-1000-R1) are the backup source, and their setpoint is 92 psig. PORV control switches and position indication are provided on the MCB and ACP per the requirements of NUREG-0737, Item II.D.3. 1RC-114 and 1RC-118 are safety related PORVs that provide for manual control of RCS pressure during a SGTR. These same two PORVs also provide low temperature overpressure protection (LTOP) when the primary is cooled down to below 350°F. The LTOP function is non-safety related but is required by Tech Specs. The following table shows the PORV valve number, PCV number and power supply.

Table 2: Pressurizer PORV Power Supplies

RC Valve	PCV	Power Supply
1RC-114	PCV-444B-SB	125 VDC DP-1B-SB ckt 20
1RC-116	PCV-445B	125 VDC DP-1A-1 ckt 22
1RC-118	PCV-445A-SA	125 VDC DP-1A-SA ckt 20

A motor operated block valve is provided immediately upstream of each PORV. The block valves are used to isolate their respective PORVs due to malfunction, leakage or for testing. Control switches and position indication are provided on the MCB and ACP. Additional component details on PRZ PORVs and PORV block valves are located in the PRZ student text. The following table shows the PORV, the block valve and block valve power supply.

Table 3: Pressurizer PORV Block Valves & Power Supplies

PORV	Block Valve	Power Supply
1RC-114	1RC-113	480V MCC 1B24 3C
1RC-116	1RC-115	480V MCC 1A24 3A
1RC-118	1RC-117	480V MCC 1A24 6D

When in automatic, PORV PCV-444B is controlled by the master pressure controller, and will open when the output of the master pressures controller reached 87.5% (corresponds to 100 psig above controller setpoint). The PORV will shut when master pressure controller output lower to 75%. The other two PORVs are controlled in automatic by bistables from PT-445 that actuate at a pressure setpoint of 2335 pisg. The PZR PORVs can be operated in manual, independent of the master pressure controller.

38. 2009A NRC RO 038

Given the following plant conditions:
- The plant is operating at 90% power
- Pressurizer pressure is 2235 psig
- RCS average temperature is 587°F
- AFD is indicating + 4%

Which ONE of the following changes will result in the unit operating closer to an OTΔT trip setpoint?

A.✓ Increasing AFD to +15%

B. Decreasing power to 88%

C. Increasing pressure to 2250 psig

D. Decreasing temperature to 585°F

Plausibility and Answer Analysis

OVERTEMPERATURE DT TRIP: The trip provides protection against DNB in the core for transients that are slow with respect to piping transient delays from the core. This circuit trips the reactor if RCS DT equals the trip setpoint in two of the three reactor coolant loops. The base setpoint is variable based on plant conditions with a nominal value of 118.5% of full power DT. Since the boiling point is affected by many factors, the trip setpoint is varied (higher or lower) by Pressurizer pressure, Tavg, and the NIS axial flux difference. The setpoint decreases from its nominal value if Pressurizer pressure decreases from 2235 psig, Tavg increases from 588.8°F or AFD is excessively positive or negative.

A *Correct. AFD is an input to OTΔT setpoint adjustment and with AFD greater than +12%, the setpoint is automatically reduced resulting in operating closer to the setpoint.*

B *Incorrect. RCS Loop ΔT is the parameter that is monitored to determine if a trip is exceeded. RCS Loop ΔT is directly proportional to Reactor Power and reads out in percent but reducing power will result in operating further from the setpoint.*

C *Incorrect. RCS Pressure is an input to OTΔT setpoint adjustment but Increasing RCS Pressure is a reward (further from DNB) and will result in operating further from the setpoint.*

D *Incorrect. RCS Temperature is an input to OTΔT setpoint adjustment but Decreasing RCS Temperature is a reward (further from DNB) and will result in operating further from the setpoint.*

KA Statement - Reactor Protection - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: Trip setpoint adjustment

Importance Rating:	2.9 3.4
Technical Reference:	T.S. Table 2.2-1 for OTDT pages 2-7, 2-8 (pages 87,88)
References to be provided:	None
Learning Objective:	RPS, Obj. 12c
Question Origin:	BANK OIT Development Bank Harris Audit RO 04 011
Comments: (K/A Match)	The OTDT setpoint is adjusted automatically by the Reactor Protection System. It is varied based on Tavg, RCS Pressure, and AFD. The candidate must understand these inputs and evaluate what will result in an automatic lowering of the setpoint.

Origin:	BANK	Cog Level:	F
Difficulty:	2	Reference:	T.S. 2.2.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	012A1.01	K/A 2:	

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left[\frac{1}{1 + \tau_3 S}\right] \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left[\frac{1}{1 + \tau_6 S}\right] - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 0 s, τ_2 = 0 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 4 s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.185;

K_2 = 0.0224/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
T'	=	Reference T_{avg} at RATED THERMAL POWER (≤588.8°F);
K_3	=	0.0012/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For q_t - q_b between -21.6% and +12.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of q_t - q_b exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of q_t - q_b exceeds + 12.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.50% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

39. 2009A NRC RO 039

Which ONE of the following Reactor trips has the primary function of assuring fuel integrity by limiting fuel rod power density (kW/ft)?

A.✓ Overpower Delta T (OPΔT)

B. Power Range High Flux Trip

C. Low Primary Coolant Flow Trip

D. Overtemperature Delta T (OTΔT)

Plausibility and Answer Analysis

A *Correct. This trip provides assurance that fuel integrity will be maintained by limiting power density of all fuel rods.*

B *Incorrect. This trip is to protect against rapid reactivity excursions.*

C *Incorrect. This trip provides protection against DNB for all loss of flow conditions.*

D *Incorrect. This trip provides protection against DNB in the core for transients that are slow with respect to piping transient delays from the core.*

KA Statement - Reactor Protection - Knowledge of the operational implications of the following concepts as they apply to the (SYSTEM): (CFR: 41.5 / 45.7) Power density

Importance Rating:	3.1 3.3
Technical Reference:	Tech Spec Bases pg B 2-3 to 2-6 (pages 96-99)
References to be provided:	None
Learning Objective:	RPS, Obj 12d
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	T.S BASES
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	012K5.02	K/A 2:	

LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor and an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by the setpoint value. The Steam Generator Low Water level portion of the trip is activated when the setpoint value is reached, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power-level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

40. 2009A NRC RO 040

Given the following plant conditions:

- A LOCA has occurred
- RCS pressure is 400 psig
- Containment pressure is 15.1 psig and rising
- Both 1CT-12 SA and 1CT-11 SB, CNMT Spray Chemical Addition valves, failed to OPEN automatically

Assuming NO operator action has occurred, which ONE of the following describes the effect on Containment?

A. Airborne Iodine removal inside containment will be reduced;
The water collecting in containment sumps will have a HIGHER pH

B.✓ Airborne Iodine removal inside containment will be reduced;
The water collecting in containment sumps will have a LOWER pH

C. The peak pressure inside containment will be higher;
The water collecting in containment sumps will have a HIGHER pH

D. The peak pressure inside containment will be higher;
The water collecting in containment sumps will have a LOWER pH

Plausibility and Answer Analysis

A *Incorrect. Reduction in airborne iodine removal is correct, but the water in the sumps will be LOWER. Plausible if candidate believes that the CSS Chem Add tank is an acidic solution vice a caustic solution.*

B *Correct. The Containment Spray Chem Add tank supplies NaOH to the water being sprayed into containment.*

C *Incorrect. Loss of the Chem Add tank will not affect the peak pressure inside containment. Plausible since the primary function of the Containment Spray system is to reduce pressure in containment, but the Chem Add tank does not affect this function. Sump pH will be lower because of the loss of NaOH. Plausible if candidate believes that the CSS Chem Add tank is an acidic solution vice a caustic solution.*

D *Incorrect. Loss of the Chem Add tank will not affect the peak pressure inside containment. Plausible since the primary function of the Containment Spray system is to reduce pressure in containment, but the Chem Add tank does not affect this function. Sump ph is correct.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Engineered Safety Features Actuation - Knowledge of the effect that a loss or malfunction of the (SYSTEM) will have on the following: (CFR: 41.7 / 45.6) Containment

Importance Rating:	4.3 4.7
Technical Reference:	ESFAS text Tech Spec Bases B 3/4 6-3 (page 452)
References to be provided:	None
Learning Objective:	CSS, Obj. 2b
Question Origin:	New
Comments:	Meets KA since it's asking for knowledge of the effect that a failure of CSAS (an ESFAS signal) has on Containment (ie is it isolated or not).

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	TECH SPEC BASES
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	013K3.03	K/A 2:	

CONTAINMENT VENTILATION SYSTEM (Continued)

gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Fan Coolers are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

The maximum and minimum volumes for the Spray Additive Tank are based on the analytical limits. The specified indicated levels used for surveillance include instrument uncertainties and unusable tank volume.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Fan Coolers ensures that adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

ESW flowrate to the Containment Fan Coolers will vary based on reservoir level. Acceptable ESW flowrate is dependent on the number of heat exchanger tubes in service. Surveillance test acceptance criteria should be adjusted for these factors.

41. 2009A NRC RO 041

Given the following plant conditions:
- The plant is operating at 100% power
- The breaker to MCC 1A34-SA has tripped OPEN

Due to this malfunction, which ONE of the following components has lost power?

A. CRDM Cooling Fan E-80A

B. Shield Cooling Fan S-2A-SA

C.✓ CNMT Fan Cooler AH-3A-SA

D. Rx Support Cooling Fan S-4A-SA

Plausibility and Answer Analysis

A *Incorrect. Power to the fan is supplied from MCC 1A24.*

B *Incorrect. Power to the fan is supplied from MCC 1A21-SA.*

C *Correct. AH-3A-SA is powered from MCC 1A-34-SA and will lose power if the MCC is lost.*

D *Incorrect. Power to the fan is also supplied from MCC 1A21-SA.*

KA Statement - Containment Cooling - Knowledge of electrical power supplies to the following: (CFR: 41.7) Containment cooling fans

Importance Rating:	3.0 3.1
Technical Reference:	OP-169 Rev. 17, Attachment 1 Pages 29-31
References to be provided:	None
Learning Objective:	CCS, Obj. 2
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	CCS TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	022K2.01	K/A 2:	

Attachment 1 - Containment Cooling and Ventilation System Electrical Lineup Checklist
Sheet 3 of 6

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	MCC-1A21-SA (RAB 286)			
1A21-SA-1D	S-4(1A-SA) RX Support Cooling Fan (Both Bkrs.)	ON		
1A21-SA-4C	S-2(1A-SA) Primary Shield Cooling Fan (Both Bkrs.)	ON		
	PP-1A211-SA (RAB 286)			
PP-1A211-SA-33	AH-2(1A-SA) Space Heater	ON		
PP-1A211-SA-35	AH-2(1A-SA) Space Heater	ON		
PP-1A211-SA-34	AH-2(1B-SA) Space Heater	ON		
PP-1A211-SA-36	AH-2(1B-SA) Space Heater	ON		
	PP-1A212-SA (RAB 286)			
PP-1A212-SA-7	AH-3(1A-SA) Space Heater	ON		
PP-1A212-SA-9	AH-3(1A-SA) Space Heater	ON		
PP-1A212-SA-8	AH-3(1B-SA) Space Heater	ON		
PP-1A212-SA-10	AH-3(1B-SA) Space Heater	ON		
PP-1A212-SA-37	S-4(1A-SA) Space Heater	ON		
PP-1A212-SA-39	S-4(1A-SA) Space Heater	ON		
PP-1A212-SA-38	S-2(1A-SA) Space Heater	ON		
PP-1A212-SA-40	S-2(1A-SA) Space Heater	ON		

Attachment 1 - Containment Cooling and Ventilation System Electrical Lineup Checklist
Sheet 4 of 6

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	MCC-1A34-SA (RAB 286)			
1A34-SA-1A	AH-3(1A-SA) Containment Fan Cooler	ON	______	______
1A34-SA-2A	AH-3(1B-SA) Containment Fan Cooler	ON	______	______
	MCC-1B21-SB (RAB 286)			
1B21-SB-4B	S-2(1B-SB) Primary Shield Cooling Fan (Both Bkrs.)	ON	______	______
1B21-SB-4C	S-4(1B-SB) Reactor Support Cooling Fan (Both Bkrs.)	ON	______	______
	PP-1B211-SB (RAB 286)			
PP-1B211-SB-33	AH-1(1A-SB) Space Heater	ON	______	______
PP-1B211-SB-35	AH-1(1A-SB) Space Heater	ON	______	______
PP-1B211-SB-34	AH-1(1B-SB) Space Heater	ON	______	______
PP-1B211-SB-36	AH-1(1B-SB) Space Heater	ON	______	______
	480V BUS 1E1 (RAB 286)			
1E1-7A	AH-37(1B-NNS) Containment Fan Coil Unit	RACKED IN	______	N/A
1E1-7B	AH-38(1B-NNS) Containment Fan Coil Unit	RACKED IN	______	N/A
1E1-7C	AH-39(1B-NNS) Containment Fan Coil Unit	RACKED IN	______	N/A

Attachment 1 - Containment Cooling and Ventilation System Electrical Lineup Checklist
Sheet 5 of 6

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	PP-1B212-SB (RAB 286)			
PP-1B212-SB-7	AH-4(1A-SB) Space Heater	ON	______	______
PP-1B212-SB-9	AH-4(1A-SB) Space Heater	ON	______	______
PP-1B212-SB-8	AH-4(1B-SB) Space Heater	ON	______	______
PP-1B212-SB-10	AH-4(1B-SB) Space Heater	ON	______	______
PP-1B212-SB-37	S-4(1B-SB) Space Heater	ON	______	______
PP-1B212-SB-39	S-4(1B-SB) Space Heater	ON	______	______
PP-1B212-SB-38	S-2(1B-SB) Space Heater	ON	______	______
PP-1B212-SB-40	S-2(1B-SB) Space Heater	ON	______	______
	MCC-1B34-SB (RAB 286)			
1B34-SB-2A	AH-4(1A-SB) Containment Fan Cooler	ON	______	______
1B34-SB-3A	AH-4(1B-SB) Containment Fan Cooler	ON	______	______
	MCC-1A24 (RAB 261)			
1A24-7B	E-80(1A-NNS) Rod Control Drive Mech. Fan (Both Bkrs.)	ON	______	N/A
1A24-7C	E-81(1A-NNS) Rod Control Drive Mech. Fan (Both Bkrs.)	ON	______	N/A

42. 2009A NRC RO 042

Given the following plant conditions:
- The plant is operating at 100% power
- 'A' Containment Spray Pump is running on recirculation per OST-1118, Containment Spray Operability Train A Quarterly Interval Modes 1-4
- A small break LOCA develops in Containment
- The crew performs a manual Reactor Trip and SI
- Containment pressure is currently 3.8 psig and rising

Which ONE of the following describes the 'A' Containment Spray System response to the conditions that have developed?

A. The 'A' Containment Spray Pump is tripped by the Sequencer

B.✓ The 'A' Containment Spray Pump is tripped by the Containment Phase A Isolation (T) signal

C. The 'A' Containment Spray Pump continues to run and the spray header isolation valve will open if Containment pressure exceeds 10 psig

D. The 'A' Containment Spray Pump continues to run, but the spray heaer isolation valve will NOT open if Containment pressure exceeds 10 psig

Plausibility and Answer Analysis

A *Incorrect. The sequencer prevents pump starting with certain conditions but does not trip the pump if currently running for testing. With the sequencer running program B or C, and a CSAS present during the first second of Load Block 2, the CSS pump starts in Load Block 2. If the CSAS occurs during load blocks 2 or 3 the CSS pump start is blocked until Load Block 4. The pump starts upon receipt of CSAS any time after Load Block 4. This prevents starting CSS pumps and other large electrical loads at the same time.*

B *Correct. OST-1118 P&L #1: If a Containment Phase A Isolation (T) Signal is received during the performance of this OST, the following components will realign as stated: Containment Spray Pump 1A-SA will trip,1CT-47, CNMT SPRAY PUMP A-SA RECIRC, will shut and 1CT-24, CONTAINMENT SPRAY EDUCTOR TEST, will shut.*

C *Incorrect. The pump will trip if a Containment Phase A Isolation (T) Signal is received. The spray header isolation valve would open if CSIS were present (HI-3 Containment pressure 2/4 >10 psig) but Containment pressure is only 3.8 psig*

D *Incorrect. Testing the SSPS system will defeat actuation of components in the tested train but testing the Containment Spray System does not have the same feature.*

KA Statement - Containment Spray - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: (CFR: 41.5 / 45.5) Containment Pressure

Importance Rating:	3.9 4.2
Technical Reference:	OST-1118 Rev. 25, page 9
References to be provided:	None
Learning Objective:	CSS, Obj. 6b
Question Origin:	Bank - Modified LOR NRC Bank question A06 016
Comments:	

Origin:	MODIFIED	Cog Level:	H
Difficulty:	3	Reference:	OST-1118
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	026A1.01	K/A 2:	

4.0 PRECAUTIONS AND LIMITATIONS

1. If a Containment Phase A Isolation Signal is received during the performance of this OST, the following components will realign as stated:

   - Containment Spray Pump 1A-SA will trip
   - 1CT-47, CNMT SPRAY PUMP A-SA RECIRC, will shut.
   - 1CT-24, CONTAINMENT SPRAY EDUCTOR TEST, will shut.

2. If a Containment Spray Actuation Signal is received during the performance of this OST, immediately suspend from this test, while maintaining the Containment Spray Pump running and its associated discharge valve open.

3. Do not allow the performance of any evolutions that would affect RWST level. If an unexplained level change occurs, secure testing and investigate.

4. To prevent lifting 1SF-217, Fuel Pool Demin 1&4X Relief Valve, verify the Fuel Pool & Refueling Water Purification System is not aligned to the RWST before running a Containment Spray Pump in recirculation.

5. Observe all radiological controls per AP-535.

6. Do not exceed a motor winding temperature of 266°F.

7. If any valve stroke time falls outside its Code Criteria, the valve will be immediately retested per the retest instructions or declared inoperable.

8. Concurrent Verification is required when operating switches/pushbuttons in SSPS, installing/removing jumpers, or lifting/landing leads.

9. Operators should be familiar with the MSDS sheet for the material being handled. Material Safety Data Sheets (MSDS) may be obtained by calling 3E Company at 1-800-451-8346. CHE-NGGC-0045 provides instructions for obtaining MSDS sheets form 3E Company.

10. Personnel will use chemical suits with hoods when required to wear a chemical suit. If chemicals come in contact with chemical suit and other PPE, then the chemical suite/PPE should be disposed of per Chemistry direction.

11. Personnel will wear a face shield manufactured for "Chemical Protection" when required to wear a face shield.

QUESTIONS REPORT
for LORNRCBNK

16. A06 016/OST-1118/OST-1119/013000K105/4.1/4.4//7/22/96/01/

Assume for this question only that containment spray pump 1ASA had been running, recirculating water from and back to the RWST, for surveillance testing at the start of this transient. How would the Containment Spray System have responded to conditions that have developed?

A. The pump would have continued to run and the spray header isolation valve would have opened when containment pressure exceeded 10 psig.

B. None of the "A" train Containment Spray System components would be affected since automatic signals are defeated during surveillance testing.

C. The pump would have been tripped by the EDG sequencer.

D✓ The pump would have been tripped by the containment phase A isolation (T) signal.

Reference:	OST-1118	Reference:	OST-1119
K/A #:	013000K105	K/A Value:	4.1/4.4
Task #:		Last Used:	7/22/96
Job Class:	01	ReviewDate:	

43. 2009A NRC RO 043

With both Containment Spray Pumps in service, the Spray Pump suctions are __(1)__ aligned to the Containment Sump upon receipt of a __(2)__ condition.

A. (1) manually
(2) low low RWST level

B. (1) manually
(2) high Containment Sump level

C.✓ (1) automatically
(2) low low RWST level

D. (1) automatically
(2) high Containment Sump level

Plausibility and Answer Analysis

A Incorrect. RHR alignment is completed manually but CNMT Spray alignment is completed automatically on RWST low low level.

B Incorrect. RHR alignment is completed manually but CNMT Spray alignment is completed automatically on RWST low low level. EPP-010 does provide a minimum CNMT Sump Level to ensure long term recirc capability but this does not initiate the transfer.

C Correct. With a CNMT Spray Pump Breaker shut and an RWST low low level the suction is automatically aligned to the sump.

D Incorrect. Spray is automatically aligned on RWST low low level. EPP-010 does provide a minimum CNMT Sump Level to ensure long term recirc capability but this does not initiate the transfer.

KA Statement - Containment Spray - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: (CFR: 41.5 / 45.5) ECCS

Importance Rating:	3.9 4.2
Technical Reference:	APP-ALB-004, Rev. 16 page 9 EPP-010 Rev. 24, page 20
References to be provided:	None
Learning Objective:	CSS, Obj. 6d
Question Origin:	BANK OIT Development Bank 026 K1.01 001
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	2	Reference:	EPP-010
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	026K1.01	K/A 2:	

2-4

DEVICES: LS-0990CW **SETPOINT**: 23.4% (144.6 in WC)
LS-0991CW 23.4% (144.6 in WC)
LS-0992CW 23.4% (144.6 in WC)
LS-0993CW 23.4% (144.6 in WC)

REFUELING WATER STORAGE TANK 2/4 LOW LOW LEVEL

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using RWST level indicators LI-990, LI-991, LI-992, and/or LI-993.

2. **VERIFY** Automatic Functions:
   a. With an SI signal and 2/4 low-low levels present, CNMT Sump to RHR Pump suction valves 1SI-300, 1SI-301, 1SI-310, and 1SI-311 open.
   b. With a CNMT Spray pump running and 2/4 low-low levels present, the following valves reposition for the running pump:
      (1) 1CT-105 or 1CT-102, CNMT Sump to CT Pump, opens.
      (2) 1CT-26 or 1CT-71, RWST to CT Pump, shuts.

3. **PERFORM** Corrective Actions:
   a. **IF** alarm is due to SI,
      THEN GO TO EOP-PATH-1. **[Reference 7]**
   b. **DISPATCH** an operator to verify RWST level using LI-7110 (ACP) or LI-7116 (RWST Pit).
   c. **IF** RWST low level is due to a tank or supply line rupture,
      THEN GO TO AOP-008, Accidental Release of Liquid Waste.
   d. **IF** RWST low level is due to makeup to the RCS
      AND RWST level is approaching 12%,
      THEN:
      (1) **REFER TO** OP-107.01, CVCS Boration, Dilution, and Chemistry Control
      (2) **INITIATE** action to fill the RWST.
   e. **IF** neither SI nor CNMT Spray are actuated,
      THEN VERIFY SHUT ALL Recirculation Sump suction valves.
   f. **MONITOR** RWST, RCS, and Fuel Pool levels as necessary.
   g. **IF** maintenance is to be performed,
      THEN GO TO OWP-ESF.

CAUSES:

1. Safety Injection or Containment Spray actuation
2. Tank or supply line rupture below low-low level setpoint
3. RHR or CSS Recirculation Sump suction valves open
4. Use of water to fill Refueling Cavity during refueling
5. 1CT-23, Fuel Pool Cooling System Supply Valve, open
6. Improper valve alignment
7. Instrument or alarm circuit malfunction

REFERENCES:

1. AOP-008, Accidental Release of Liquid Waste
2. EOP-PATH-1
3. OP-107.01, CVCS Boration, Dilution, and Chemistry Control
4. OWP-ESF
5. 6-B-401 0046C, 0416-0419, 0636, 1039-1042
6. ~~Technical Specifications 3.1.2.5, 3.1.2.6, 3.3.2, 3.3.3.6,~~ and 3.5.4
7. FSAR Section 6.3.5

Instructions	Response Not Obtained

NOTE: Additional foldout item, "AFW SUPPLY SWITCHOVER CRITERIA", applies.

Instructions	Response Not Obtained
9. Implement Function Restoration Procedures As Required.	
10. Align CNMT Spray For Recirculation:	
a. Any CNMT spray pump - RUNNING	a. GO TO Step 11.
b. Verify CNMT sump to CNMT spray suction valves - OPEN 1CT-105 1CT-102	
c. Verify RWST to CNMT spray pump suction valves - SHUT 1CT-26 1CT-71	

44. 2009A NRC RO 044

Given the following plant conditions:

- GP-007, Normal Plant Cooldown, is in progress
- RCS Cold Leg Temperature is 255°F
- The BOP has been directed to establish the required Cooldown Rate (CDR) using Condenser Steam Dumps

A maximum CDR of __(1)__ °F/Hour is allowed based on limiting the __(2)__ stresses on the Reactor Vessel Inner Wall.

A. (1) 30
(2) tensile

B. (1) 30
(2) compressive

C.✓ (1) 50
(2) tensile

D. (1) 50
(2) compressive

Plausibility and Answer Analysis

A *Incorrect. 30°F/Hour is from T.S. Table 4.4-6 but this limit is used when less than 120°F. The concern is tensile stress on the inner wall where allowable stress is lowest.*

B *Incorrect. 30°F/Hour is from T.S. Table 4.4-6 but this limit is used when less than 120°F. Compressive stresses will exist during a cooldown, but the concern is tensile stress on the inner wall where allowable stress is lowest.*

C *Correct. 50°F/Hour is the correct Cooldown Limit per T.S. Table 4.4-6. The concern is tensile stress on the inner wall where allowable stress is lowest.*

D *Incorrect. 50°F/Hour is the correct Cooldown Limit per T.S. Table 4.4-6. Compressive stresses will exist during a cooldown, but the concern is tensile stress on the inner wall where allowable stress is lowest.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Main and Reheat Steam - Knowledge of the operational implications of the following concepts as they apply to the (SYSTEM): (CFR: 41.5 / 45.7) Bases for RCS cooldown limits

Importance Rating:	2.7 3.1
Technical Reference:	T.S 3.4.9.2 pages 3/4 4-34,4-35 (pages 260, 261) T.S. Table 4.4-6, page 3/4 4-38 (page 264) T.S. Bases pages B3/4 4-12, 4-13
References to be provided:	None
Learning Objective:	LP-GP-3.7, Obj. 5b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	T.S 3.4.9.2
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	039K5.05	K/A 2:	

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

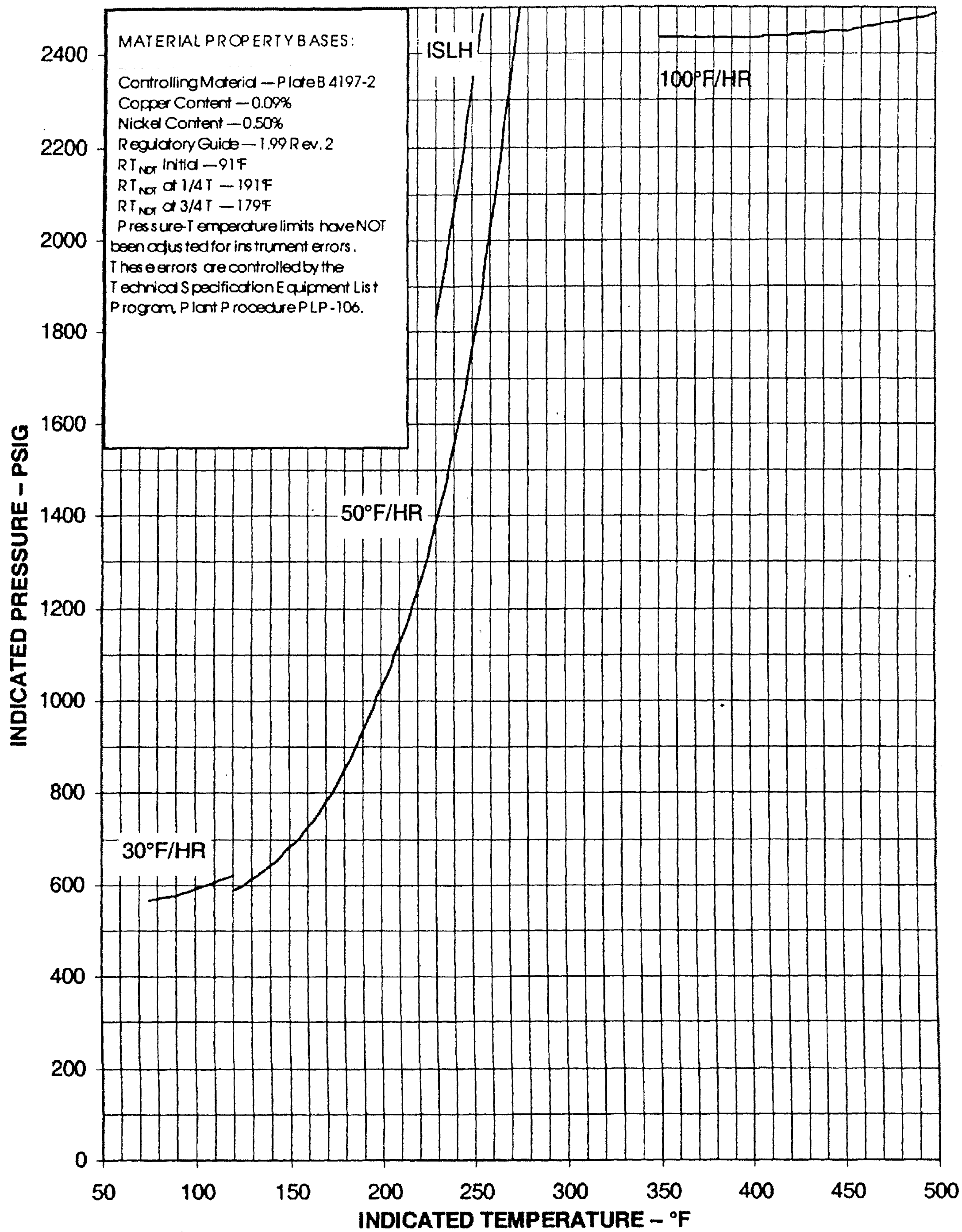


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS – APPLICABLE TO UP TO 36 EFPY

Ref for RO #44, 039K5.05

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD*
350-120°F	50°F
< 120°F	30°F

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD*
<350°F	50°F

*Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

PRESSURE/TEMPERATURE LIMITS (Continued)

metal temperature at that time. K_{IR} is obtained from reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

Vessel regions:

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp [0.02(T-RT_{NDT} + 100°F)] \quad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp [0.0145(T-RT_{NDT} + 160°F)] \quad (1b)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. The pressure stress intensity factors are obtained and allowable pressures are calculated from equation 2.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall and the inlet nozzle corner. During cooldown, the controlling location of the flaw is always at the inside surface because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest. The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell or the inlet nozzle.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T inside surface location is at a higher temperature than the fluid adjacent to the inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside surface. The thermal gradients during heatup produce compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of

45. 2009A NRC RO 045

Given the following plant conditions:
- A reactor trip has just occurred from full power

The following plant conditions currently exist:
- RCS Tavg is slowly increasing
- All SG levels are 28% and slowly decreasing
- Loop Low Tavg Bistable lights are lit on TSLB 3
- Loop Low-Low Tavg Bistable lights are NOT lit on TSLB 3
- The crew has entered EPP-004, Reactor Trip Response

Which ONE of the following describes the status of the Main Feedwater system and the action required to stabilize and maintain Steam Generator level?

A. **ONLY** Main Feed Reg Valves auto shut;
Adjust MDAFW FCVs manually to control level

B.✓ **ONLY** Main Feed Reg Valves auto shut;
Establish Main Feed Reg Bypass Valve flow in Manual

C. Main Feed Reg **AND** Main Feed Reg Bypass Valves auto shut;
Adjust MDAFW FCVs manually to control level

D. Main Feed Reg **AND** Main Feed Reg Bypass Valves auto shut;
Establish Main Feed Reg Bypass Valve flow in Manual

Plausibility and Answer Analysis

A *Incorrect. It is correct that only the MFRVs shut. Auto start of AFW is plausible (and imminent) but incorrect at current SG levels.*

B *Correct. MFRVs will shut automatically due to the P-4 w/ Low Tavg signal. Flow should be established using MFRBVs.*

C *Incorrect. MFRBVs shut on Main Feed Water Isolation Signal but do not shut on P-4 w/Low Tavg. Auto start of AFW is plausible (and imminent) but incorrect at current SG levels.*

D *Incorrect. MFRBVs shut on Main Feed Water Isolation Signal but do not shut on P-4 w/Low Tavg. Flow should be established using MFRBVs.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Main Feedwater - Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) Feed regulating valve controller

Importance Rating:	3.0 2.9
Technical Reference:	EPP-004 Rev. 18, Page 10
References to be provided:	None
Learning Objective:	CFW, Obj. 9b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	2	Reference:	EPP-004
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	059A4.08	K/A 2:	

REACTOR TRIP RESPONSE

Instructions	Response Not Obtained
5. Check Feed System Status:	
a. RCS Temperature - LESS THAN 564°F	a. WHEN RCS temperature less than 564°F, THEN do Steps 5b AND c. Continue with Step 6.
b. Verify feed reg valves - SHUT	
c. Check feed flow to SGs - GREATER THAN 210 KPPH	c. Perform the following: 1) Establish feed flow to SGs using any of the following: o AFW (Refer to OP-137, "AUXILIARY FEEDWATER SYSTEM" operation.) o Main feed flow using the feed reg bypass valves in MANUAL. 2) Maintain total feed flow greater than 210 KPPH until level greater than 25% in at least one SG.

46. 2009A NRC RO 046

Given the following plant conditions:

- Reactor Trip and Safety Injection have occurred from 100% power
- Containment pressure is 3.7 psig
- 'A' SG pressure is 1100 psig
- 'B' SG pressure is 880 psig
- 'C' SG pressure is 990 psig
- 80 KPPH AFW Flow exists to each SG

Which ONE of the following describes the status of AFW Isolation and action required?

An AFW Isolation Signal. . .

A. has NOT occurred and AFW Flow should be controlled to maintain SG Water Levels between 25 and 50%.

B. has NOT occurred and AFW Flow should be controlled to maintain SG Water Levels between 40 and 50%.

C✓ should have occurred on the 'B' SG and AFW Flow to 'B' SG should be isolated by shutting the MOV Isolations.

D. should have occurred on the 'B' and 'C' SGs and AFW Flow to 'B' and 'C' SGs should be isolated by shutting the MOV Isolations.

Plausibility and Answer Analysis

A *Incorrect. Plausible if candidate fails to recognize that a Main Steam Line Isolation should have occurred and therefore a 100 psi difference on 'B' SG will generate an AFW Isolation signal. This answer would be correct if the Main Steam Line Isolation on high containment pressure had not existed.*

B *Incorrect. Plausible if candidate fails to recognize that the setpoint for auto AFW isolation is a 100 psi difference. This answer would be correct if 'B' SG pressure were not 100 psi lower than both other SGs (ie., if 'B' SG pressure was 900 psig).*

C *Correct. When containment increases above 3.0 psig then an automatic Main Steam Isolation signal will be generated. A Main Steam Line Isolation Signal in conjunction with one SG being 100 psi lower than both other SGs will generate an automatic AFW Isolation signal to the low pressure SG.*

D *Incorrect. Plausible if candidate misunderstands the coincidence associated with the automatic AFW isolation signal since 'C' SG is 100 psi lower than the 'A' SG. The coincidence for the Automatic AFW Isolation is 2 out of 3 detectors with 100 psi difference on 2 out of 3 steam generators.*

KA Statement - Auxiliary/Emergency Feedwater - Ability to (a) predict the impacts of the following on the (SYSTEM) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:(CFR:41.5 / 43.5 / 45.3 / 45.13) Air or MOV failure

Importance Rating:	3.4 3.5
Technical Reference:	APP-ALB-014 Rev. 20, Window 2-1A AFW Student Text
References to be provided:	None
Learning Objective:	AFW, Obj. 7h
Question Origin:	New
Comments:	(K/A Match Statement) An AFW Isolation Signal is present on the B SG but the MOVs failed to properly position based on 80 KPPH AFW flow continues to the B SG. This is considered an MOV failure because it will have an adverse affect on the plant conditions. The candidate must recognize the MOVs are out of required position and prescribe the appropriate actions.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	APP-ALB-014
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	061A2.02	K/A 2:	

LOOP B AFW LINE ISOL

DEVICES:
PB474B, PB475B, PB476B
(2 out of 3 logic)

PB484B, PB485B, PB486B
(2 out of 3 logic)

SETPOINT:
P1-P2 100 psid with Main Steam Isolation Signal present

P2-P3 100 psid with Main Steam Isolation Signal present

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PI-474.1 SB, PI-484.1 SB, PI-494.1 SB, Steam Pressure indicators
   b. Valve position indication for SG B AFW isolation valves 1AF-93, 1AF-143, 1AF-51, 1AF-130
   c. TSLB-1 status lights

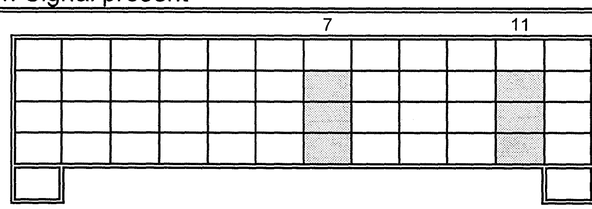
2. **VERIFY** Automatic Functions:
   a. SG B AFW isolation valves 1AF-93, 1AF-143, 1AF-51, 1AF-130 shut

3. **PERFORM** Corrective Actions:
   a. **IF** Reactor trips,
   THEN GO TO EOP-PATH-1.

NOTE

B SG AFW Isolation signal will result if all of the following conditions are met:
- Any two lights in TSLB-1 column 7 lit
- Any two lights in TSLB-1 column 11 lit
- Main Steam Isolation Signal present

TSLB-1



   b. **IF** the AFW isolation signal is required,
   THEN VERIFY SHUT 1AF-93, 1AF-143, 1AF-51, and 1AF-130.
   c. **IF** the AFW isolation signal is NOT required,
   THEN RESET AND OPEN 1AF-93, 1AF-143, 1AF-51, and 1AF-130 as desired.

CAUSES:

1. Main Steam Isolation Signal with a high SG differential pressure received
2. Control failure
3. Alarm circuit or instrument malfunction

REFERENCES:

1. Technical Specification 3.7.1.2
2. EOP-PATH-1
3. 5-S-0544
4. 6-B-401 0636
5. 1364-00870, 02776 sheet 33

47. 2009A NRC RO 047

Given the following plant conditions:
- The plant is operating at 35% power
- The 6.9 KV Aux buses are being supplied by the UATs
- Breaker 52-7 is open for inspection by transmission personnel

Current conditions
- Breaker 52-9 trips open on a fault
- The crew enters AOP-015, Secondary Load Rejection, and stabilizes the plant

Which ONE of the following describes the effect of this on the automatic fast bus transfer and manual transfer of the 6.9 KV Aux Buses?

If a Generator Lockout occurs, an automatic fast bus transfer to the SUT __(1)__ occur. 6.9 KV Aux buses __(2)__ be transferred manually by the operator.

A. (1) will
(2) can

B. (1) will
(2) can NOT

C✓ (1) will NOT
(2) can

D. (1) will NOT
(2) can NOT

Plausibility and Answer Analysis

A *Incorrect. (1) is incorrect and (2) is correct for the given conditions. This is plausible for (1) because it is normal operation of the fast transfer if the Generator Lockout occurs but either 52-7 OR 52-9 must be closed to satisfy the logic.*

B *Incorrect. (1) and (2) are incorrect for the given conditions. This is plausible for (1) because it is normal operation of the fast transfer if the Generator Lockout occurs but either 52-7 OR 52-9 must be closed to satisfy the logic. This is plausible for (2) because the manual transfer to the UATs require either 52-7 OR 52-9 to be closed to satisfy the logic but the manual transfer to the SUTs does not.*

C *Correct. (1) and (2) are correct for the given conditions. For (1), because neither 52-7 OR 52-9 are closed and the fast transfer logic is not satisfied. For (2), because the status of 52-7 and 52-9 is not part of the logic.*

D *Incorrect. (2) is incorrect and (1) is correct for the given conditions. This is plausible for (2) because the manual transfer to the UATs require either 52-7 OR 52-9 to be closed to satisfy the logic but the manual transfer to the SUTs does not.*

KA Statement - AC Electrical Distribution - Knowledge of (SYSTEM) design feature(s) and or interlock(s) which provide for the following: (CFR: 41.7) Interlocks between automatic bus transfer and breakers

Importance Rating:	2.9 3.1
Technical Reference:	6.9KV Student Text page 16 and 17 give the logic for these bus transfers. (These have been confirmed to be accurate using CAR 2166-G-0040)
References to be provided:	None
Learning Objective:	6.9KV, Obj. 8
Question Origin:	New
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	6.9 KV STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	062K4.03	K/A 2:	

Overvoltage relays (59 UTAX and 59 UTAY) provide alarms but do not trip the transformer lockout relays.

6.9 kV Switchgear

The 6.9 kV switchgear is manufactured by Siemens-Allis Corporation. Buses 1A, 1D, 1B, and 1E are rated at 3000 amperes. Bus 1C is rated at 1200 amperes. General Service bus 1-4A is rated at 2000 amperes.

Safety related buses 1A-SA and 1B-SB are rated at 1200 amperes. The 6.9 kV switchgear is protected against bus faults by differential relays (87) that trip each incoming bus breaker (and the Emergency Diesel Generator for 1A-SA and 1B-SB) in the event of a fault on the switchgear bus. All outgoing feeders from the 6.9 kV switchgear are protected against feeder short circuit by overcurrent relays (50/51) or (51) in each phase. Ground protection relays (64) provide alarms only.

All energized parts of the 6.9 kV switchgear are enclosed within grounded metal barriers. All primary bus work and joints are completely enclosed with insulating material. The switchgear is arranged with the circuit breaker's drawout compartment behind the hinged instrument panel. The 6.9 kV circuit breakers are rated at 1200, 2000, or 3000 amperes depending upon the application. All 6.9 kV circuit breakers are controlled at the Main Control Board except for non-safety station service transformer feeders, General Service Bus feeder breakers 501 and 503, and the General Service Bus crosstie breaker 502. Breakers 501, 502, & 503 are controlled locally at the General Service Bus. Control of several breakers, directly related to powering safety-related buses 1A-SA and 1B-SB, is also available from the Emergency Diesel Generator Control Panels upon actuation of ACP transfer relays.

There are two types of special limit switches typically seen in 6.9kV switchgear: Mechanism-Operated Cell switches (MOC), and Truck-Operated Cell switches (TOC). These limit switch contacts are used within the breaker control circuit. The MOC switch contacts (Figure 4) operate by following breaker position only when the breaker is in the "OPERATE" position. There are two sets of TOC switch contacts: one set is actuated when the breaker is in the "TEST" position, and the other is actuated in the "OPERATE" position. When the breaker is in "TEST", the associated TOC contacts are shut and the "OPERATE" TOC contacts are open. This enables local trip and closing capability of the breaker, inhibiting remote operation. The opposite is also true. When the breaker is in the "OPERATE" position, the "TEST" TOC contacts are open, thereby removing local breaker operating capability.

Ref for RO #47, 062K4.03

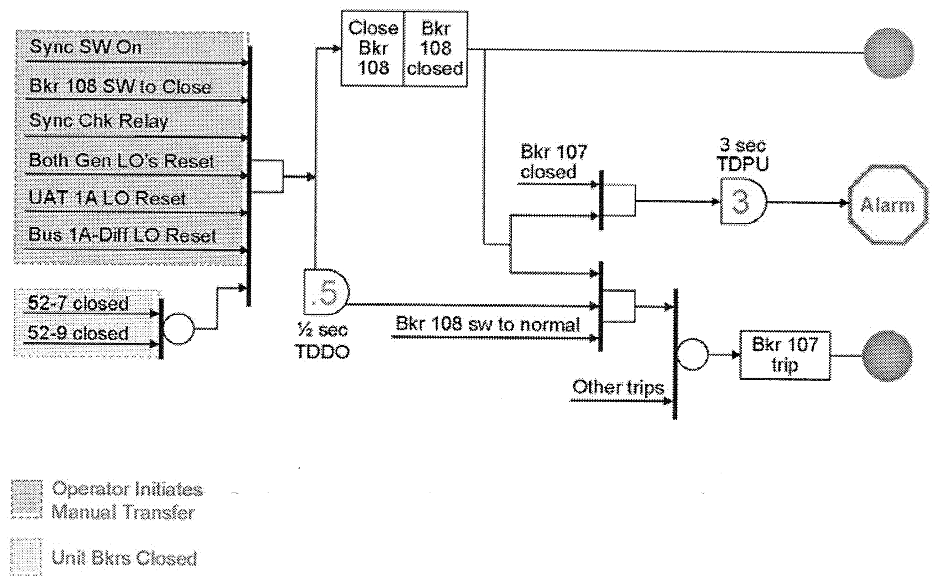
CHAPTER 3

System Operation

Modes of Operation

NORMAL OPERATIONS

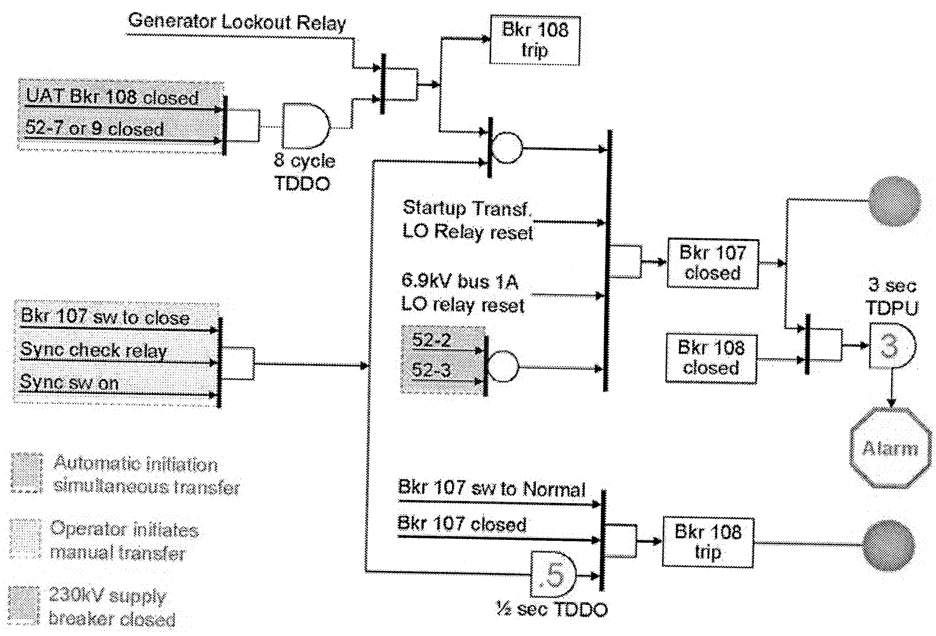
Figure 7: SUT to UAT Transfer Logic



During plant shutdown and start-up, all loads are powered through the Start-Up Transformers (SUT). When the main generator is connected to the grid and is producing sufficient power to carry the plant loads, the loads are manually shifted to the Unit Auxiliary Transformers (UAT). The feeder breakers that connect the Start-Up Transformers and Unit Auxiliary Transformer to the same 6.9 kV bus are interlocked with each other so that when one breaker is closed during a power shift the other will open when the control switch is returned to normal. As long as the switch is held in the closed position, the original supply breaker will not open. Holding the control switch CLOSED for longer than three seconds causes a MCB annunciation. See Figure 7 for details on the transfer logic.

Ref for RO #47, 062K4.03

Figure 8: UAT to SUT Transfer Logic



When the plant is transitioning from an operational mode to a shutdown mode, the 6.9 kV buses are transferred from the Unit Auxiliary Transformer power source to the Start-Up Transformer power source.

AFTER A REACTOR OR TURBINE GENERATOR TRIP

An emergency bus transfer from UAT to SUT occurs on loss of the Main Generator. This logic is detailed in Figure 8. This feature is an automatic fast bus transfer, initiated by the Main Generator lockout relays, which simultaneously closes the SUT feeder breakers and open the UAT feeder breakers for 6.9 kV buses 1A, 1B, 1D and 1E. The bus transfer is completed within a few cycles. If the standby (preferred) power source is not available, as monitored by switchyard breaker position and SUT lockout relays, then transfer will not occur, resulting in loss of power to 6.9 kV buses 1A, 1B, 1D and 1E (and the non-safety-related 6900/480-V station service transformers).

LOSS OF POWER

Loss of power to 1D and 1E will cause the bus tie breakers to the emergency 6.9 kV buses (1A-SA and 1B-SB) to trip, which isolates the safety-related buses from the non-safety-related buses. The loss of voltage on 1A-SA and 1B-SB will cause the associated Emergency Diesel Generator (EDG) to start and will shed all loads (except small static loads) from the emergency buses. Emergency loads are started in sequential load blocks in accordance with the Emergency Safeguards Sequencer when the EDG has attained rated speed and voltage and the associated feeder breaker (BKR 106/126 for 1A-SA/1B-SB) is shut.

48. 2009A NRC RO 048

Which ONE of the following answers the statement below regarding power to the Reactor Trip Breakers?

Each Reactor Trip Breaker undervoltage (UV) coil is powered by __(1)__.
Each shunt coil is powered by __(2)__.

A. (1) 48 VDC from SSPS
(2) 48 VDC from SSPS

B.✓ (1) 48 VDC from SSPS
(2) 125 VDC from a DC Vital bus

C. (1) 125 VDC from a DC Vital bus
(2) 48 VDC from SSPS

D. (1) 125 VDC from a DC Vital bus
(2) 125 VDC from a DC Vital bus

Plausibility and Answer Analysis

A *Incorrect. UV coil is powered by SSPS. Shunt coil is not. Plausible since most Reactor Protection functions come from Solid State Protection.*

B *Correct. The UV coil receives it's power directly from SSPS. The shunt coil is powered from the 125VDC Safety bus.*

C *Incorrect. Plausible because it is exactly the opposite of actual configuration.*

D *Incorrect. Plausible since 125 VDC is the most common power supply to controlling coils.*

KA Statement - DC Electrical Distribution - Knowledge of electrical power supplies to the following: (CFR: 41.7) Major DC loads

Importance Rating:	2.9 3.1
Technical Reference:	CWD 6-B-401 0091 through 0095 RPS Student Text, page 6-7
References to be provided:	None
Learning Objective:	RPS, Obj. 2
Question Origin:	Bank (Direct) OIT Development Bank 001 K2.02
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	3	Reference:	RPS STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	063K2.01	K/A 2:	

CHAPTER 2

Major Components

SSPS CABINETS

The SSPS performs the following three functions:

- Generates an automatic reactor trip when plant conditions exceed any reactor trip setpoint.
- Generates an ESF actuation when plant conditions exceed any ESFAS actuation setpoint.
- Provides multiplexed information to the main control board and ERFIS computer for indications of system status.

The SSPS cabinets for each train are located in the Auxiliary Electrical Equipment Room #1 inside the Reactor Auxiliary Building on elevation 305'. The SSPS is a dual train redundant system, consisting of 2 four-bay cabinets, one single bay control board demultiplexer cabinet, and a computer mounted demultiplexer assembly. Each of the four-bay cabinets is composed of an input relay bay, a logic bay, and two output relay bays. Figure 1 shows the simplified interface diagram of the SSPS.

SSPS INPUT RELAY BAY

The SSPS input relays receive inputs from Process Instrumentation Cabinet (PIC) bistables, Nuclear Instrument System (NIS) bistables, and field contacts. When a trip or actuation signal is received from one of the bistable or field contact inputs, the associated input relay de-energizes (in most cases), which closes a contact to provide the trip or actuation signal to the logic cabinet. The relays also provide isolation between the SSPS and plant instrumentation, and isolation between SSPS trains.

The input relay bay for each train contains approximately 180 general purpose relays. Their coils are 118 VAC or 24 VDC, energized by remote bistables and contact closures. For remote contact closures, power for the coils is provided from within the input relay cabinets. The 24 VDC relays are supplied from the PIC cabinets. For NIS associated input relays, 118 VAC is supplied from NIS cabinet control power.

Each input relay bay is divided into four compartments (one per channel), each containing approximately forty-five relays. Wireway access to these compartments is designed so that each of the four wireways, running from the top to the bottom of the sides of the bay, opens only into one of the compartments. Contacts to the relay coils from plant inputs are made through terminal boards located on the walls of the compartments. The logic signal connections to the relays feeding the logic bay are made at the rear of the compartments.

The front of each input bay contains an indicating light for the AC input voltage and indicating fuses. 120 VAC power to the 4 channels is supplied as follows:

- 1DP-1A-SI Channel I (color code Red)
- 1DP-1B-SII Channel II (color code White)
- 1DP-1A-SIII Channel III (color code Blue)
- 1DP-1B-SIV Channel IV (color code Yellow)

SSPS LOGIC BAY

The SSPS logic bay performs the following functions:

- Receives input from the input relays and determines if the required actuations logic (coincidence) is met for a reactor trip or ESF signal. If the required coincidence is met, the logic bay initiates the reactor trip or ESF signal.
- Receives inputs from MCB switches for manual ESF actuation/reset and for blocking or unblocking specific trip or ESF functions when the associated permissive is satisfied.
- Produces multiplexed status signals that are sent to the MCB and ERFIS demultiplexers.
- Provides a means to test logic functions (including coincidence) for reactor trip and ESF functions while the plant is operating.

The Logic Bay contains the logic tester, spray test panel, logic card cage, DC power supplies, and a connector panel. The Logic Card Cage is divided into five levels containing seven types of printed circuit boards, some of which will be discussed later in this text. Power for the circuitry of each train is taken from 2 of the 4 vital instrument buses through redundant 48 VDC and 15 VDC power supplies at the top of the cabinet. Train A power supplies are powered from 1DP-1A-SI and 1DP-1A-SIII. Train B power supplies are

49. 2009A NRC RO 049

Given the following plant conditions:
- The plant is operating at 100% power
- A loss of DC Bus 1A-SA occurs

Which ONE of the following answers the statement below describing the effects on the 'A' EDG?

The EDG Output breaker __(1)__ closed from the MCB.
The Governor and Generator Excitation circuits will be __(2)__.

A. (1) can be
(2) unaffected

B. (1) can be
(2) deenergized

C. (1) can NOT be
(2) unaffected

D✓ (1) can NOT be
(2) deenergized

Plausibility and Answer Analysis

A *Incorrect. Plausible if the candidate believes that the Governor, Excitation, and Control power circuits are supplied by the AC Electrical Distribution System.*

B *Incorrect. The Governor and Excitation circuits will de-energize, however the EDG Output breaker can not be operated from the MCB as it normally is without DC Control Power.*

C *Incorrect. Plausible if the candidate believes that the Governor and Excitation circuits are supplied by the AC Electrical Distribution System.*

D *Correct. The EDG output breaker can not be operated from the MCB as normal and the Governor and Excitation circuits will be de-energized.*

QUESTIONS REPORT
for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - DC Electrical Distribution - Knowledge of the effect that a loss or malfunction of the (SYSTEM) will have on the following: (CFR: 41.7 / 45.6) ED/G

Importance Rating:	3.7 4.1
Technical Reference:	AOP-025 Rev. 25, pages 42 & 43 OP-155 Rev. 45, pages 117 & 118 EDG Student Text
References to be provided:	None
Learning Objective:	DE, Obj. 11
Question Origin:	BANK OIT Development Bank 063 K3.01
Comments:	K/A match. Question was used on 2007 NRC RO Exam. (Cosmetic changes only) Changed loss of DC bus from '1B-SB' to '1A-SA' and affects from 'B' EDG to 'A' EDG. Reformated stem and changed order of all answers to match rewording.

Origin:	PREVIOUS NRC	Cog Level:	F
Difficulty:	3	Reference:	OP-155
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	063K3.01	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.3 Loss of DP-1A-SA Emergency DC Bus (125V)	
☐ 4. **MONITOR** inverter output voltage and instrument readings for indication of proper voltage on instrument bus.	
☐ 5. **CHECK** power to all instrument buses is ON.	☐ 5. **REFER TO** AOP-24, Loss of Uninterruptible Power Supply.
☐ 6. **DISPATCH** operator to inspect the following for fault conditions: • DP-1A-SA Emergency DC Bus • 125V Battery Chargers 1A-SA AND 1B-SA	
☐ 7. **CONTACT** Maintenance as necessary to initiate repair or other corrective actions.	

NOTE

- Loss of DP-1A-SA will result in all equipment on that side becoming inoperable from the loss of EDG and Load Sequencer or DC control power.
- Local manual operation will be necessary for any breakers that have lost DC control power. **[C.1]**
- Loss of A-SA Emergency DC Bus will make the Turbine Driven AFW Pump inoperable due to loss of power to 1MS-70.

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ 8. **OPEN** all load breakers on DP-1A-SA.	
9. **WHEN** conditions causing loss of DP-1A-SA are corrected, **THEN PERFORM** the following:	
☐ a. **RESTORE** power to DP-1A-SA from the appropriate battery charger(s) per OP-156.01, DC Electrical Distribution.	

(Continued Next Page)

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.3 Loss of DP-1A-SA Emergency DC Bus (125V)

9. (continued)

☐ b. **CLOSE** load breakers one at a time on DP-1A-SA while monitoring battery charger(s) for overload.

☐ c. **VERIFY** proper ventilation through the battery room.

☐ d. **MONITOR** battery charger output current during battery re-charge for indications of possible overload.

NOTE

DP-1A-SA supplies the EDG governor and generator excitation control circuits.

☐ 10. **REFER TO** Attachment 2 **AND VERIFY** any equipment affected by the loss of power is returned to normal line-up.

☐ 11. **WHEN** required repairs are completed, **THEN PERFORM** the following:

a. **RETURN** affected equipment to normal

b. **VERIFY** operability as required.

☐ 12. **EXIT** this procedure.

--END OF SECTION 3.3--

Attachment 4 - Emergency Diesel Generator 1A-SA Electrical Lineup Checklist

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	MCC 1D13			
1D13-3C	Emer DG 1A-SA Lube Oil Heater	ON	______	______
1D13-3D	Emer DG 1A-SA Jacket Water Heater	ON	______	______
	MCC 1A35-SA			
1A35-SA-1B	Diesel F.O. Xfer Pump 1A-SA	ON	______	______
	PP 1D131			
PP-1D131-3	Engine 1ASA Control Panel	ON	______	______
PP-1D131-5	Starting Air Dryer 1A-SA	ON	______	______
PP-1D131-7	Starting Air Dryer 1B-SA	ON	______	______
PP-1D131-9	Gen. Control Pnl Receps & Lts (1A-SA Panel Convenience Lights)	ON	______	______
PP-1D131-11	Gen. Control Pnl Sp Htr (1A-SA Panel Space Htrs)	ON	______	______
	DP-1A1-SA			
DP-1A1-1	Engine 1ASA Control Panel	ON	______	______
DP-1A1-2	Engine 1ASA Control Panel	ON	______	______
DP-1A1-5	Generator 1ASA Control Panel	ON	______	______
DP-1A1-8	EDG 1A-SA Monitoring Panel Breaker	ON	______	______
DP-1A1-13	Generator 1ASA Control Panel	ON	______	______

Attachment 4 - Emergency Diesel Generator 1A-SA Electrical Lineup Checklist

Sheet 4 of 8

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
	DP-1A-SA			
DP-1A-SA-15	Isolation Cabinet 1A (1A1)	ON		
DP-1A-SA-21	Isolation Cabinet 2A (2A1)	ON		
	DP-1A11 286 (TB South)			
DP-1A11-8	Diesel Engine Control Panel 1A	ON		
	PP-1A211-SA			
PP-1A211-SA-10	Bus 1A-SA Cub. 10 Space Heater	ON		
	1A-SA			
1A-SA-13	Generator Space Htr	ON		
1A-SA-13	Generator Output Breaker (Bkr 106)	RACKED IN		
1A-SA-13	Generator Output Breaker (Bkr 106) Control Power	CLOSED		
	Main Control Board (MCB)			
N/A	DIESEL GEN A-SA VOLTAGE	A-B or B-C or C-A		
N/A	DIESEL GEN A-SA CURRENT	A or B or C		
N/A	DIESEL GEN A-SA STOP	NEUTRAL		
N/A	DIESEL GEN A-SA MANUAL VOLT ADJUST	NEUTRAL		

50. 2009A NRC RO 050

Given the following plant conditions:
- The plant is operating at 25% power
- 'A' EDG is operating in parallel with the grid for surveillence testing
- A loss of off-site power occurs

Which ONE of the following describes the response of the 'A' EDG output breaker and the mode in which the diesel will be running after the event?

	EDG Output Breaker	EDG Mode
A.	Remains closed	Droop mode
B.	Remains closed	Isochronous mode
C.	Opens and then recloses	Droop mode
D.✓	Opens and then recloses	Isochronous mode

Plausibility and Answer Analysis

A *Incorrect. The breaker will open. Droop mode is incorrect. This is the mode used when operating in parallel with the grid.*

B *Incorrect. The breaker will open. Plausible because the breaker is already closed. Isochronous is the correct mode.*

C *Incorrect. Plausible since the EDG breaker operation is correct and droop mode is the current mode of operation, but once the EDG is no longer in parallel with the grid (LOSP is in progress) then the mode will shift automatically to Isochronous.*

D *Correct. DG Sequencer will initiate a trip of the DG output breaker when off-site power is lost. Once the breaker is open, the sequencer starts its process for reclosing the breaker. Isochronous is the correct mode.*

KA Statement - Emergency Diesel Generator - Ability to monitor automatic operations of the (SYSTEM) including: (CFR: 41.7 / 45.5) Rpm controller/megawatt load control (breaker-open/breaker-closed effects)

Importance Rating:	3.0 2.9
Technical Reference:	OP-155 Rev. 45, P&L 24, page 9 EDG Student Text
References to be provided:	None
Learning Objective:	DE, Obj. 4c
Question Origin:	Modified from 2007 NRC (013K1.12 02) RO #12
Comments:	

Origin:	MODIFIED	Cog Level:	H
Difficulty:	3	Reference:	OP-155
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	064A3.13	K/A 2:	

4.0 PRECAUTIONS AND LIMITATIONS (continued)

21. Verify UPP-1A is aligned to its normal power source (the inverter) before EDG parallel operation with offsite power. If UPP-1A cannot be aligned to its normal power source, EDG parallel operation with offsite power may proceed if the EDG is declared inoperable.

22. Verify that annunciator window ALB-15-4-1, "ISOL CABINET TRAIN A/B DOOR OPEN OR POWER FAILURE" is unlit before EDG parallel operation with offsite power. If lit, EDG parallel operation with offsite power may proceed if the EDG is declared inoperable during the parallel operation.

R 23. An EDG should only be operated in parallel with offsite power for short periods of time for the sole purpose of EDG testing. (Reference 2.8.1)

24. If a loss of offsite power (LOOP) occurs while an EDG is paralleled to the grid, breakers 105 (125) and 106 (126) should automatically trip open, which will leave the diesel running unloaded. Breaker 106 (126) should then automatically reclose and the sequencer start to load. If breaker 106 (126) fails to open, operator action is required to manually open the breaker.

    The LOOP signal to open breakers 105 (125) and 106 (126) is generated by:

    Both breakers 101 (121) and 102 (122) open

    OR

    Breaker 101 (121) open and either main generator lockout tripped

    Indication that this circuit has actuated properly include proper sequencer operation, or indication that the EDG is running in the emergency mode.

    If these conditions occur while the EDG is paralleled to the grid, and breaker 106 (126) fails to open, (that is, no indication of proper actuation as described above exists), the breaker must be manually tripped from the MCB or, if control is transferred to the local Generator Control Panel (GCP), from the GCP. Since occurrence of a LOOP may not be obvious at the GCP, action to trip breaker 106 (126) from the GCP must be directed from the Main Control Room.

25. The automatic closure of breaker 106 (126) on a LOOP is a one shot signal. Further automatic closure is blocked by the UVX relay of the sequencer. If breaker 106 (126) is inadvertently opened after sequencer actuation, manual operation will be required to open breakers that have been sequenced on and reclosing breaker 106 (126).

R 26. If an EDG is operating in parallel with off-site power and the off-site power reliability or performance becomes questionable, disconnect the EDG from the off-site power source. (Reference 2.8.1)

8.1.2 Procedural Steps (continued)

NOTE: EI-6953A SA (B SB) has a scaling factor of x300. Therefore the range of the meter reflects an actual voltage range of -9000 VAC to +9000 VAC.

4. **POSITION** DIESEL GEN A-SA (B-SB) AUTO VOLTAGE ADJUST control switch to RAISE or LOWER as necessary to adjust EDG voltage to match the associated Emergency 6.9KV Bus voltage as indicated by zero differential voltage indicated on EI-6953A SA (B SB), A (B) SYNC Δ VOLTS. _____

5. **POSITION** DIESEL GEN A-SA (B-SB) GOVERNOR CONTROL switch to RAISE or LOWER as necessary to adjust EDG speed until the synchroscope is rotating slowly in the FAST direction (CLOCKWISE). _____

6. **CHECK** SYNCHRONIZING LIGHTS are cycling (out when the synchroscope is at the 12 o'clock position) in agreement with the synchroscope rotation. _____

7. As necessary, **READJUST** EDG voltage on the MCB to zero differential voltage as indicated on EI-6953A SA (B SB), A (B) SYNC Δ VOLTS. _____

CAUTION

EDG switches from Isochronous to Droop mode of operation upon paralleling. Depending on reactive load on the grid an EDG load decrease may occur. Completion of the next three steps should be expedited to prevent an EDG reverse power trip due to low diesel loading.

8. **WHEN** the synchroscope reaches the 12 o'clock position and the SYNCHRONIZING LIGHTS are TOTALLY DARK, **THEN PLACE** EMERGENCY BUS A-SA TO AUX BUS D TIE BREAKER 105 SA (EMERGENCY BUS B-SB TO AUX BUS E TIE BREAKER 125 SB) breaker control switch to *CLOSE*. _____

9. **PLACE** EMERGENCY BUS A-SA TO AUX BUS D SYNCHRONIZER (EMERGENCY BUS B-SB TO AUX BUS E SYNCHRONIZER) Switch in OFF. _____

10. **IF** EDG load has dropped below 2.2 MW,
THEN re-establish an EDG load of 2.2 to 2.4 MW, while maintaining the ratio of MW to MVARs per Attachment 9. _____

11. **BEGIN UNLOADING** the EDG per Section 7.1 starting at Step 7.1.2.2. _____

QUESTIONS REPORT
for NRC 2007 & 2008

2. 013 K1.12 002/BANK/HARRIS/HIGHER//RO/HARRIS/7/2007/YES

Given the following:

- The unit is operating at 25% power.
- Emergency Diesel Generator (EDG) 1A-SA is loaded to 3800 KW while operating in parallel with the grid during the performance of OST-1013, 1A-SA Emergency Diesel Generator Operation.
- A loss of off-site power occurs.

Which ONE (1) of the following describes the response of the EDG output breaker and how bus loads will be restored?

EDG output breaker...

A. remains closed;

verify automatic load sequencing occurs.

B. remains closed;

loads must be manually started as required.

C.✓ opens and then recloses;

verify automatic load sequencing occurs.

D. opens and then recloses;

loads must be manually started as required.

C is correct. DG Sequencer will initiate a trip of the DG output breaker when off-site power is lost. Once the breaker is open, the sequencer starts its process for reclosing the breaker and placing appropriate loads on the bus

A is incorrect because the breaker will open. Plausible because the breaker is already closed, and all actions are correct except for the EDG breaker

B is incorrect because the breaker will open. This action may have to be taken only if additional failures are present, such as the sequencer not performing its function

D is incorrect. Plausible since the EDG was running in the first place. The EDG would have to be reset if the emergency stop feature was used. In this case, an auto start would be generated by the sequencer

QUESTIONS REPORT
for NRC 2007 & 2008

Knowledge of the physical connections and/or cause-effect relationships between the ESFAS and the following systems: ED/G

Question Number: 12

Tier 2 Group 1

Importance Rating: RO 4.1

Technical Reference: OP-155
Proposed references to be provided to applicants during examination: None
Learning Objective: EDG text obj 8
10 CFR Part 55 Content: 41.7

Comments:
Modified distractors

Category 1:	BANK	Category 2:	HARRIS
Category 3:	HIGHER	Category 4:	
Category 5:	RO	Category 6:	HARRIS
Category 7:	7/2007	Category 8:	YES

51. 2009A NRC RO 051

The Main Control Room receives the following reports from the field:
- Specific gravity of the fuel oil in both Fuel Oil Day Tanks is .835
- 'A' EDG Fuel Oil Day Tank indicated level is 47%
- 'A' EDG Fuel Oil Storage Tank indicates 90,000 gallons
- 'B' EDG Fuel Oil Day Tank indicated level is 42%
- 'B' EDG Fuel Oil Storage Tank indicates 110,000 gallons

Which ONE of the following is the current OPERABILITY status of the Emergency Diesel Generators? (**Reference provided**)

	'A' EDG	'B' EDG
A.	OPERABLE	OPERABLE
B.	OPERABLE	INOPERABLE
C✓	INOPERABLE	OPERABLE
D.	INOPERABLE	INOPERABLE

Plausibility and Answer Analysis

Tech Spec 3.8.1.1

b.1. A separate day tank containing a minimum of 1457 gallons of fuel, (IAW Curve D-X-20, the minimum MCB level indication for a specific gravity of .835 is ~41.6%)

b.2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel

A *Incorrect. The 'A' EDG is not operable, the requirement for DFOST level is >100,000 gallons. 'B' EDG is operable. Though the 'B' EDG Fuel Oil Day Tank level is less than the alarm setpoint (45.4%), it is still above the Tech Spec Limit of Curve D-X-20.*

B *Incorrect. The 'A' EDG is not operable, the requirement for DFOST level is >100,000 gallons. The 'B' EDG is operable. Though the 'B' EDG Fuel Oil Day Tank level is less than the alarm setpoint (45.4%), it is still above the Tech Spec Limit of Curve D-X-20.*

C *Correct. The 'A' EDG is inoperable due to low DFOST level <100,000 gallons. The 'B' EDG is operable. Though the 'B' EDG Fuel Oil Day Tank level is less than the alarm setpoint (45.4%), it is still above the Tech Spec Limit of Curve D-X-20.*

D *Incorrect. The 'A' EDG is inoperable due to low DFOST level <100,000 gallons. The 'B' EDG is operable. Though the 'B' EDG Fuel Oil Day Tank is only below the alarm setpoint (45.4%), it is still above the Tech Spec Limit of Curve D-X-20.*

KA Statement - Emergency Diesel Generator - Knowledge of the effect that a loss or malfunction of the following will have on the (SYSTEM): (CFR: 41.7 / 45.7) Fuel oil storage tanks

Importance Rating:	3.2 3.3
Technical Reference:	Curve D-X-20 Rev 1 (provided to students) T.S. 3.8.1.1 page 3/4 8-1 (page 331) APP-ALB-025 Rev. 11, Window 4-3
References to be provided:	Provide curve D-X-20 Rev 1
Learning Objective:	DE, Obj. 13a
Question Origin:	New
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	T.S. 3.8.1.1
~~Ref. Provided?:~~	~~YES~~	~~Key Words:~~	~~2009A NRC RO~~
K/A 1:	064K6.08	K/A 2:	

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. A separate day tank containing a minimum of 1457 gallons of fuel,

   2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

   3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable:

   1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

   2. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

   3. Verify required feature(s) powered from the OPERABLE offsite A.C. source are OPERABLE. If required feature(s) powered from the OPERABLE offsite circuit are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 24 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

DEVICES:
LS-01FO-2462BS
LS-01FO-2462BS

SETPOINT:
95 IABT (90.8%)* Hi-Level
47.5 IABT (45.4%)* Lo-Lo Level
* MCB indicated level related to inches above bottom of tank (IABT) and assuming a specific gravity of 0.85 for the fuel oil

REFLASH: NO

DIESEL B DAY TANK HI- / LO-LO LEVEL

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. LI-2461B, Day Tank B-SB Level.
   b. 1DFO-191, Fuel Oil to Day Tank position indication.
   c. FOTP B-SB status indication.
2. **VERIFY** Automatic Functions: None
3. **PERFORM** Corrective Actions:

NOTE
Failure of Level Switch LS-2463B will inhibit both manual and auto starts of the FOTP.

   a. **IF** HI level exists,
      THEN:
      (1) **OPERATE** FOTP B-SB manually to maintain level between 48% and 84%.
      (2) **IF** Fuel Oil Day Tank B-SB has overflowed and caused a spill,
          THEN REFER TO PLP-500, Hazardous Substances Spill Notification, Oil Spill Notification, and Fish Kill Reporting.
   b. **IF** LO-LO level exists,
      THEN:
      (1) **VERIFY** 1DFO-191, Fuel Oil to Day Tank is OPEN.
      (2) **IF** FOTP B-SB has failed to start **AND** white light is on at controller,
          THEN RESET breaker at MCC 1B35-SB-3C per OP-156.02, AC Electrical Distribution.
      (3) **IF** FOTP B-SB has failed to start **AND** white light is off at controller,
          THEN OPERATE pump manually to maintain level between 48% and 84%.
   c. **IF** maintenance is to be performed,
      THEN REFER TO OWP-DG, Diesel Generator System.

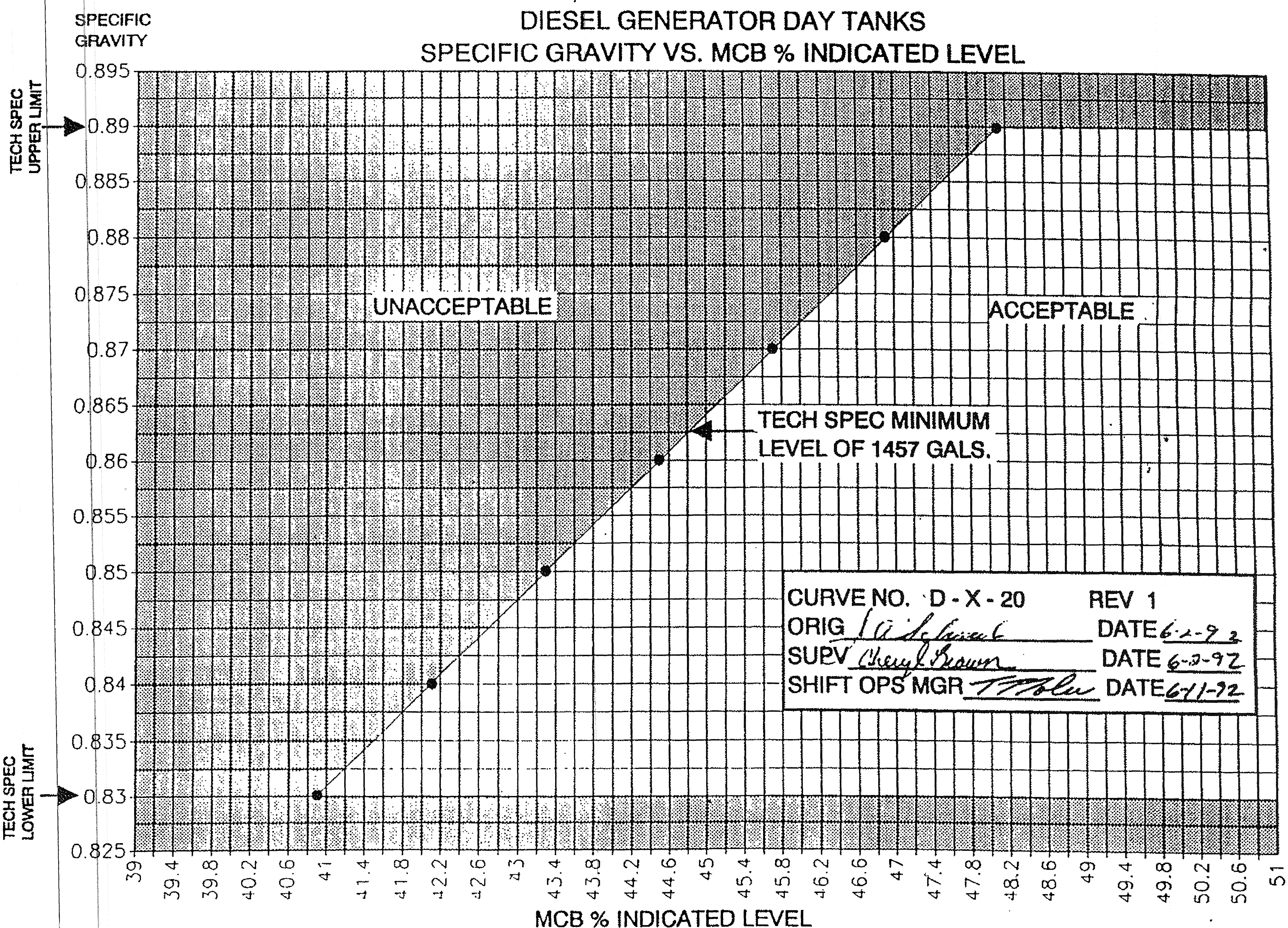
CAUSES:

NOTE
All percent indicated levels are assuming a fuel oil specific gravity value of 0.85.

1. Hi level:
   - Fuel Oil Transfer Pump (FOTP) B-SB failed to stop
2. Lo-Lo level:
   - Improper valve lineup
   - Fuel oil leak
   - FOTP B-SB failed to start at 47.5% level
3. Instrument or alarm circuit malfunction

REFERENCES:

1. Tech. Specs. 3.8.1.1 and 3.8.1.2
2. 6-B-401 Sheet 2547, 2016
3. 6-B-430 Sheet 19.4
4. OP-155, Diesel Generator Emergency Power System
5. OWP-DG, Diesel Generator System
6. ~~PLP-500, Hazardous Substances Spill Notification, Oil Spill Notification, and Fish Kill Reporting~~
7. OP-156.02, AC Electrical Distribution
8. ~~PCR 5475~~



Ref for RO #51, 064K6.08 (Ref provided)

52. 2009A NRC RO 052

Given the following plant conditions:
- Control Room Ventilation is in a normal lineup with 'A' Train fans in operation
- Power is lost to the 'B' Train North MCR Emergency Outside Air Intake (OAI) Radiation Monitor, RM-3505B2SB

Which ONE of the following describes the status of the Control Room Isolation Signal and the action required by OWP-RM-01, Control Room OAI Radiation Monitors?

A Control Room Isolation Signal. . .

A. has NOT occurred;
Isolate the respective OAI within 1 hour.

B. has NOT occurred;
Maintain MCR Ventilation in recirculation with ALL OAIs isolated.

C.✓ has occurred;
Isolate the respective OAI within 1 hour.

D. has occurred;
Maintain MCR Ventilation in recirculation with ALL OAIs isolated.

Plausibility and Answer Analysis

The coincidence for a CRIS is 1 of 2 channels at any OAI. With less than 2 OPERABLE channels per intake, within 1 hour initiate isolation of the respective air intake.

A *Incorrect. This is plausible because for other systems the coincidence is different. CNMT Vent Isolation is 2 of 4. The Tech Spec Action is correct per OWP-RM-01 and T.S. table 3.3-6.*

B *Incorrect. This is plausible because for other systems the coincidence is different. CNMT Vent Isolation is 2 of 4. The Tech Spec Action is also incorrect per OWP-RM-01 and T.S. table 3.3-6 but this is plausible because this is the required action for NO MCR OAIs operable.*

C *Correct. A CRIS has occurred, only one channel is required to trip. The Tech Spec Action is correct per T.S. table 3.3-6.*

D *Incorrect. A CRIS has occurred, only one channel is required to trip. The Tech Spec Action is incorrect per OWP-RM-01 and T.S. table 3.3-6 but this is plausible because this is the required action for NO MCR OAIs operable.*

KA Statement - Process Radiation Monitoring - Ability to (a) predict the impacts of the following on the (SYSTEM) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:(CFR: 41.5 / 43.5 / 45.3 / 45.13) Erratic or failed power supply

Importance Rating:	2.5 2.9
Technical Reference:	OWP-RM-01 Rev. 29, Page 5 Tech Spec Table 3.3-6 (pages 196-198)
References to be provided:	None
Learning Objective:	CRAV, Obj. 3b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	OWP-RM-01
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	073A2.01	K/A 2:	

EIR Number: ______
W/O Number: ______
Clearance Number: ______

1. OWP-RM-01

2 System: Radiation, Effluent, and Explosive Gas Monitoring

3. Component: Control Room OAI Radiation Monitors

4. Scope: Maintenance on Rad Monitors: (circle one)

MCR Normal Outside Air Intake	(OAI 8)	RM-01CZ-3504ASA or BSB
MCR Emerg Outside Air Intake South	(OAI 10)	RM-01CZ-3505A1SA or B1SB
MCR Emerg Outside Air Intake North	(OAI 11)	RM-01CZ-3505A2SA or B2SB

5. Applicable Requirements: 3.3.3.1 (All MODEs and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools).

6. Precautions: (1) With less than 2 OPERABLE channels per intake, within 1 hour initiate isolation of the respective air intake. With no OAI available, maintain operation of the Control Room Emergency Filtration System in the recirculation mode. (2) Refer to Tech Spec Action Statements. (3) If failure of the Rad Monitor resulted in a Control Room Isolation Signal (CRIS), then Normal RAB ventilation will be automatically secured (including any running E-5 fan). Start of RAABES Fans E-6A (CRIS Train A) and E-6B (CRIS Train B) is train specific. Ventilation may be restored after defeating the inoperable Rad Monitor in Sheet 2 using OP-172.

7. LCO Action Log initiated. ______ / ______
Signature Date

8. Component lineups completed per attached sheet (s) ______ / ______
Signature Date

9. Testing required on redundant equipment while component is inoperable:
None

> NOTE: RM-11 historical trends may not be available or sufficient for determining if channel checks are satisfactory. ERFIS or OSI-PI data should be used when available to determine if current values are 'normal' in relation to previous data.

10. Testing/Action required to restore operability. (N/A if tracked on EIR)

	Signature / Date
MST-I0359/I0360 for RM-3504ASA	/
MST-I0361/I0362 for RM-3504BSB	/
MST-I0363/I0364 for RM-3505A1SA	/
MST-I0365/I0366 for RM-3505A2SA	/
MST-I0367/I0368 for RM-3505B1SB	/
MST-I0369/I0370 for RM-3505B2SB	/
OST-1021, 1022, or 1033	/
HPP-780	/

11. LCO Action Log completed. ______ / ______
Signature Date

12. Component lineups restored per attached sheet (s) ______ / ______
Signature Date

13. Remarks: ______

14. Reviewed by: ______ ______
Superintendent- Shift Operations Date

After receiving the final review signature, this OWP becomes a QA Record and should be submitted to Document Services.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment Radioactivity--					
a. Containment Ventilation Isolation Signal Area Monitors	2	3	1, 2, 3, 4, 6	#	27
b. Airborne Gaseous Radioactivity					
1) RCS Leakage Detection	1	1	1, 2, 3, 4	$\leq 1.0\times10^{-3}$ μCi/ml	26, 27
2) Pre-entry Purge	1	1	##	$\leq 2.0\times10^{-3}$ μCi/ml	30
c. Airborne Particulate Radioactivity					
1) RCS Leakage Detection	1	1	1, 2, 3, 4	$\leq 4.0\times10^{-8}$ μCi/ml	26, 27
2) Pre-entry Purge	1	1	##	$\leq 1.5\times10^{-8}$ μCi/ml	30
2. Spent Fuel Pool Area--Fuel Handling Building Emergency Exhaust Actuation					
a. Fuel Handling Building Operating Floor--South Network	1/train***	1/train 2 trains	**	≤ 100 mR/hr	28
b. Fuel Handling Building Operating Floor--North Network	1/train***	1/train 2 trains	*	≤ 100 mR/hr	28
3. Control Room Outside Air Intakes--					
a. Normal Outside Air Intake Isolation	1	2	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.	$\leq 4.9\times10^{-6}$ μCi/ml	29

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
3. Control Room Outside Air Intakes-- (Continued)					
b. Emergency Outside Air Intake Isolation--South Intake	1	2	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.	$\leq 4.9 \times 10^{-6}$ μCi/ml	29
c. Emergency Outside Air Intake Isolation--North Intake	1	2	1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools.	$\leq 4.9 \times 10^{-6}$ μCi/ml	29

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

*	With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.
**	With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.
***	Each channel consists of 3 detectors with 1 of 3 logic. A channel is OPERABLE when 1 or more of the detectors are OPERABLE.
#	For MODES 1, 2, 3 and 4, the setpoint shall be less than or equal to three times detector background at RATED THERMAL POWER. During fuel movement the setpoint shall be less than or equal to 150 mR/hr.
##	Required OPERABLE whenever pre-entry purge system is to be used.

ACTION STATEMENTS

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, declare the associated train of Fuel Handling Building Emergency Exhaust inoperable and perform the requirements of Specification 3.9.12.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour initiate isolation of the respective air intake. With no outside air intakes available, maintain operation of the Control Room Emergency Filtration System in the Recirculation Mode of Operation.

ACTION 30 - With less than the Minimum Channels OPERABLE requirement, pre-entry purge operations shall be suspended and the containment pre-entry purge makeup and exhaust valves shall be maintained closed.

53. 2009A NRC RO 053

Given the following plant conditions:

- A Reactor Trip and Safety Injection Actuation have occurred

Which ONE of the following sets of components are directly supplied by the Emergency Service Water system?

A. CSIP Oil Coolers
EDG Jacket Water Heat Exchangers
Containment Fan Coil Units (AH-37,-38,-39)

B.✓ CSIP Oil Coolers
CCW Heat Exchangers
Containment Fan Coolers (AH-1,-2,-3,-4)

C. RHR Heat Exchangers
CCW Heat Exchangers
AFW Pump Emergency Makeup

D. RCP Bearing Oil Coolers
EDG Jacket Water Heat Exchangers
Containment Fan Coolers (AH-1,-2,-3,-4)

Plausibility and Answer Analysis

A Incorrect. Plausible since EDG Jacket Water HX and CSIP Oil Coolers are cooled by ESW. Containment Fan Coil units are supplied by NSW, not ESW.

B Correct. All three components are cooled by ESW.

C Incorrect. Plausible since AFW Emergency Makeup and cooling to CCW HXs are supplied by ESW. RHR HX is a vital load, but it is cooled by CCW, not ESW

D Incorrect. Plausible since Containment Fan Coolers and EDG Jacket Water Heat Exchangers are cooled by ESW. RCP Bearing Oil Coolers are considered a vital load but are cooled by CCW, not ESW.

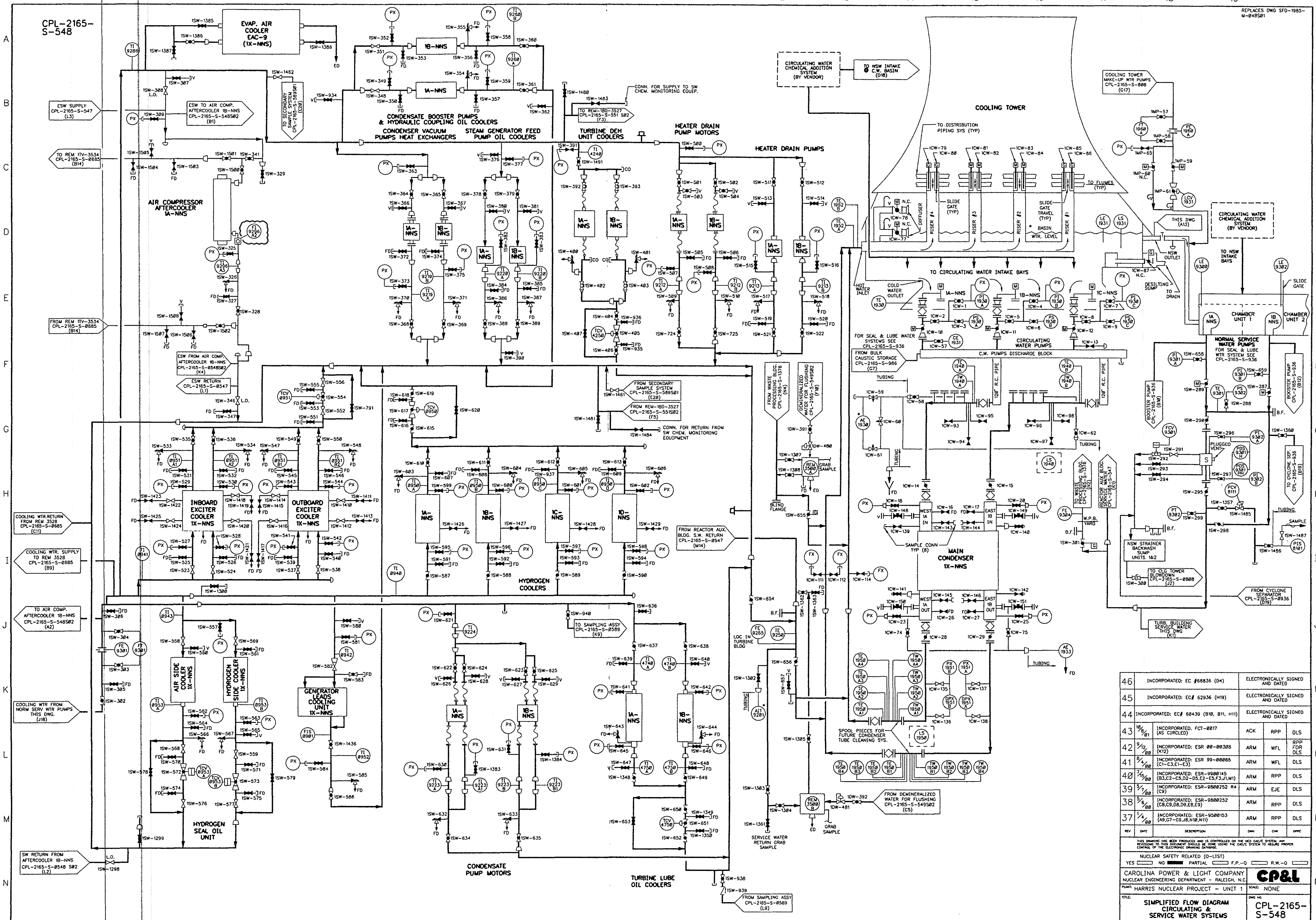
QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Service Water - Ability to monitor automatic operations of the (SYSTEM) including: (CFR: 41.7 / 45.5) Emergency heat loads

Importance Rating:	3.7 3.7
Technical Reference:	SFD 2165-S-0548 SWS Student Text
References to be provided:	None
Learning Objective:	SWS, Obj. 5h
Question Origin:	Bank (Direct) OIT Development Bank 076 A3.02
Comments:	K/A match

Origin:	BANK	Cog Level:	F
Difficulty:	2	Reference:	SWS STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	076A3.02	K/A 2:	



REV	DATE	DESCRIPTION	DWN	CHK	APPR
46		INCORPORATED: EC #66836 (D4)	ELECTRONICALLY SIGNED AND DATED		
45		INCORPORATED: EC# 62936 (H19)	ELECTRONICALLY SIGNED AND DATED		
44		INCORPORATED: EC# 60439 (B10, B11, H11)	ELECTRONICALLY SIGNED AND DATED		
43	10/8/01	INCORPORATED: FCT-0017 (AS CIRCLED)	ACK	RPP	DLS
42	9/12/00	INCORPORATED: ESR 00-00308 (K12)	ARM	WFL	RPP FOR DLS
41	8/4/00	INCORPORATED: ESR 99-00085 (C1-C3,E1-E3)	ARM	WFL	DLS
40	7/15/00	INCORPORATED: ESR-9900145 (B3,C2-C5,D2-D5,E2-E5,F3,J1,M1)	ARM	RPP	DLS
39	5/7/00	INCORPORATED: ESR-9800252 R4 (C9)	ARM	EJE	DLS
38	5/6/00	INCORPORATED: ESR-9800252 (C8,C9,D8,D9,E8,E9)	ARM	RPP	DLS
37	1/4/00	INCORPORATED: ESR-9500153 (A9,G7-G9,J8,N10,N11)	ARM	RPP	DLS

THIS DRAWING HAS BEEN PRODUCED AND IS CONTROLLED ON THE NED CAD/E SYSTEM. ANY REVISIONS TO THIS DOCUMENT SHOULD BE DONE USING THE CAD/E SYSTEM TO ASSURE PROPER CONTROL OF THE ELECTRONIC DRAWING DATABASE.

NUCLEAR SAFETY RELATED (Q-LIST)
YES ☐ NO ■ PARTIAL ☐ F.P.-Q ☐ R.W.-Q ☐

CAROLINA POWER & LIGHT COMPANY
NUCLEAR ENGINEERING DEPARTMENT - RALEIGH, N.C.

PLANT: HARRIS NUCLEAR PROJECT - UNIT 1
SCALE: NONE

TITLE: SIMPLIFIED FLOW DIAGRAM CIRCULATING & SERVICE WATER SYSTEMS

DWG NO. CPL-2165-S-548

54. 2009A NRC RO 054

Given the following plant conditions:
- The Reactor is at 3% power
- Main Feed Reg Valve Bypasses are controlling SG Level in AUTO
- GP-005, Power Operation Mode 2 to Mode 1 is in progress

Current Plant Conditions:
- Instrument Air (IA) Pressure begins lowering due to a leak and cannot be stabilized

Which ONE of the following describes the action required in accordance with AOP-017, Loss of Instrument Air and at what pressure this action is required?

	Action Required	IA Pressure
A.	Trip the Reactor and Perform Path-1	60 psig
B.✓	Trip the Reactor and Perform Path-1	35 psig
C.	Initiate AFW flow to maintain Steam Generator levels	60 psig
D.	Initiate AFW flow to maintain Steam Generator levels	35 psig

Plausibility and Answer Analysis

A *Incorrect. MFRVs receive an Auto Shut Signal at 60 psig but flow is on the MFRBVs. If flow was on the MFRVs, the continuous action to "maintain main feedwater flow to all SGs" (step 1) could not be met and the RNO would direct a reactor trip if above the POAH.*

B *Correct. If IA Pressure cannot be maintained above 35 psig, the RNO directs a reactor trip anytime the reactor is critical.*

C *Incorrect. MFRVs receive an Auto Shut Signal at 60 psig but flow is on the MFRBVs. If flow was on the MFRVs, the continuous action to "maintain main feedwater flow to all SGs" (step 1) could not be met and the RNO would directing initiating AFW flow to maintain Steam Generator levels is correct when below the point of adding heat but the reactor is at 3%.*

D *Incorrect. If IA Pressure cannot be maintained above 35 psig, the RNO directs a reactor trip anytime the reactor is critical and does not differentiate between above or below the POAH.*

KA Statement - Instrument Air - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

Importance Rating:	4.2 4.4
Technical Reference:	AOP-017 Rev. 28, Step 2, page 4
References to be provided:	None
Learning Objective:	LP-AOP-3.17, Obj. 2
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-017
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	078G2.2.44	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

- This procedure contains no immediate actions.
- FW regulating valves receive a shut signal when pressure falls to 60 psig on the Control Air header.
- PI-9751.1, Instrument Air Header Pressure, may not be indicative of pressure throughout the Instrument Air System. The plant should be monitored closely for possible spurious valve operations due to low system pressure.

INSTRUCTIONS	RESPONSE NOT OBTAINED
✱☐ 1. **MAINTAIN** BOTH of the following: • ALL Steam Generator levels greater than 30% • Main Feedwater flow to ALL Steam Generators	1. **PERFORM** the following: ☐ a. **IF** Reactor is ABOVE POAH, **THEN TRIP** the Reactor **AND PERFORM** EOP Path-1 while continuing with this AOP. b. **IF** BELOW POAH, **THEN**: ☐ (1) **VERIFY** AFW capable of feeding Steam Generators. ☐ (2) **MAINTAIN** Steam Generator levels as required by current plant conditions.
✱☐ 2. **CHECK** Instrument Air pressure MAINTAINED ABOVE 35 PSIG.	2. **PERFORM** the following: ☐ a. **IF** Reactor is CRITICAL, **THEN TRIP** the Reactor **AND PERFORM** EOP Path-1 while continuing with this AOP. **(Continued on Next Page)**

55. 2009A NRC RO 055

Following a Containment Isolation Phase 'A' signal, which ONE of the following can be sampled without resetting the Phase 'A' signal?

A.✓ RHR System

B. RCS Hot Legs 'B' and 'C'

C. Pressurizer Liquid Space

D. Safety Injection Accumulators

Plausibility and Answer Analysis

A Correct. The RHR sample valves are NOT Containment Isolation valves. The system is used during post accident conditions and could contain highly radioactive liquid. The valves are not listed in PLP-106 Attachment 5 and will not reposition on a Phase A signal. These sample valves can be positioned at the operators discretion (open/closed) without resetting the Phase A signal.

B Incorrect. RCS sample valves are listed in PLP-106 Attachment 5 as Containment isolation valves that isolate on a Phase A signal. The RCS sample valves are also labled on the MCB with an 'A' in the upper left corner of the control switch to indicate that they close with a Phase A signal. The valves cannot be opened for sampling without first resetting the Phase A signal.

C Incorrect. The Pressurizer liquid sample valves are listed in PLP-106 Attachment 5 as Containment Isolation valves that isolate on a Phase A signal. The RCS sample valves are also labled on the MCB with an 'A' in the upper left corner of the control switch to indicate that they close with a Phase A signal. The valves cannot be opened for sampling without first resetting the Phase A signal.

D Incorrect. Safety Injection accumulator sample valves are listed in PLP-106 Attachment 5 as Containment isolation valves that isolate on a Phase A signal. The RCS sample valves are also labled on the MCB with an 'A' in the upper left corner of the control switch to indicate that they close with a Phase A signal. The valves cannot be opened for sampling without first resetting the Phase A signal.

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Containment - Knowledge of the physical connections and/or cause-effect relationships between (SYSTEM) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Containment isolation/containment integrity

Importance Rating:	3.9 4.1
Technical Reference:	PLP-106 Rev. 44, Attachment 5 (pages 17-20) Primary Sample System Student Text
References to be provided:	None
Learning Objective:	PSS, 2b
Question Origin:	BANK OIT Exam Bank PSS (04A) 002
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	2	Reference:	PLP-106 ATT. 5
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	103K1.02	K/A 2:	

Containment Isolation Valves

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1.	PHASE A ISOLATION				
1	1MS-29 (MS-V126)	MS LINE C TO SAMPLING SYSTEM	60	1, 6,13	NONE
2	1MS-27 (MS-V124)	MS LINE B TO SAMPLING SYSTEM	60	1, 6,13	NONE
3	1MS-25 (MS-V122)	MS LINE A TO SAMPLING SYSTEM	60	1, 6,13	NONE
7	1CS-7 (CS-V511)	CVCS NORMAL LTDN ISOL	10	7,13	1CS-11
7	1CS-8 (CS-V512)	CVCS NORMAL LTDN ISOL	10	7,13	1CS-11
7	1CS-9 (CS-V513)	CVCS NORMAL LTDN ISOL	10	7,13	1CS-11
7	1CS-11 (CS-V518)	CVCS NORMAL LTDN ISOL	10	7,13	1CS-7,8, 9 and 10
12	1CS-470 (CS-V516)	RCP SEAL WTR RETURN & EXCESS LTDN	10	7,13	1CS-472
12	1CS-472 (CS-V517)	RCP SEAL WTR RETURN & EXCESS LTDN	10	7,13	1CS-470 and 471
33	1SP-209 (SP-V408)	GAS RETURN FROM PASS SKID #2	5	7,13	1SP-208
33	1SP-208 (SP-V409)	GAS RETURN FROM PASS SKID #2	5	7,13	1SP-209
37	1CC-176 (CC-V172)	CCW TO RCDT & EXCESS LTDN HEAT EXCHS	10	1, 6,13	NONE
38	1CC-202 (CC-V182)	CCW FROM RCDT & EXCESS LTDN HEAT EXCHS	10	1, 6,13	NONE
40	1RC-161 (RC-D525)	REACTOR MAKEUP WTR TO PRT	60	7,13	1RC-164
42	1ED-121 (WL-L600)	RCDT PUMPS DISCH	10	7,13	1ED-125
42	1ED-125 (WL-D650)	RCDT PUMPS DISCH	10	7,13	1ED-121 and 119

Containment Isolation Valves

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1. PHASE A ISOLATION (continued)					
73A	1SP-12 (SP-V300)	HYDROGEN ANALYZER A	60	7,13	1SP-915
73A	1SP-915 (SP-V348)	HYDROGEN ANALYZER A	60	7,13	1SP-12
73B	1SP-941 (SP-V301)	HYDROGEN ANALYZER A	60	7,13	1SP-917
73B	1SP-917 (SP-V349)	HYDROGEN ANALYZER A	60	7,13	1SP-941
74	1ED-94 (MD-V36)	CNMT SUMP PUMP DISCH	60	7,13	1ED-95
74	1ED-95 (MD-V77)	CNMT SUMP PUMP DISCH	60	7,13	1ED-94
76A	1SI-179 (SI-V554)	ACCUMULATOR FILL FROM RWST	10	7,13	1SI-182
76B	1SI-263 (SI-V555)	ACCUMULATOR DRAIN TO RWST	10	7,13	1SI-264
76B	1SI-264 (SI-V550)	ACCUMULATOR DRAIN TO RWST	10	7,13	1SI-263
77A	1SI-287 (SI-V530)	NITROGEN SUPPLY	10	7,13	1SI-290
77B	1RC-141 (RC-D528)	PRT NITROGEN CONNECTION	10	7,13	1RC-144
77B	1RC-144 (RC-D529)	PRT NITROGEN CONNECTION	10	7,13	1RC-141
77C	1ED-164 (WG-D590)	RCDT HYDROGEN CONNECTION	10	7,13	1ED-161
77C	1ED-161 (WG-D291)	RCDT HYDROGEN CONNECTION	10	7,13	1ED-164
78A	1SP-948 (SP-V111)	RCS SAMPLE	60	7,13	1SP-949
78A	1SP-949 (SP-V23)	RCS SAMPLE	60	7,13	1SP-948
78B	1SP-40 (SP-V11)	PRESSURIZER LIQ SAMPLE	60	7,13	1SP-41
78B	1SP-41 (SP-V12)	PRESSURIZER LIQ SAMPLE	60	7,13	1SP-40

Containment Isolation Valves

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1.	PHASE A ISOLATION (continued)				
78C	1SP-59 (SP-V1)	PRESSURIZER STEAM SAMPLE	60	7,13	1SP-60
78C	1SP-60 (SP-V2)	PRESSURIZER STEAM SAMPLE	60	7,13	1SP-59
78D	1SP-78 (SP-V113)	ACCUMULATOR A SAMPLE	60	7,13	1SP-85
78D	1SP-81 (SP-V114)	ACCUMULATOR B SAMPLE	60	7,13	1SP-85
78D	1SP-84 (SP-V115)	ACCUMULATOR C SAMPLE	60	7,13	1SP-85
78D	1SP-85 (SP-V116)	ACCUMULATORS SAMPLE	60	7,13	1SP-78, 81, and 84
80	1IA-819 (IA-V192)	INSTRUMENT AIR SUPPLY	60	7,13	1IA-220
83A	1SP-916 (SP-V448)	RADIATION MONITOR REM-3502A	60	7,13	1SP-16
83A	1SP-16 (SP-V449)	RADIATION MONITOR REM-3502A	60	7,13	1SP-916
83B	1SP-918 (SP-V450)	RADIATION MONITOR REM-3502A	60	7,13	1SP-939
83B	1SP-939 (SP-V451)	RADIATION MONITOR REM-3502A	60	7,13	1SP-918
86A	1SP-42 (SP-V308)	HYDROGEN ANALYZER B	60	7,13	1SP-919
86A	1SP-919 (SP-V314)	HYDROGEN ANALYZER B	60	7,13	1SP-42
86B	1SP-62 (SP-V309)	HYDROGEN ANALYZER B	60	7,13	1SP-943
86B	1SP-943 (SP-V315)	HYDROGEN ANALYZER B	60	7,13	1SP-62
88	1SP-201 (SP-V406)	LIQUID RETURN FROM PASS SKID #1	5	7,13	1SP-200
88	1SP-200 (SP-V407)	LIQUID RETURN FROM PASS SKID #1	5	7,13	1SP-201

Containment Isolation Valves

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1.	PHASE A ISOLATION (continued)				
91	1SW-240 (SW-B89)	SERVICE WTR FROM NNS FAN COILS	60	7,13	1SW-242
91	1SW-242 (SW-B90)	SERVICE WTR FROM NNS FAN COILS	60	7,13	1SW-240
92	1SW-231 (SW-B88)	SERVICE WTR TO NNS FAN COILS	60	7,13	1SW-233
105	1FP-347 (FP-B1)	FIRE WATER SPRINKLER SUPPLY	60	7,13	1FP-349

2. PHASE B ISOLATION

NOTE: Valve 1CC-207 in penetration 35 is not classified as a containment isolation valve.

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
35	1CC-208 (CC-V170)	CCW TO RCPS	10	7,13	1CC-211
36	1CC-297 (CC-V184)	CCW FROM RCP BEARING OIL HXS	10	7,13	1CC-299
36	1CC-299 (CC-V183)	CCW FROM RCP BEARING OIL HXS	10	7,13	1CC-297 and 298
39	1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIER HXS	10	7,13	1CC-251
39	1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIER HXS	10	7,13	1CC-249 and 250

3. SAFETY INJECTION ACTUATION

NOTE: Valve 1CS-235 in penetration 8 is not classified as a containment isolation valve; refer to Specifications 3.1.2.1, 3.1.2.2, 3.3.3.5.b, 3.5.2, and 3.5.3.

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
8	1CS-238 (CS-V610)	CVCS NORMAL CHARGING	10	7,13	1CS-477

56. 2009A NRC RO 056

Which ONE of the following describes the power supplies to the Rod Drive Motor Generator (MG) sets and the breakers that would be locally tripped first during an ATWS in accordance with the local operator aid?

	RDMG Power Supplies	Breakers
A.	1D2 and 1E2	MG set motor breakers
B.✓	1D2 and 1E2	MG set generator output breakers
C.	1D3 and 1E3	MG set motor breakers
D.	1D3 and 1E3	MG set generator output breakers

Plausibility and Answer Analysis

From the local operator aid:
The reactor is locally tripped using any of the following methods (listed in order of preference):

1. *Locally trip the reactor trip breakers (depress the red, circular trip plate on the cubicle door of each trip breaker)*
2. *Locally trip both rod drive MG set generator output breakers*
3. *Locally trip both rod drive MG set motor breakers*

A Incorrect. Plausible since power supplies are correct but tripping the RDMG motor breaker is not the first breaker tripped in accordance with the local operator aid but is listed on the operator aid.

B Correct. Power for the MG sets 1A and 1B is supplied by 480 V AC auxiliary buses 1D2 and 1E2. The RDMG generator output breaker is locally tripped first to immediately cut power to the control rod system (which is correct).

C Incorrect. Plausible since the RDMG sets are supplied by 480V power supplies downstream of the 6.9KV Aux Buses D and E but it is 1D2 and 1E2 suppling power, not 1D3 and 1E3 and tripping the RDMG motor breaker is not the first breaker tripped in accordance with the local operator aid but is listed on the operator aid.

D Incorrect. Plausible since the RDMG sets are supplied by 480V power supplies downstream of the 6.9KV Aux Buses D and E but it is 1D2 and 1E2 suppling power, not 1D3 and 1E3. The RDMG generator output breaker is locally tripped first to immediately cut power to the control rod system (which is correct).

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Control Rod Drive - Knowledge of electrical power supplies to the following: (CFR: 41.7) M/G sets

Importance Rating:	3.1 3.5
Technical Reference:	OP-104 Rev. 28, Page 46 FRP-S.1 Rev. 15, Page 8
References to be provided:	None
Learning Objective:	LP-EOP-3.15, Obj. 5b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	FRP-S.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	001K2.05	K/A 2:	

Attachment 1 - Rod Control System Electrical Lineup Prestart Checklist
Sheet 2 of 7

COMPONENT NUMBER	COMPONENT DESCRIPTION	POSITION	CHECK	VERIFY
1D2-6D	Rod Drive MG Set 1A	OPEN		
1E2-2A	Rod Drive MG Set 1B	OPEN		
DP-1A-1-18	Rod Drive Power Supply Cubicle 2 (Motor Genrtr 1A)	ON		
DP-1A-2-18	Rod Drive Power Supply Cubicle 1 (Motor Genrtr 1B)	ON		
PP-1D211 Ckt 16 & 18	Rod Position Indication Distribution Panel	ON		
PP-1E211 Ckt 16 & 18	Rod Position Indication Distribution Panel	ON		
PP-1E212-15	Full Length Rod Control	ON		
PP-1E212-15-BU External Breaker Box	Full Length Rod Power Cab SCD	ON		
1EE-E004:002	Stationary Coil Knife Switch Power Disconnect Switch For: Shutdown Bank C Group 1	ON		
1EE-E004:003	Movable Coil Knife Switch Power Disconnect Switch For: Shutdown Bank C Group 1	ON		
1EE-E004:004	Lift Coil Knife Switch Power Disconnect Switch For: Shutdown Bank C Group 1	ON		
1EE-E005:002	Stationary Coil Knife Switch Power Disconnect Switch For: Control Bank A Group 1 Control Bank C Group 1 Shutdown Bank A Group 1	ON		
1EE-E005:003	Movable Coil Knife Switch Power Disconnect Switch For: Control Bank A Group 1 Control Bank C Group 1 Shutdown Bank A Group 1	ON		

Instructions	Response Not Obtained
9. Check Trip Status:	
a. Check reactor - TRIPPED	a. Locally trip reactor using any of the following (listed in order of preference): 1) Locally trip reactor trip breakers. 2) Locally trip both rod drive MG set generator output breakers. 3) Locally trip both rod drive MG set motor breakers.
b. Check turbine - TRIPPED	
1) Check for any of the following: o All turbine throttle valves - SHUT o All turbine governor valves - SHUT o All MSIVs AND bypass valves - SHUT	1) Locally trip turbine.
10. Check Reactor Subcritical:	
a. Check for both of the following: o Power range channels - LESS THAN 5% o Intermediate range startup rate channels - NEGATIVE	a. Observe <u>NOTE</u> prior to Step 11 AND GO TO Step 11.
b. Observe <u>CAUTION</u> prior to Step 25 AND GO TO Step 25.	

57. 2009A NRC RO 057

Given the following plant conditions:
- Following a rapid power reduction from 100% power, the plant is stabilized at 40% power

The Reactor Operator reports the following indications:
- Pressurizer pressure is 2275 psig and decreasing
- Pressurizer level is 47% and decreasing
- RCS Tavg is at Tref
- BOTH Pressurizer spray valves indicate mid-position
- ALL Pressurizer backup heaters are deenergized

These conditions are indicative of a. . .

A. normal system response following an insurge into the Pressurizer.

B. normal system response following an outsurge from the Pressurizer.

C. failure of the Pressurizer spray valves to shut.

D✓ failure of the Pressurizer backup heaters to energize.

Plausibility and Answer Analysis

A *Incorrect. Plausible since the response is correct, with the exception of the pressurizer heaters not being energized, for an insurge to the pressurizer.*

B *Incorrect. Plausible since a continuous outsurge is normally maintained during power transients which would cause the spray valves to open, but the heaters should also be energized.*

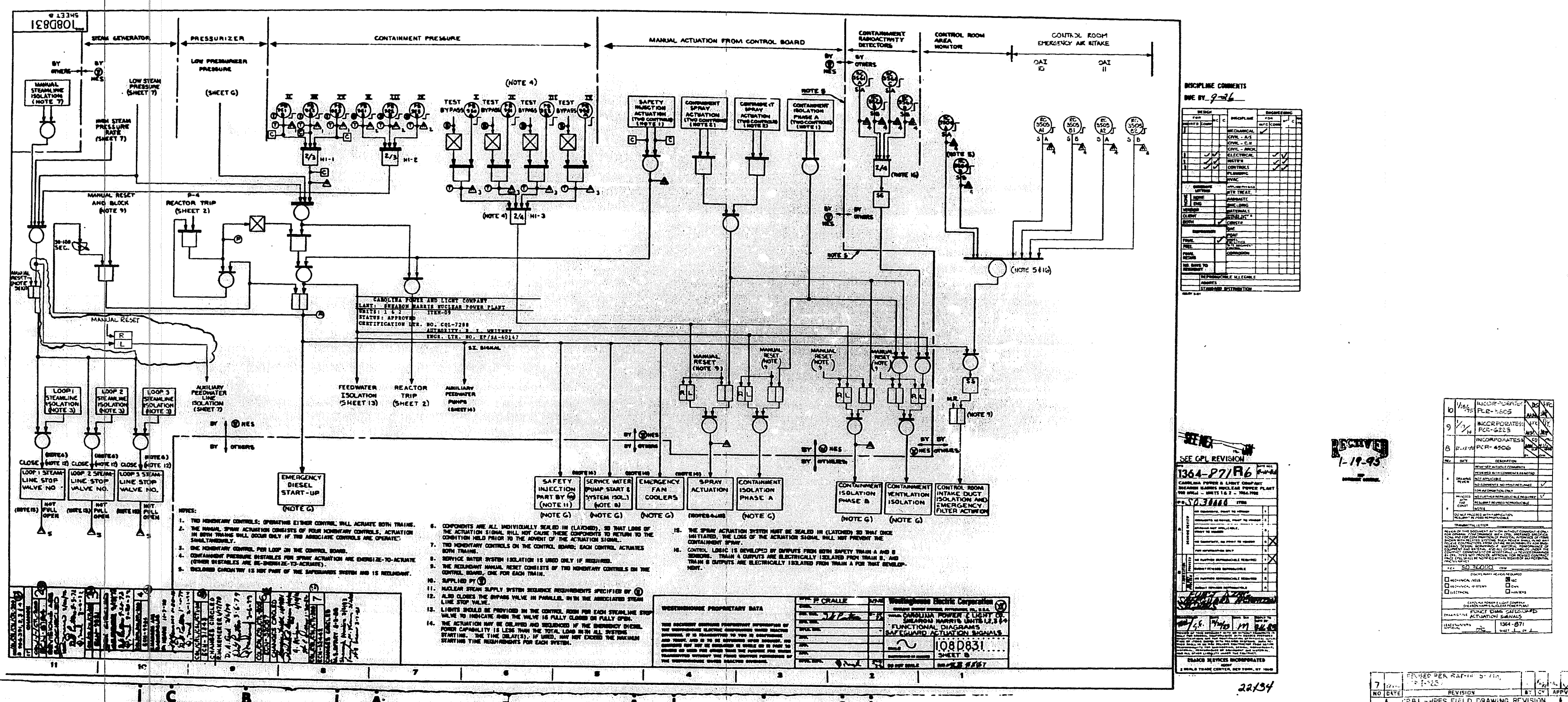
C *Incorrect. Plausible since pressurizer pressure is decreasing, but the setpoint for spray valves shutting is 2260.*

D *Correct. A rapid downpower transient can result in an insurge to the pressurizer. This should result in the conditions noted, including a high pressurizer level causing the heaters to be energized even during a high pressure condition causing the spray valves to be open. The heaters not being energized with level more than 5% high is indicative of a level control system failure.*

KA Statement - Pressurizer Level Control - Ability to predict and/or monitor changes in parameters associated with operating the (SYSTEM) controls including: (CFR: 41.5 / 45.5) PZR level and pressure

Importance Rating:	3.5 3.6
Technical Reference:	DWG 1364-874 sheet 11 PZRLC Student Text
References to be provided:	None
Learning Objective:	PZRLC, Obj. 7c
Question Origin:	BANK OIT Exam Bank Question PRZLC (09C) 001
Comments:	

Origin:	BANK	Cog Level:	H
Difficulty:	4	Reference:	PRZLC TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	011A1.01	K/A 2:	


108D831
SHEET 8
STEAM GENERATOR
PRESSURIZER
CONTAINMENT PRESSURE
MANUAL ACTUATION FROM CONTROL BOARD
CONTAINMENT RADIOACTIVITY DETECTORS
CONTROL ROOM AREA MONITOR
CONTROL ROOM EMERGENCY AIR INTAKE
MANUAL STEAMLINE ISOLATION (NOTE 7)
LOW STEAM PRESSURE (SHEET 7)
HIGH STEAM PRESSURE RATE (SHEET 7)
LOW PRESSURIZER PRESSURE (SHEET 6)
(NOTE 4)
TEST BYPASS
MANUAL RESET AND BLOCK (NOTE 9)
P-4 REACTOR TRIP (SHEET 2)
MANUAL RESET
SAFETY INJECTION ACTUATION (TWO CONTROLS) (NOTE 1)
CONTAINMENT SPRAY ACTUATION (TWO CONTROLS) (NOTE 2)
CONTAINMENT ISOLATION PHASE A (TWO CONTROLS) (NOTE 1)
NOTE 5
(NOTE 16)
(NOTE 5416)
CAROLINA POWER AND LIGHT COMPANY
PLANT: SHEARON HARRIS NUCLEAR POWER PLANT
UNITS: 1 & 2 ITEM-09
STATUS: APPROVED
CERTIFICATION LTR. NO. CQL-7288
AUTHORITY: R. T. WHITNEY
ENGR. LTR. NO. EF/SA-40147
SI SIGNAL
LOOP 1 STEAMLINE ISOLATION (NOTE 3)
LOOP 2 STEAMLINE ISOLATION (NOTE 3)
LOOP 3 STEAMLINE ISOLATION (NOTE 3)
AUXILIARY FEEDWATER LINE ISOLATION (SHEET 7)
FEEDWATER ISOLATION (SHEET 13)
REACTOR TRIP (SHEET 2)
AUXILIARY FEEDWATER PUMPS (SHEET 14)
LOOP 1 STEAM LINE STOP VALVE NO.
LOOP 2 STEAM LINE STOP VALVE NO.
LOOP 3 STEAM LINE STOP VALVE NO.
NOT FULL OPEN
EMERGENCY DIESEL START-UP (NOTE 6)
SAFETY INJECTION PART BY (NOTE 11) (NOTE 6)
SERVICE WATER (PUMP START & SYSTEM ISOL.) (NOTE 8) (NOTE 6)
EMERGENCY FAN COOLERS (NOTE 6)
SPRAY ACTUATION
CONTAINMENT ISOLATION PHASE A (NOTE 6)
CONTAINMENT ISOLATION PHASE B (NOTE 6)
CONTAINMENT VENTILATION ISOLATION (NOTE 6)
CONTROL ROOM INTAKE DUCT ISOLATION AND EMERGENCY FILTER ACTUATION
BY NES
BY OTHERS
NOTES:
1. TWO MOMENTARY CONTROLS; OPERATING EITHER CONTROL WILL ACTUATE BOTH TRAINS.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS, ACTUATION IN BOTH TRAINS WILL OCCUR ONLY IF TWO ASSOCIATE CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE MOMENTARY CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE-TO-ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. ENCLOSED CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM AND IS REDUNDANT.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD; EACH CONTROL ACTUATES BOTH TRAINS.
8. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
9. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
10. SUPPLIED BY
11. NUCLEAR STEAM SUPPLY SYSTEM SEQUENCE REQUIREMENTS SPECIFIED BY
12. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
13. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAMLINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
14. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
15. THE SPRAY ACTUATION SYSTEM MUST BE SEALED IN (LATCHED) SO THAT ONCE INITIATED, THE LOSS OF THE ACTUATION SIGNAL WILL NOT PREVENT THE CONTAINMENT SPRAY.
16. CONTROL LOGIC IS DEVELOPED BY OUTPUTS FROM BOTH SAFETY TRAIN A AND B SENSORS. TRAIN A OUTPUTS ARE ELECTRICALLY ISOLATED FROM TRAIN B, AND TRAIN B OUTPUTS ARE ELECTRICALLY ISOLATED FROM TRAIN A FOR THAT DEVELOPMENT.
WESTINGHOUSE PROPRIETARY DATA
P. CRALLE
Westinghouse Electric Corporation
CAROLINA POWER & LIGHT CO.
SHEARON HARRIS UNITS 1,2,3 & 4
FUNCTIONAL DIAGRAMS
SAFEGUARD ACTUATION SIGNALS
108D831
SHEET 8
DO NOT SCALE
DISCIPLINE COMMENTS
DUE BY 9-26
SEE GPL REVISION
1364-871 R6
EBASCO SERVICES INCORPORATED
22134
RECEIVED
1-19-93
INCORPORATES PCR-5805
INCORPORATES PCR-6213
INCORPORATES PCR-4506
CP&L - HPES FIELD DRAWING REVISION
REVISION
NO DATE BY CK APPVD

58. 2009A NRC RO 058

Given the following plant conditions:

- A reactor startup is in progress in accordance with GP-004, Reactor Startup
- The Reactor Operator is currently withdrawing Control Bank 'C' rods

Which ONE of the following is the indicated rod height on the Control Bank 'C' Step Counters at which Control Bank 'D' rods are expected to begin withdrawing and what is the expected rod speed?

	Control Bank 'C' Step Counter Indication	Control Bank 'D' Rod Speed
A.	103 Steps	48 SPM
B.	103 Steps	64 SPM
C.✓	128 Steps	48 SPM
D.	128 Steps	64 SPM

Plausibility and Answer Analysis

A *Incorrect. Bank Overlap starts at an indicated height of 128 steps. 103 steps is the duration of Bank Overlap. 48 SPM is correct for Manual withdrawal of the Control Banks.*

B *Incorrect. Bank Overlap starts at an indicated height of 128 steps. 103 steps is the duration of Bank Overlap. 64 SPM is for the withdrawal of the Shutdown Banks.*

C *Correct. Bank Overlap starts at an indicated height of 128 steps. 48 SPM is correct for Manual withdrawal of the Control Banks.*

D *Incorrect. Bank Overlap starts at an indicated height of 128 steps. 64 SPM is for the withdrawal of the Shutdown Banks.*

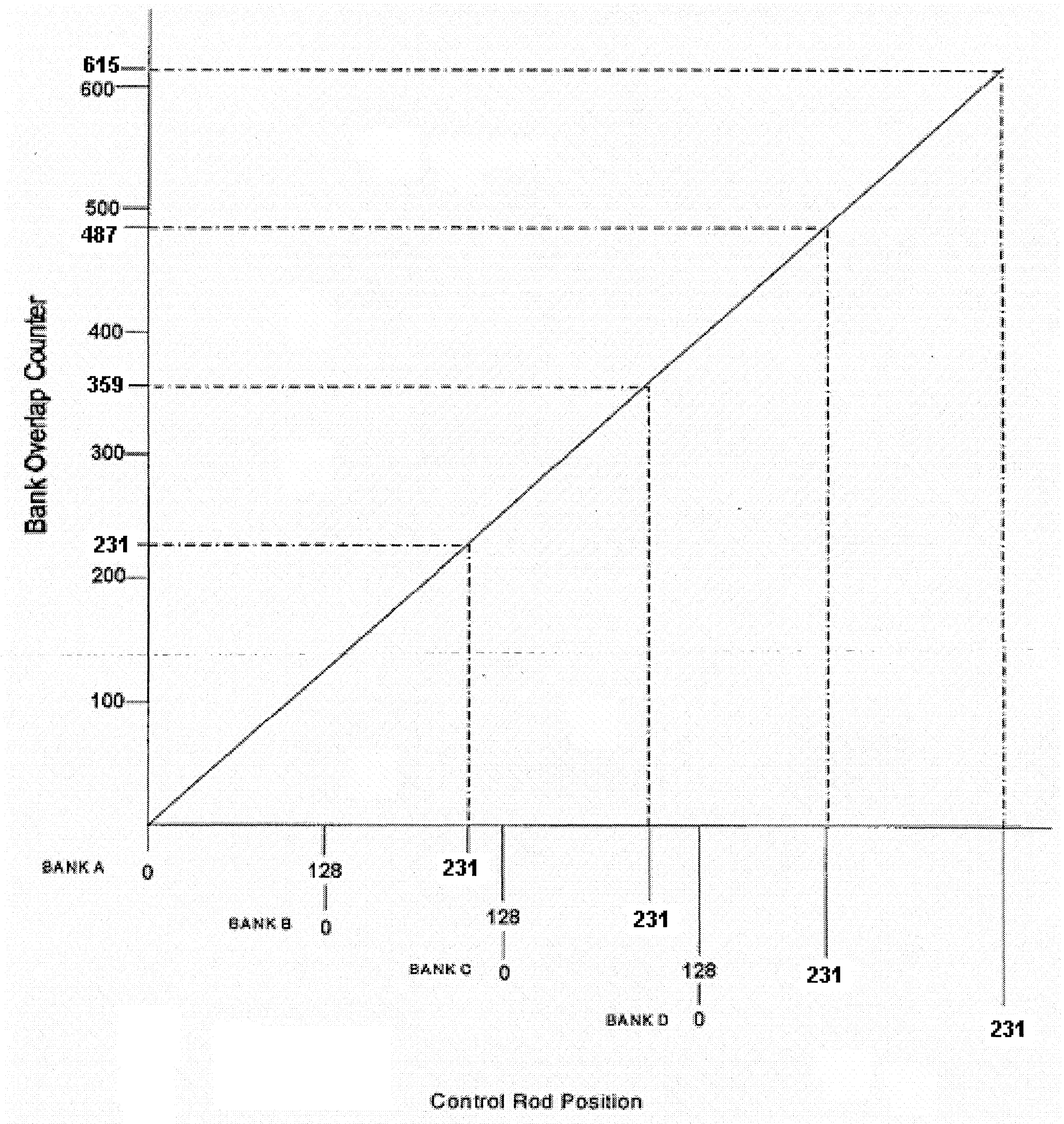
KA Statement - Rod Position Indication - Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) Rod selection control

Importance Rating:	3.3 3.1
Technical Reference:	AOP-001 Rev 32, Attachment 3, page 43 OP-104 Rev. 28, pages 15 & 17
References to be provided:	None
Learning Objective:	RODCS, Obj. 5a
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	OP-104
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	014A4.01	K/A 2:	

Attachment 3 - Bank Overlap Counter Indication

Sheet 1 of 1



--END OF ATTACHMENT 3--

5.3.2 Procedural Steps (continued)

c. Rod Control IN-OUT Direction Lamps are extinguished. ______

d. ROD CONTROL URGENT ALARM annunciator is extinguished. ______

e. ROD CONTROL NON-URGENT ALARM annunciator is extinguished. ______

f. ROD DRIVE MG SET TROUBLE annunciator is extinguished. ______

g. Pulse-Analog Converter is reset to zero. ______

> **NOTE:** 231 steps is the normal full out position for the shutdown banks. Per PLP-106 the Shutdown Rod Insertion Limit is greater than or equal to 225 steps.

6. **WITHDRAW** Shutdown Rod Banks to 231 steps as follows:

a. At the MCB, **ROTATE** the ROD BANK SELECTOR Switch to SBC. ______

b. At the MCB, **POSITION** ROD MOTION Switch to WITHDRAW and **VERIFY** the RODS OUT Direction Lamp lights. ______

c. **OBSERVE** that Shutdown Bank C Group Step Counter SC-SBC1 indicates withdrawal steps. ______

d. **VERIFY** the rods in Shutdown Bank C are moving on the Digital Rod Position Indication display. ______

e. At the MCB, **WHEN** Shutdown Rod Bank C reaches 231 steps, **RELEASE** the ROD MOTION Switch allowing it to return to the neutral position and **VERIFY** the RODS OUT Direction Lamp extinguishes. ______

f. At the MCB, **ROTATE** the ROD BANK SELECTOR Switch to SBA. ______

g. At the MCB, **POSITION** ROD MOTION Switch to WITHDRAW and **VERIFY** the RODS OUT Direction Lamp lights. ______

h. **VERIFY** rod speed of 64 steps per minute on SI-408. ______

i. **OBSERVE** that Shutdown Bank A Step Counters SC-SBA1 and SC-SBA2 indicate withdrawal steps. ______

j. **VERIFY** the rods in Shutdown Bank A are moving on the Digital Rod Position Indication Display. ______

5.4. Manual Control Bank Withdrawal
R (Reference 2.3.1)

5.4.1. Initial Conditions

1. All shutdown rods have been withdrawn, per Section 5.3, by observing the Group Step Counters and Digital Rod Position Indication System. All Shutdown Rod Group Step Counters must read greater than or equal to 225 steps. _____

5.4.2. Procedural Steps

1. At the MCB, **ROTATE** the ROD BANK SELECTOR Switch to MAN. _____

2. **VERIFY** Rod Speed of 48 steps per minute on SI-408. _____

NOTE: During a Reactor Startup, Steps 5.4.2.3 through 5.4.2.6 are repeated multiple times, with rod motion stopped to observe reactivity affects, record 1/M data, or for other reasons. The intent is to initial for these Steps at the completion of the entire evolution, not for each time it is performed.

3. At the MCB, **POSITION** ROD MOTION Switch to WITHDRAW. **OBSERVE** that the RODS OUT Direction Lamp lights. _____

4. **OBSERVE** Bank Step Counters for proper rod motion, overlap and sequencing. _____

5. **VERIFY** the rods are moving out by **OBSERVING** the Digital Rod Position Indication System Display. _____

6. At the MCB, **STOP** rod motion by **RELEASING** the ROD MOTION Switch allowing it to return to the neutral position. **VERIFY** the RODS OUT Direction Lamp extinguishes. _____

7. **IF** necessary,
THEN REPEAT Steps 5.4.2.3 through 5.4.2.6. _____

8. **IF** using this Section to withdraw rods for temperature control only,
AND IF automatic rod control is desired,
THEN RESTORE the ROD BANK SELECTOR Switch to AUTO. _____

59. 2009A NRC RO 059

Given the following plant conditions:

Initial conditions:
- The plant was operating at 100% power

Current conditions:
- Control Bank 'D' Rod H-2 has just dropped to 170 steps
- ALL other Control Bank 'D' Rods are currently indicating 218 steps

Which ONE of the following describes the effect on AFD for the channel NEAREST the dropped rod and the operational implications if the dropped control rod is not recovered within the time constraints provided by Tech Specs?

A.✓ AFD will be more negative;
Localized power peaking may occur during recovery resulting in clad damage

B. AFD will be more negative;
Dilutions performed to compensate for xenon build in will result in inadequate Shutdown Margin

C. AFD will be less negative;
Localized power peaking may occur during recovery resulting in clad damage

D. AFD will be less negative;
Dilutions performed to compensate for xenon build in will result in inadequate Shutdown Margin

Plausibility and Answer Analysis

A *Correct. With a dropped control rod, flux will be suppressed in this region of the core resulting in more negative AFD. With a dropped control rod, xenon will building in this region of the core. If the control rod is recovered after a long duration, when the flux is restored, xenon will burnout resulting in localized power peaking and possible clad damage. See RODCS Student Text OE (SOER-84-2).*

B *Incorrect. With a dropped control rod, flux will be suppressed in this region of the core resulting in more negative AFD. With a dropped control rod for a long duration, dilutions for reactivity changes may be required but these reactivities will offset each other and shutdown margin will not become inadequate.*

C *Incorrect. With a dropped control rod, xenon will building in this region of the core. If the control rod is recovered after a long duration, when the flux is restored, xenon will burnout resulting in localized power peaking and possible clad damage. See RODCS Student Text OE (SOER-84-2). However, AFD will not be less negative. Flux will be suppressed and AFD will be more negative.*

D *Incorrect. With a dropped control rod for a long duration, dilutions for reactivity changes maybe required but these reactivities will offset each other and shutdown margin will not become inadequate. Additionally, AFD will not be less negative. Flux will be suppressed and AFD will be more negative.*

KA Statement - Nuclear Instrumentation - Knowledge of the operational implications of the following concepts as they apply to the (SYSTEM): (CFR: 41.5 / 45.7) Axial flux imbalance, including long-range effects

Importance Rating:	3.3 3.7
Technical Reference:	AOP-001 Rev. 32, page 17, page 39 T.S. Bases for 3/4.2.1, pages 401 & 402
References to be provided:	None
Learning Objective:	LP-AOP-3.01, Obj. 4
Question Origin:	Modified from March 2008 Exam RO#29 (001A3.04)
Comments:	

Origin:	MODIFIED	Cog Level:	H
Difficulty:	3	Reference:	RODCS STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	015K5.11	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.4 Misaligned Control Rod

☐ 1. **RECORD** the time at which rods became misaligned: **[C.2]**

Time _______

CAUTION

Movement of control rods, except as directed by this procedure, may cause further misalignment, or worsen an already adverse core power distribution.

✱ ☐ 2. **MINIMIZE** rod motion and power level changes, except as directed by this procedure.

3. **CONFIRM** that a rod misalignment exists:

☐ a. **REFER TO** Attachment 1, Indications of Misaligned Rod.

b. **CHECK** the following:

- ☐ Group step counter indication
- ☐ DRPI
- ☐ Quadrant power tilt ratio (QPTR)
- ☐ Axial flux difference (AFD)

RESPONSE NOT OBTAINED

3. **GO TO** ONE of the following:

- ☐ **IF** a control rod group is misaligned from the associated bank, **THEN GO TO** Section 3.5, Misaligned Control Rod Group.
- ☐ **IF** a rod or group misalignment is not confirmed, **THEN GO TO** Section 3.6, Malfunctioning Rod Position Indicator.

4. **CHECK** ALL Rod Control Power and Logic Cabinets for normal operation, as follows:

- ☐ NO blown fuses
- ☐ NO other visible malfunctions

RESPONSE NOT OBTAINED

☐ 4. **DIRECT** Maintenance to perform corrective action on ALL affected Rod Control Power and Logic cabinets.

Attachment 1 – Indications of Misaligned Rod

Sheet 1 of 1

The table below indicates the variation in plant parameters which may be indicative of rod misalignment. This variation refers to relative changes in indication from a reference condition at which the suspect rod's position was known to be properly aligned. The reference case may be taken from prior operating records, or it may be updated each time the proper rod positioning is verified by in-core measurements. In general, greater misalignment will cause larger variations. Variations in NI channel indication are also affected by the core location of the suspect rod. For example, a misaligned rod that is closest to the N-44 detector should indicate that N-44 flux parameters are abnormal when compared with flux parameters of the other Power Range NI channels. If the parameters below exhibit no abnormal variations with an individual DRPI differing from its group step counter demand position by more than 12 steps, it is probably a rod position indication problem.

PLANT PARAMETER	VALUE INDICATIVE OF ROD MISALIGNMENT
Quadrant Power Tilt Ratio (QPTR)	Greater than 1.02
Power Range Instrumentation	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Delta Flux Indicators	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Core Outlet Thermocouples	Greater than 10°F difference between thermocouples adjacent to the misaligned rod and the average of symmetric thermocouples (**PERFORM** Attachment 2)
Axial Flux Traces (in-core movable detector)	**CONSULT** Reactor Engineering **AND EVALUATE** using in-core movable detectors per EST-922, Control Rod Position Determination Via Incore Instrumentation

--END OF ATTACHMENT 1--

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the design DNBR value during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power;

$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power, with an allowance to account for measurement uncertainty.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference (target AFD) is determined at equilibrium xenon conditions. The rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target AFD at RATED THERMAL POWER for the associated core burnup conditions. Target AFD for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic measurement of the target flux difference value is necessary to reflect core burnup considerations. The target AFD may be updated between measurements based on the change in the predicted value with burnup.

AXIAL FLUX DIFFERENCE (Continued)

The target band about the target AFD is specified in the COLR. The target band limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits.

The computer determines the one-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the acceptable AFD target band. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

QUESTIONS REPORT
for NRC 2007 & 2008

1. 001 A3.04 002/BANK/WCNOC 2007 NRC/HIGHER//RO/HARRIS/3/2008/NO

Initial conditions:

- The plant is at 80% Reactor power and ramping UP at 1/2% per minute.
- Tave and Tref are matched.
- Rod Control is in AUTOMATIC.

Current conditions:

- ONE (1) Control Bank D Rod has just dropped and is stuck at 170 steps.
- All Other Control Bank D Rods are currently indicating 206 Steps.

Assuming NO operator action is taken, which ONE of the following describes the DEMAND for rod motion, and the effect on Axial Flux Difference (AFD) for the channel NEAREST the control rod problem?

	ROD MOTION	AFD EFFECT
A.	Outward	Less Negative
B.✓	Outward	More Negative
C.	Inward	Less Negative
D.	Inward	More Negative

B is correct. With a dropped/slipped control rod during a power escalation, rate of change of turbine load vs. reactor power will turn towards the negative, requiring outward demand. (Turbine load rate of change would remain constant while reactor power will drop or slow rate of increase) Also, Tave will lower because of the dropped/slipped rod. Rod slip/drop will cause flux to shift toward the bottom detector in a power range NI, making AFD go more negative.

A is incorrect because AFD will become more negative. To make positive, rod would have to be withdrawn or boration would be performed.

C is incorrect because rod motion would be outward, and AFD would be more negative.

D is incorrect because rod motion would be outward.

The KA topic is met because discussion of AFD in one quadrant (1 PRNI detector), and 1 rod dropping is related to QPTR, which is radial imbalance. Though AFD is the term used, it is used in terms of 1 indicator, not in terms of AFD due to a rod group position.

QUESTIONS REPORT
for NRC 2007 & 2008

Ability to monitor automatic operation of the CRDS, including: Radial imbalance

Question Number: 29

Tier: 2
Group: 2

Importance Rating: 3.5

Technical Reference: ROD CS

Proposed references to be provided to applicants during examination:

Learning Objective: ROD CS-9

Question History: Actual item selected was Byron 2001 but item has also been used for Wolf Creek 2007 (Bank 41860 selected)

10 CFR Part 55 Content: 55.41.7

Comments:

Category 1:	BANK	Category 2:	WCNOC 2007 NRC
Category 3:	HIGHER	Category 4:	
Category 5:	RO	Category 6:	HARRIS
Category 7:	3/2008	Category 8:	NO

60. 2009A NRC RO 060

Given the following plant conditions:

- EPP-004, Reactor Trip Response, has been entered following a reactor trip
- The crew is verifying Natural Circulation conditions as a result of a loss of power to all RCPs
- Five (5) core exit thermocouples have been identified as failed but no additional actions have been taken to address these thermocouples

Which ONE of the following describes the expected indication on the RVLIS Plasma Display for the failed thermocouples and the effect on the Subcooling Indication used to verify Natural Circulation?

	Expected Indication	RCS Subcooling Indication
A.	50°F	MORE subcooling than actual
B.✓	50°F	SAME subcooling as actual
C.	2500°F	LESS subcooling than actual
D.	2500°F	SAME subcooling as actual

Plausibility and Answer Analysis

A *Incorrect. This is plausible because thermocouples will fail to an indication of 50°F and if this value were used it would indicate MORE subcooling than actual. However, the subcooling calculator uses the average of the five hottest.*

B *Correct. The ICCM will set failed thermocouples to 50°F. The subcooling calculator uses the average of the five hottest.*

C *Incorrect. The ICCM will set failed thermocouples to 50°F. Plausible if candidate believes that thermocouples fail high as RTDs fail. The second half is plausible if it is believed that this value is used, but the subcooling calculator uses the average of the five hottest.*

D *Incorrect. The ICCM will set failed thermocouples to 50°F. Plausible if candidate believes that thermocouples fail high as RTDs fail. The second half is correct since the subcooling calculator uses the average of the five hottest.*

KA Statement - In-core Temperature Monitor - Knowledge of the effect that a loss or malfunction of the (SYSTEM) will have on the following: (CFR: 41.7 / 45.6) Natural circulation indications

Importance Rating:	3.5 3.7
Technical Reference:	OST-1020 Rev. 18 ICCM Student Text, page 10
References to be provided:	None
Learning Objective:	ICCM, Obj. 3b
Question Origin:	BANK OIT Development Bank 017 K3.01 002
Comments:	Reformated the question for question simplification. Thermocouples are used to monitor the temperature of the fluid just above the vessel's upper-core plate (range -245° to 2500°F). The system must be able to measure up to 2300°F per Reg. Guide 1.97. When a thermocouple fails a reading of 50°F will be displayed for that thermocouple on the MCR RVLIS display screen.

TERRY

Origin:	BANK	Cog Level:	H
Difficulty:	3	Reference:	ICCM STUDENT TEXT
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	017K3.01	K/A 2:	

If at least one of the narrow range channels has a good quality code, then the lowest of the narrow range pressures is used. Otherwise, the lowest of the wide range pressures is used.

A saturation temperature is calculated based on PRC9445. This saturation temperature is then compared with the average of the 5 hottest of all 51 operable thermocouples (Point TRC9300) **OR** the average of RCS wide range T_H temperatures, TE-413/423/433 (Point TRC9413), used only if TRC9300 is not available.

A value of subcooling will be calculated using one of the following equations:

Equation 1: Subcooling Calculations

$$T_{subcool} = T_{SAT} - TRC9300 \qquad T_{subcool} = T_{SAT} - TRC9413$$

Negative numbers for point TRC9400 (subcooling) indicate a superheated condition.

Incore Thermocouples (T/Cs)

Thermocouples are used to monitor the temperature of the fluid just above the vessel's upper-core plate (range -245 to 2500°F). The system must be able to measure up to 2300°F per Reg. Guide 1.97.

When a thermocouple fails or is taken out of service, a reading of 50°F will be displayed for that thermocouple on the MCR RVLIS display screen.

Train "A" ICCM has 26 T/Cs and Train "B" ICCM has 25 T/Cs. The type of T/C used is a Type-K chromel-alumel. Each train has a sealed Reference Junction Box (RJB) associated with its thermocouples. Both reference junction boxes are located on the left-hand wall at the seal table room entrance (261' el, containment).

Temperature of the RJBs is measured by three RTDs per train. The average of the good quality code (i.e., valid) RTDs is used for reference junction temperature.

The range of measurement of the RJB RTDs is 32° to 762°F. Nominal temperature of the RJB at 100 percent power is about 115°F.

The reference junction box temperature is not maintained at a given point, but is accurately measured so that a correction factor can be added to the thermocouple outputs by the ICC Monitor microprocessor to determine the actual Core Exit temperature measured.

61. 2009A NRC RO 061

Given the following plant conditions:
- The plant is in Mode 5 with the RCS loops filled
- Train 'B' RHR is operable and in operation
- Train 'A' RHR pump tripped due to a motor electrical fault
- The following indications are observed:
  - SG 'A' Level is 28%
  - SG 'B' Level is 28%
  - SG 'C' Level is 24%

Which ONE of the following describes the Tech Spec action requirements (if any) for the conditions listed above?

A. No actions required provided that Train 'B' RHR remains operable and in operation **OR** narrow range levels remain greater than 25% in two S/Gs.

B. No actions required provided that Train 'B' RHR remains operable and in operation **AND** narrow range levels remain greater than 25% in two S/Gs.

C.✓ Immediately initiate corrective actions to return Train 'A' RHR to operable **OR** restore two S/Gs narrow range levels to greater than 30% as soon as possible.

D. Immediately initiate corrective actions to return Train 'A' RHR to operable **AND** restore two S/Gs narrow range levels to greater than 30% as soon as possible.

Plausibility and Answer Analysis

A *Incorrect. Plausible since this would be correct in mode 6. Mode 6 with cavity level >23', only one RHR pump is required to operable and SG NR Level in the EOP Network is maintained >25% (non-adverse).*

B *Incorrect. Plausible because in Mode 6 only one RHR pump is required to operable and SG NR Level in the EOP Network is maintained >25% (non-adverse).*

C *Correct. T.S. 3.4.1.4.1 requires one RHR pump operable and in operation with the other RHR pump operable* ***OR*** *two steam generators >30 % NR level.*

D *Incorrect. Plausible if the candidate does not recognize that either the RHR pump on SG NR Level is required NOT both.*

KA Statement - Steam Generator - Ability to recognize system parameters that are entry level conditions for Technical Specifications

Importance Rating:	3.9 4.6
Technical Reference:	T.S. 3.4.1.4.1 (page 231)
References to be provided:	None
Learning Objective:	RHR, Obj. 9b
Question Origin:	BANK OIT Exam Bank RCS (13) 2
Comments:	

Origin:	BANK	Cog Level:	H
Difficulty:	4	Reference:	T.S. 3.4.1.4.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	035G2.2.42	K/A 2:	

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 74% wide range (WR) or greater than 30% narrow range (NR).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

62. 2009A NRC RO 062

Given the following plant conditions:
- The plant is operating at 100% power
- A leak has developed in the DEH system
- DEH fluid pressure and reservoir level are lowering

The following indications are received:
- ALB-020-4-2B, EH Fluid Low Press, is in alarm
- ALB-020-4-4A, EH RSVR Low Level, is in alarm
- ALB-020-4-4B, EH RSVR Low-Low Level, is in alarm
- The LFT LOCKOUT relay on Generator Protective Relay Panel 1A has tripped

Which ONE of the following describes the operation of the standby DEH Pump if DEH fluid pressure and reservoir level continue to lower?

The standby DEH Pump will. . .

A. start when DEH fluid pressure lowers to 1450 psig.

B. start when DEH fluid pressure lowers to 1500 psig.

C.✓ NOT start because the LFT LOCKOUT relay has tripped.

D. NOT start because the EH RSVR Low-Low Level annunciator is in alarm.

Plausibility and Answer Analysis

A *Incorrect. This is plausible because the standby DEH pump would normally start on lowering pressure at 1500 psig and 1450 psig comes from ALB-20-4-2A (low low press) but with the LFT LOCKOUT tripped the auto pump start is defeated.*

B *Incorrect. This is plausible because the standby DEH pump would normally start on lowering pressure at 1500 psig but with the LFT LOCKOUT tripped the auto pump start is defeated.*

C *Correct. Per APP-ALB-020 for window 4-4B, LFT LOCKOUT relay defeats auto pump start.*

D *Incorrect. This is plausible because APP-ALB-020 for window 4-4B lists the auto actions for this alarm and deenergizing reservoir heaters and defeating of the auto pump start are both listed but defeating of the auto pump start occurs at a lower level (7.625 inches vice 11.75 inches).*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Main Turbine Generator - Ability to monitor automatic operations of the (SYSTEM) including: (CFR: 41.7 / 45.5) Electrohydraulic control

Importance Rating:	2.6 2.9
Technical Reference:	APP-ALB-020 Rev. 27 Window 4-2A APP-ALB-020 Rev. 27 Window 4-2B APP-ALB-020 Rev. 27 Window 4-4B OP-131.05 Rev. 18, P&L #10, page 4
References to be provided:	None
Learning Objective:	DEH, Obj. 6d
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	APP-ALB-020
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	045A3.05	K/A 2:	

4-2A

DEVICES:
PS-01TA-4106AV (63-1/LP)

SETPOINT:
1450 psig

EH FLUID LOW LOW PRESS

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PI-4221, DEH Fluid Pressure indication
   b. PI-4220A and PI-4220B, Local DEH Pump discharge pressure indicators
2. **VERIFY** Automatic Functions:
   a. Standby DEH Pump starts at 1500 psig, as sensed by PS-01TA-4223V
3. **PERFORM** Corrective Actions:
   a. **IF** the Reactor is tripped,
      THEN GO TO EOP-PATH-1.
   b. **START** the standby DEH Pump.
   c. **DISPATCH** an operator to perform the following:
      (1) **MONITOR** DEH Pump operation.
      (2) **VERIFY OPEN** the following:
         (a) 1EH-1, A EH Pump Suction Vlv
         (b) 1EH-7, A EH Pump Disch Vlv
         (c) 1EH-8, B EH Pump Suction Vlv
         (d) 1EH-14, B EH Pump Disch Vlv
         (e) 1EH-99, A Train Trip Test Press Switch 63-1 Isol Vlv
      (3) **INVESTIGATE** system for leaks.
      (4) **IF** a leak is found,
         THEN ISOLATE the leak
         AND IMMEDIATELY NOTIFY Control Room.
   d. **IF** a leak is found,
      THEN:
      (1) **REFER TO** PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification, to classify leak.
      (2) **REFER TO** PLP-500 for notification of proper agencies.

CAUSES:

1. Filter high differential pressure
2. Pump trip
3. Relief valve malfunction
4. DEH pump pressure control valve malfunction
5. Leakage
6. Suction strainer clogged
7. Check valve failure
8. Improper valve alignment
9. Alarm circuit or instrument malfunction

REFERENCES:

1. 6-B-401 1446
2. EOP-PATH-1
3. 5-S-0553 S02
4. PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification

4-2B

DEVICES: PS-01TA-4224V (63/LP) **SETPOINT**: 1600 psig

EH FLUID LOW PRESS

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PI-4221, DEH Fluid Pressure indication
   b. PI-4220A and PI-4220B, Local DEH Pump discharge pressure indicators

2. **VERIFY** Automatic Functions:
   a. Standby DEH Pump starts at 1500 psig, as sensed by PS-01TA-4223V

3. **PERFORM** Corrective Actions:
   a. **IF** the Reactor is tripped,
      THEN GO TO EOP-PATH-1.
   b. **START** the standby DEH Pump.
   c. **DISPATCH** an operator to perform the following:
      (1) **MONITOR** DEH Pump and PCV operation.
      (2) **VERIFY OPEN** the following:
         (a) 1EH-1, A EH Pump Suction Vlv
         (b) 1EH-8, B EH Pump Suction Vlv
         (c) 1EH-31, Main Hdr Press Switch Isol Vlv
      (3) **INVESTIGATE** system for leaks.
      (4) **IF** a leak is found,
         THEN ISOLATE the leak
         AND IMMEDIATELY NOTIFY Control Room.
   d. **IF** a leak is found,
      THEN:
      (1) **REFER TO** PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification, to classify leak.
      (2) **REFER TO** PLP-500 for notification of proper agencies.

CAUSES:

1. Filter high differential pressure
2. Pump trip
3. Relief valve malfunction
4. DEH pump pressure control valve malfunction
5. Leakage
6. Suction strainer clogged
7. Check valve failure
8. Improper valve alignment
9. Alarm circuit or instrument malfunction

REFERENCES:

1. 6-B-401 1448
2. EOP-PATH-1
3. PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification

4-4B

DEVICES: LS-01TA-4221V (71/FL-1) **SETPOINT**: 11.75 IABT (inches above bottom of tank)

EH RSVR LOW-LOW LEVEL

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using local indications as directed in Corrective Actions step.

2. **VERIFY** Automatic Functions:
   a. DEH Reservoir heaters de-energize at low-low level of 11.75 inches

NOTE
The DEH Pump lockout relay will prevent an auto start of the standby DEH pump. The operator can still start the standby pump manually if indications warrant.

   b. Auto start of the standby DEH Pump is defeated at 7.625 inches.

3. **PERFORM** Corrective Actions:
   a. **IF** Reactor trips,
      THEN GO TO EOP-PATH-1.
   b. **DISPATCH** an operator to perform the following:
      (1) **CHECK** local level indication.
      (2) **IF** level is low,
          THEN:
          (a) **VERIFY SHUT** 1EH-127, Press Drn Line Sample Vlv.
          (b) **VERIFY SHUT** 1EH-141, LG-01TA-4221 & EH Reservoir Drain Valve.
          (c) **INSPECT** system for leaks **AND INFORM** Control Room of leak status.
          (d) **IF** NO excessive leaks are found,
              THEN ADD DEH fluid to reservoir.
   c. **IF** excessive leaks are found,
      THEN:
      (1) **IF** necessary,
          THEN SHUT DOWN to effect repairs.
      (2) **REFER TO** PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification, to classify leak.
      (3) **REFER TO** PLP-500 for notification of proper agencies.

CAUSES:

1. Improper valve alignment
2. System leakage
3. Alarm circuit or instrument malfunction

REFERENCES:

1. 6-B-401 1373, 1448
2. EOP-PATH-1
3. 5-S-0553 S02
4. PLP-500, Fish Kill Reporting, Hazardous Substances Release Notification, Non-Routine Radioactive Release Notification, and Oil Spill Notification

3.0 PREREQUISITES

1. EHC Reservoir is filled with FYRQUEL EHC.
2. Turbine Building Service Water is available to EHC coolers as per OP-139.
3. The EH Fluid temperature is greater than 70°F.
4. The low pressure accumulators are charged with N_2 to 24 to 30 psi.
5. The high pressure accumulators are charged with N_2 to 1150 to 1300 psi.
6. The system is filled and vented (including the pump casing if after an overhaul).
7. The AC Electrical Lineup complete per OP-156.02.

4.0 PRECAUTIONS AND LIMITATIONS

1. Prolonged operation with hydraulic fluid below 70°F is not recommended.
2. Operation with hydraulic fluid below 50°F is prohibited.
3. High pressure side discharge filters must be changed when the delta P alarms at 100 psid.
4. Do not use solvents containing chlorides to clean any EHC components.
5. EHC fluid is toxic. Avoid contact with skin, inhalation, or oral ingestion. Refer to Material Safety Data Sheet for safety precautions.
6. EHC fluid can damage electrical wire jacketing. Clean any spills immediately and contact Unit SCO.
7. New EHC FLUID (FYRQUEL EHC) must be filtered prior to being used.
8. Recirculate FYRQUEL EHC through two ten micron filters for a minimum of 15 minutes before adding to the reservoir.
9. Reference PLP-500 to evaluate leaks/spills.
10. The DEH pump Lockout relay (LFT LOCKOUT) will prevent an auto start of the standby DEH pump. The operator still has the ability to manually start the standby pump if indications warrant. (PCR-4495)

63. 2009A NRC RO 063

Given the following plant conditions:

- A Release of Waste Gas Decay Tank 'J' is in progress in accordance with OP-120.07, Waste Gas Processing

Which ONE of the following Radiation Monitors will initiate an automatic isolation of the Waste Gas Release in response to high activity and which setpoint is used during this release?

	Radiation Monitor	Setpoint to be used
A.	RM-3546, Stack 5 PIG	fixed setpoint from the RMDSL
B.✓	RM-3546, Stack 5 PIG	setpoint listed in the Release Package
C.	RM-3546-1, Stack 5 WRGM	fixed setpoint from the RMDSL
D.	RM-3546-1, Stack 5 WRGM	setpoint listed in the Release Package

Plausibility and Answer Analysis

The RMDSL lists the setpoint for each Rad Monitor (which the HPs use to verify operabilty of the Rad Monitors) but during a release the setpoint is changed to that found in the release package.

A *Incorrect. RM-3546, Stack 5 PIG will shut 3WG-229 to isolate the release but the setpoint is changed in the Rad Monitor to the setting listed in the Release Package IAW OP-120.07 section 8.12.*

B *Correct. RM-3546, Stack 5 PIG will shut 3WG-229 to isolate the release and the setpoint is changed in the Rad Monitor to the setting listed in the Release Package IAW OP-120.07 section 8.12.*

C *Incorrect. RM-3546-1, Stack 5 WRGM monitors the release stream and its setpoints are changed IAW OP-120.07 section 8.12, but the PIG initiates release isolation not the WRGM.*

D *Incorrect. RM-3546-1, Stack 5 WRGM monitors the release stream and its setpoints are changed IAW OP-120.07 section 8.12, but the PIG initiates release isolation not the WRGM.*

KA Statement - Waste Gas Disposal - Knowledge of (SYSTEM) design feature(s) and or interlock(s) which provide for the following: (CFR: 41.7) Isolation of waste gas release tanks

Importance Rating:	2.9 3.4
Technical Reference:	OP-120.07 section 8.12 AOP-005-BD Rev. 6, page 11
References to be provided:	None
Learning Objective:	LP-RWO-3.0, Obj. 5
Question Origin:	NEW
Comments:	From HPP-780, **RMDSL** - Radiation Monitor Data Sheet Library

Need title for OP-120.07

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AOP-005-BD
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	071K4.04	K/A 2:	

Attachment 2 — FHB Monitors

Step	Description
5	**I: Perform the following:** **Makeup to the Fuel Pool(s), as necessary.** **Verify proper operation of the Spent Fuel Pool Cooling and Cleanup System.** **Start the Fuel Pool Purification System. [A.3]** These actions ensure the system is operating properly, level in the pools is at the correct levels, and the Fuel Pool Purification System is placed in service to reduce activity, as required by FSAR Section 9.1.3.
6	**I: Refer to Tech Spec 3.9.11, Water Level - New And Spent Fuel Pools.** Water level is of concern for minimizing radiation levels in the FHB.
7	**I: Exit this attachment.** Actions provided in this attachment are completed. Other attachments may also apply. Therefore, direction is only to exit the attachment. It expected that a return to section 3.0, Operator Actions, will take place to determine if actions are complete for procedure exit.

Attachment 3 — RAB and WPB Monitors

Step	Description
1	**I: If REM-*1WV-3546 (WPB Stack 5) is in high alarm, then:** **a. Verify 3WG-229, WG Decay Tanks E & F To Plant Vent Vlv, is shut.** **b. Refer to OP-120.07, Waste Gas Processing.** This action isolates any release in progress and refers to the applicable procedure.

64. 2009A NRC RO 064

Given the following plant conditions:
- A rupture in the Instrument Air system has occurred
- Instrument Air header pressure is 85 psig and lowering slowly

Which ONE of the following describes the status of 1SA-506, Instrument Air from Service Air Isolation Valve, and the status of ALB-002-8-1, Instrument Air Low Pressure Alarm?

	1SA-506	Alarm status
A.	OPEN	Lit
B.	OPEN	NOT Lit
C.	CLOSED	Lit
D.✓	CLOSED	NOT Lit

Plausibility and Answer Analysis

A *Incorrect. 1SA-506 closes at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017). Alarm will NOT be lit. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.*

B *Incorrect. 1SA-506 will shut automatically at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017)*

C *Incorrect. 1 SA-506 will close automatically to isolate the Service Air System from the Instrument Air system, however the alarm is not lit until 75 psig. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.*

D *Correct. 1SA-506 shuts at 90 psig. The alarm however will not come in until 75 psig.*

KA Statement - Station Air - Knowledge of the physical connections and/or cause-effect relationships between (SYSTEM) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) IAS

Importance Rating:	3.0 3.1
Technical Reference:	AOP-017 Rev. 28, Attachment 7, page 57 APP-ALB-002 Rev 41, page 38
References to be provided:	None
Learning Objective:	LP-AOP-3.17, Obj. 2
Question Origin:	Modified from 2007 NRC Exam, RO#53, 065G2.1.28
Comments:	

Origin:	MODIFIED	Cog Level:	F
Difficulty:	3	Reference:	AOP-017
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	079K1.01	K/A 2:	

Attachment 7

Sheet 1 of 1

Significant Instrument Air Pressure Setpoints

NOTE

- Each compressor start listed below applies only when the applicable compressor is isolated from the CAS Control Panel Sequencer, or when power has been lost to the CAS Panel. If connected to the CAS Panel with the panel operating normally, the lead compressor will load at 98 psig and unload at 114 psig.
- For the compressor starts listed below, MCB pressure indication may be 7 psig below the given setpoint before action occurs, due to air dryer ΔP.
- Normal air receiver pressure is between 100 and 115 psig.

101 psig	--	Air Compressor 1C starts (if in STANDBY and isolated from CAS Panel)
96 psig	--	Air Compressor 1A starts (if in STANDBY and isolated from CAS Panel)
95 psig	--	Air Compressor 1B starts (if in STANDBY and isolated from CAS Panel)
90 psig	--	1SA-506 shuts to isolate Service Air from Instrument Air. (All stored air is available to Instrument Air System)
85 psig	--	RCS letdown flowpath valves begin to fail to mid-position
75 psig	--	Low air header pressure alarms on MCB for Instrument and Service Air.
60 psig	--	FW Flow Control valves auto shut (1FW-133, 1FW-191, 1FW-249)
35 psig	--	Remaining air-operated valves no longer considered reliable

--END OF ATTACHMENT 7--

8-1

DEVICES: PS-9750 **SETPOINT**: 75 psig

INSTRUMENT AIR LOW PRESS

REFLASH: NO

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   a. PI-9751.1, Instrument Air Header Pressure
2. **VERIFY** Automatic Functions:
   a. Standby Air Compressors start (1C at 101 psig; 1A at 96 psig; 1B at 95 psig)
   b. Service Air isolates (1SA-506 shuts at 90 psig in IA header)
   c. FW FRVs (1FW-133, 1FW-249, 1FW-191) shut at 60 psig.
3. **PERFORM** Corrective Actions:
   a. **IF** instrument air system pressure is low,
      THEN GO TO, AOP-017, Loss of Instrument Air.

CAUSES:

1. Heavy demand on instrument air system
2. Running compressor failure
3. Ruptured air header
4. Alarm circuit or instrumentation malfunction

REFERENCES:

1. SOER 81-9
2. AOP-017, Loss of Instrument Air
3. 6-B-401 2453

QUESTIONS REPORT
for NRC 2007 & 2008

1. 065 G2.1.28 002/NEW//LOWER//RO/HARRIS/7/2007/NO

Given the following:

- A loss of instrument air is in progress.
- Instrument Air header pressure is 70 psig and lowering slowly.

Which ONE (1) of the following describes the status of 1SA-506, Instrument Air from Service Air Isolation Valve, and the status of the Instrument Air Header Low Pressure alarm?

A. 1SA-506 is CLOSED;
Header Low Pressure Alarm is **NOT** lit.

B. 1SA-506 is OPEN;
Header Low Pressure Alarm is **NOT** lit.

C.✓ 1SA-506 is CLOSED;
Header Low Pressure Alarm is lit.

D. 1SA-506 OPEN;
Header Low Pressure Alarm is lit.

A is incorrect. Alarm will be lit
B is incorrect because position is incorrect.
C is Correct. 90 psig, valve closes to isolate potential leak in the Service Air System. Alarm lit at 75 psig
D is incorrect because position and alarm are incorrect

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Question Number: 53

Tier 1 Group 1

Importance Rating: RO 3.2

Technical Reference: AOP-017
Proposed references to be provided to applicants during examination: NONE
Learning Objective: ISA Text Obj 2
10 CFR Part 55 Content: 41.5

Comments:

Category 1:	NEW	Category 2:	
Category 3:	LOWER	Category 4:	
Category 5:	RO	Category 6:	HARRIS
Category 7:	7/2007	Category 8:	NO

65. 2009A NRC RO 065

Given the following plant conditions:

- An Ultra Violet Detector has failed in the EDG Building resulting in ALB-030-8-1, Fire Detection System Fire, going into alarm

Which ONE of the following describes the type of sprinkler system in the EDG Building and the status of the sprinkler system deluge valve in response to this condition?

	Sprinkler Type	Sprinkler Deluge Valve status
A.	Multicycle	Actuated
B.✓	Multicycle	NOT Actuated
C.	Wet Pipe	Actuated
D.	Wet Pipe	NOT Actuated

Plausibility and Answer Analysis

A *Incorrect. This is plausible because it lists the correct sprinkler type and UV detectors actuate the sprinkler system in the Security Bldg DG room but are for warning only in other areas.*

B *Correct. It lists the correct sprinkler type and UV detectors actuate the sprinkler system in the Security Bldg Diesel Generator room but are for warning only in other areas.*

C *Incorrect. This is plausible because UV detectors actuate the sprinkler system in the Security Bldg DG room but are for warning only in other areas. The sprinkler type is incorrect. The EDG building uses a multicycle. Other locations in the plant do use wet pipe sprinkler systems and the requirements of FPP-013 are for Multicycle, Preaction, and Wet Pipe systems.*

D *Incorrect. Correct that the deluge valve would not be actuated, but the sprinkler type is incorrect. The EDG building uses a multicycle. Other locations in the plant do use wet pipe sprinkler systems and the requirements of FPP-013 are for Multicycle, Preaction, and Wet Pipe systems.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Fire Protection - Knowledge of the effect that a loss or malfunction of the following will have on the (SYSTEM): Fire, smoke and heat detectors

Importance Rating:	2.6 2.9
Technical Reference:	OP-149 Rev. 38, page 10 ALB-030 Rev 22, page 28 Fire Protection Text
References to be provided:	None
Learning Objective:	FP, Obj. 3b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	2	Reference:	OP-149
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	086K6.04	K/A 2:	

4.0 PRECAUTIONS AND LIMITATIONS

30. The UV (Ultraviolet) detectors for the Security Diesel Generator will actuate the Security Building Pre-Action Sprinkler System. All other UV detectors are for warning only.

31. The Thermal detectors located over the 'A & B' Emergency Diesel Generator's Starting Air Compressors do not actuate the Emergency Diesel Generator Multi-Cycle Sprinkler Systems.

32. The Thermal detectors located around the Reactor Coolant Pumps do not actuate the Containment Multi-Cycle Sprinkler System.

33. Apply the following starting duty for the Motor Driven Fire Pump:

- Two successive starts are permitted if the motor is at ambient temperature.
- Subsequent starts are permitted when the motor has cooled by running for a period of 30 minutes, OR by standing idle for a period of 60 minutes.

DEVICES: MFDIC-CCU/CP703VP **SETPOINT**: None

FIRE DETECTION SYSTEM FIRE

REFLASH: **NO**

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   - ERFIS CRT fire alarm status screen via turn on code FIRE
2. **VERIFY** Automatic Functions:

NOTE
If fire condition alarm is from area supplied with automatic fire suppression system, automatic fire suppression may be activated.

CAUTION
If fire testing is in progress, the ERFIS FIRE Screen must be displayed and monitored in the Main Control Room to ensure a fire in the plant does not go undetected.

3. **PERFORM** Corrective Actions:
   a. **CHECK** fire points in alarm on ERFIS CRT fire alarm status screen via turn on code FIRE **AND DETERMINE** location of alarming detector.
   b. **IF** ERFIS is NOT available,
   THEN DISPATCH personnel to check any of the following to determine location of the alarming detector:
      - MFDIC MCR Display Console PC
      - MFDIC-CCU CPU(CP703VP) - Secondary Alarm Station
      - MFDIC Display Console PC - Secondary Alarm Station
      - MFDIC Hard-Wired Annunciator Panel - Secondary Alarm Station
   c. **DISPATCH** an Operator to validate the alarm.
   d. **IF** fire is present at location indicated,
   THEN GO TO FPP-002, Fire Emergency.
   e. **IF** NO fire is present,
   THEN RESET local detector panel
   AND SILENCE alarm on MFDIC.

CAUSES:

1. Fire Condition Alarm from MFDIC
2. Detector or alarm circuit malfunction

REFERENCES:

1. 6-B-401 9518
2. FPP-002
3. ~~PCR-0942~~
4. ESR 96-00511

66. 2009A NRC RO 066

Given the following plant conditions:
- AOP-004, Remote Shutdown, is in progress due to a fire in the MCR
- You have been directed to isolate the SI Accumulators

Which ONE of the following describes the required RCS pressure range to isolate the SI Accumulators in accordance with AOP-004 and where the SI Accumulator Discharge Valves controls are located?

	RCS Pressure Range	Location
A.	900-1000 psig	Auxiliary Control Panel
B.✓	900-1000 psig	Auxiliary Transfer Panels
C.	1900-1950 psig	Auxiliary Control Panel
D.	1900-1950 psig	Auxiliary Transfer Panels

Plausibility and Answer Analysis

A Incorrect. Plausible because the pressure range is correct and most components operated during AOP-004 are located on the ACP but the Discharge Valve Control Switches are on the ATPs.

B Correct. The SI Accumulator Discharge Valves are shut at the ATPs between 900 and 1000 psig.

C Incorrect. Pressure band is incorrect. Plausible because this pressure band is listed in AOP-004 but it is for blocking SI not isolating SI accumulators. Location is incorrect. Plausible because most components operated during AOP-004 are on the ACP.

D Incorrect. Pressure band is incorrect. Plausible because this pressure band is listed in AOP-004 but it is for blocking SI not isolating SI accumulators. The location is correct.

KA Statement - Conduct of operations - Ability to locate and operate components, including local controls.

Importance Rating:	4.4 4.0
Technical Reference:	AOP-004 Rev. 44 Attachment 1, Page 45-46
References to be provided:	None
~~Learning Objective:~~	~~SIS, Obj. 5a~~
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AOP-004
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.1.30	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.1 Remote Shutdown Due To Fire

NOTE

The 70°F RCS subcooling requirement only applies until the initial stages of RHR operation. Subsequent operations are controlled by RCS pressure/temperature limits shown on Curve F-X-4.

INSTRUCTIONS	RESPONSE NOT OBTAINED
ACP / Unit SCO ☐ 35. **CHECK** at least two CRDM fans RUNNING.	*ACP / Unit SCO* ✱☐ 35. **MAINTAIN** at least 120°F RCS subcooling during the remainder of cooldown and depressurization.
ACP / Unit SCO ✱☐ 36. **CHECK** PRZ level STABLE OR LOWERING.	*ACP / Unit SCO* 36. **IF** PRZ level rises unexpectedly, **THEN**: ☐ a. **REFER TO** Curve F-X-4 **AND RAISE** RCS pressure to within TS curve cooldown limits, to collapse potential RV head voids. ☐ b. **WHEN** within cooldown curve limits, **THEN CONTINUE** cooldown.
37. **WHEN** RCS pressure is 900 to 1000 psig, as indicated on PI-402.2, **THEN ISOLATE** SI accumulators: *286' RAB / RO with locked valve key* a. **UNLOCK AND TURN ON** accumulator discharge valve breakers: ☐ • Accumulator A: 1A21-SA-5C (both breakers) ☐ • Accumulator C: 1A21-SA-3D (both breakers) ☐ • Accumulator B: 1B21-SB-5C (both breakers)	

(Continued on Next Page)

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.1 Remote Shutdown Due To Fire

37. (continued)

b. **SHUT** SI accumulator discharge valves at the Auxiliary Transfer Panels listed:

Cable Vault A / RO with ATP cabinet key

☐ • 1SI-246, Accumulator A Discharge (at ATP A)

Cable Vault A / RO with ATP cabinet key

☐ • 1SI-248, Accumulator C Discharge (at ATP A)

Cable Vault B / RO with ATP cabinet key

☐ • 1SI-247, Accumulator B Discharge (at ATP B)

286' RAB / RO with locked valve key

c. **TURN OFF AND LOCK** accumulator discharge valve breakers:

☐ • Accumulator A: 1A21-SA-5C (both breakers)

☐ • Accumulator C: 1A21-SA-3D (both breakers)

☐ • Accumulator B: 1B21-SB-5C (both breakers)

67. 2009A NRC RO 067

Given the following plant conditions:
- The plant is in Mode 6
- Fuel Handling activities are in progress
- Boron concentration in the Refueling Cavity has been determined to be less than the required Refueling Boron Concentration

Which ONE of the following actions should be initiated immediately in accordance with GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity?

A.✓ Suspend core alterations;
Initiate Emergency Boration in accordance with AOP-002, Emergency Boration

B. Suspend core alterations;
Initiate Rapid Addition of Boric Acid to RCS in accordance with OP-107.01, CVCS Boration, Dilution, and Chemistry Control

C. Shut Fuel Transfer Tube Gate Valve;
Initiate Emergency Boration in accordance with AOP-002, Emergency Boration

D. Shut Fuel Transfer Tube Gate Valve;
Initiate Rapid Addition of Boric Acid to RCS in accordance with OP-107.01, CVCS Boration, Dilution, and Chemistry Control

Plausibility and Answer Analysis

A *Correct. GP-009 precaution and limitation states that if refueling boron concentration falls below COLR limits then emergency boration should be initiated immediately IAW with AOP-002 and suspend core alterations.*

B *Incorrect. Plausible since core alterations will be suspended, but GP-009 specifically directs use of AOP-002 in this situation. OP-107.01 provides guidance on performing Rapid Addition of Boric Acid to RCS however that section of OP-107.01 specifically states that it is only to be used when entry conditions for AOP-002 are NOT met.*

C *Incorrect. Plausible if candidate believes that isolation of the connection between Spent Fuel Pools and the refueling cavity is required for low boron concentration in the cavity. Emergency Boration in accordance with AOP-002 is correct.*

D *Incorrect. Plausible if candidate believes that isolation of the connection between Spent Fuel Pools and the refueling cavity is required for low boron concentration in the cavity. GP-009 specifically directs use of AOP-002 in this situation. OP-107.01 provides guidance on performing Rapid Addition of Boric Acid to RCS however that section of OP-107.01 specifically states that it is only to be used when entry conditions for AOP-002 are NOT met.*

KA Statement - Conduct of operations - Knowledge of RO duties in the control room during fuel handling.

Importance Rating:	3.9 3.8
Technical Reference:	GP-009 Rev. 47, P&L 9, page 12
References to be provided:	None
Learning Objective:	LP-GP-3.09, Obj. 3
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	GP-009
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.1.44	K/A 2:	

4.0 PRECAUTIONS AND LIMITATIONS (continued)

3. Strict adherence to Radiation Work Permits and ALARA practices is required during refueling operations.

4. If fuel loading is suspended, continue surveillance testing at the established frequencies. Count rates from both Source Range channels should be monitored and recorded per OST-1033, to evaluate any possible trends. For channel check, counts may differ if evaluated to be caused by the fuel load pattern.

5. If fuel movement has been suspended for any reason other than to accommodate the planned work schedule, movement shall not be resumed until the conditions which caused the suspension have been understood and corrected, or have been evaluated and judged acceptable by the Superintendent - Shift Operations with concurrence of the SRO-Fuel Handling.

6. If a fuel assembly is to be stored in a temporary core location, follow the guidance provided in FHP-014. If the geometry conditions in FHP-014 cannot be achieved, the fuel assembly may be temporarily stored in one of the following:
   - the Manipulator Crane Mast, with the Manipulator Crane moved away from the Reactor Core if possible;
   - the Fuel Transfer System Upender;
   - the RCC Change Fixture.
   - The fuel assembly may always be returned to the FHB for temporary storage.

7. The detailed loading sequence may be altered if difficulty is encountered positioning an assembly provided that Precaution 4.0.6 is observed. Any difficulty in positioning a fuel assembly in the Reactor Core shall be evaluated to determine the cause.

8. When inserting a fuel assembly, counts on the Source Range Channels shall be monitored per FHP-014.

9. If RCS boron concentration decreases below the limits specified in the COLR (PLP-106), immediately initiate emergency boration per AOP-002. (Reference 2.2.13)

10. If a fuel assembly is dropped or otherwise damaged, refer to AOP-013.

11. All tools and loose equipment used over the Refueling Cavity shall have retrieving lines connected to a point above the operating deck, such that they may be easily retrieved if they are dropped.

68. 2009A NRC RO 068

Given the following plant conditions:
- The plant is operating at 100% power
- Troubleshooting is in progress on the A-SA ESCW Chiller in accordance with AP-929, Troubleshooting Guide
- Leads must be lifted in the Control Circuit to support the troubleshooting
- The lead lift does not pose a risk of personnel injury or equipment damage

Of the following, which are acceptable forms of documentation for the lead lift in accordance with AP-929?

1. Clearance Order (OPS-NGGC-1301, Equipment Clearance)
2. Verification Sign-Off Sheet (OPS-NGGC-1303, Independent Verification)
3. Component Manipulation Sign-Off Sheet (OPS-NGGC-1308, Plant Status Control)
4. Add to the Work Order Instructions at the time of lift (ADM-NGGC-0104, Work Management Process)

A. 1 and 2

B.✓ 2 and 3

C. 3 and 4

D. 1 and 4

Plausibility and Answer Analysis

A *Incorrect. OPS-NGGC-1303 is correct as listed in P&L#7 of AP-929 but OPS-NGGC-1301 is not. OPS-NGGC-1301 does maintain configuration but is used for safety of personnel and equipment and should not be used for configuration alone.*

B *Correct. Both OPS-NGGC-1303 and* OPS-NGGC-1308 are listed i*n P&L#7 of AP-929*

C *Incorrect. OPS-NGGC-1308 is correct as listed in P&L#7 of AP-929 but adding to the Work Order Instructions at the time of lift is not. OPS-NGGC-1308 does allow use of a work order to maintain configuration but the components must be added to the work order during planning, not in the field.*

D *Incorrect. OPS-NGGC-1301 does maintain configuration but is used for safety of personnel and equipment and should not be used for configuration alone. OPS-NGGC-1308 does allow use of a work order to maintain configuration but the components must be added to the work order during planning, not in the field.*

KA Statement - Equipment Control - Knowledge of the process for managing troubleshooting activities.

Importance Rating:	2.6 3.8
Technical Reference:	AP-929 Rev. 13, page 15
References to be provided:	None
Learning Objective:	
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AP-929
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.2.20	K/A 2:	

5.0 PROCEDURE

5.1. Precautions and Limitations

1. Standard safety precautions for working on energized or operating equipment shall be adhered to. Refer to SAF-NGGC-2172 and SAF-NGGC-2175.

2. When a complex test or abnormal system lineup is needed to facilitate troubleshooting, an approved procedure with detailed instructions may be required. Refer to ADM-NGGC-0104.

3. Whenever a discrepancy is discovered between installed equipment and the documents being utilized for troubleshooting, a resolution shall be obtained from the implementing organizations responsible supervisor prior to proceeding and a CR written if required.

4. TCFs shall not be used to gather plant performance data or install Temporary Modifications.

5. If at any time during troubleshooting, unexpected plant equipment responses are observed, stop work, place the system/plant in a safe condition, and reevaluate the troubleshooting plan before proceeding. A CR should be written if required.

6. If, during troubleshooting, it is determined that troubleshooting must be aborted, one of the following shall be performed:

   - Equipment is returned to its design configuration
   - Equipment is tagged out of service per OPS-NGGC-1301
   - A Temporary Modification is developed and installed per EGR-NGGC-0005
   - Use of the off-normal process in OMM-001

7. Lifting of leads, installation of jumpers, and so forth shall be documented on one of the following:

   - A Verification Sign-Off Sheet (OPS-NGGC-1303)
   - A Component Manipulation Sign-Off Sheet (OPS-NGGC-1308)
   - In the body of the troubleshooting plan.

8. When necessary to move wiring during troubleshooting, inspect Terminations for tightness. Obtain proper authorization, including initiation of a new W/O if required, before tightening Terminations outside the scope of the existing W/O. (ACFR 91-00208)

69. 2009A NRC RO 069

Which ONE of the following describes the setpoint and Technical Specification Basis for the High Pressurizer Water Level reactor trip?

A. 78%;
Provides a backup trip to PZR High Pressure reactor trip and ensures that water relief through the PZR safety valves will NOT occur

B. 78%;
Provides primary protection for loss of load events and ensures that the PZR safety valves will NOT lift

C.✓ 92%;
Provides a backup trip to PZR High Pressure reactor trip and ensures that water relief through the PZR safety valves will NOT occur

D. 92%;
Provides primary protection for loss of load events and ensures that the PZR safety valves will NOT lift

Plausibility and Answer Analysis

A *Incorrect. Setpoint is for the High Steam Generator Water Level reactor trip. Plausible if examinee confuses setpoints. Reason is correct.*

B *Incorrect. Setpoint is for the High Steam Generator Water Level reactor trip. Plausible if examinee confuses setpoints. Reason is incorrect, the trip does not provide primary protection. It is a backup for loss of load events.*

C *Correct. TS Basis 3.3.1, provides for backup to PZR High Pressure. If a level channel failed, Pressure overshoot from the low rate of charging flow would not cause SV lift prior to reactor trip.*

D *Incorrect. Setpoint is correct, however the trip does not provide primary protection. It is a backup for loss of load events.*

KA Statement - Equipment Control - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Importance Rating:	3.2 4.2
Technical Reference:	T.S. Table 2.2-1 (page 84) T.S. 2.2.1 Bases (page 98)
References to be provided:	None
Learning Objective:	RPS, Obj. 12c
Question Origin:	BANK
Comments:	

Origin:	BANK	Cog Level:	F
Difficulty:	3	Reference:	T.S. 3.3.1
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.2.25	K/A 2:	

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

**RTP = RATED THERMAL POWER

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

70. 2009A NRC RO 070

Given the following plant conditions:

- The plant was operating at 100% power
- A Steam Generator Tube Rupture has occurred in the 'B' Steam Generator
- The crew is performing the actions in PATH-2 to isolate the 'B' Steam Generator

Which ONE of the following describes the actions required in accordance with PATH-2 for the controller of the 'B' Steam Generator PORV to minimize radiation releases?

A. Verify the PORV controller setpoint at 84% and place in AUTO

B.✓ Adjust the PORV controller setpoint to 88% and place in AUTO

C. Place the PORV controller in MANUAL and adjust the demand to 0%

D. Place the PORV controller in MANUAL and ONLY open the PORV if pressure exceeds 1145 psig

Plausibility and Answer Analysis

A Incorrect. A Setpoint of 84% and in AUTO is the normal lineup for the PORV controller and it will minimize radioactive releases but the desire is to further minimize releases so the correct answer is to adjust the setpoint to 88%.

B Correct. PATH-2 directs adjusting the PORV controller setpoint to 88% and placing in AUTO to minimize radioactive releases.

C Incorrect. The action to place the PORV controller in MANUAL and adjust the demand to 0% would be taken in PATH-2 but ONLY after the correct action was taken AND it was found that the controller was failed. No indications of a failed PORV are given in the stem.

D Incorrect. The action to place the PORV controller in MANUAL would be taken in PATH-2 but ONLY after the correct action was taken AND it was found that the controller was failed. No indications of a failed PORV are given in the stem. The 1145 psig comes from the setpoint of 88% in the correct action and PATH-2 checks ruptured SG pressure less the 1145 psig.

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Radiation Control - Ability to control radiation releases

Importance Rating:	3.8 4.3
Technical Reference:	PATH-2 Rev. 19, page 8 of Guide WOG ERG for PATH-2
References to be provided:	None
Learning Objective:	LP-EOP-3.2, Obj. 4b
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	2	Reference:	PATH-2
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.3.11	K/A 2:	

FOLDOUT C

- ALTERNATE MINIFLOW OPEN/SHUT CRITERIA

  - IF RCS pressure decreases to less than 1800 PSIG, THEN verify alternate miniflow isolation OR miniflow block valves - SHUT

  - IF RCS pressure increases to greater than 2200 PSIG, THEN verify alternate miniflow isolation AND miniflow block valves - OPEN

- SI REINITIATION CRITERIA

  Following SI termination, IF any of the following occurs:

  - RCS subcooling - LESS THAN 10°F [40°F] - C
    20°F [50°F] - M

  - PRZ level - CAN NOT BE MAINTAINED GREATER THAN 10% [30%]

  THEN perform the following:

  a. IF CSIP suction aligned to VCT, THEN realign to RWST.

  b. Shut charging line isolation valves AND open BIT valves.

  c. Verify normal miniflow isolation valves - SHUT

  d. IF necessary to restore conditions, THEN restart standby CSIP.

  e. IF reinitiation occurs after entry point S (Step 29), THEN GO TO EPP-020, "SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY", Step 1.

- SECONDARY INTEGRITY CRITERIA

  IF any of the following occurs, THEN GO TO EPP-014, "FAULTED STEAM GENERATOR ISOLATION", Step 1.

  - Any SG pressure - DECREASES IN AN UNCONTROLLED MANNER AND THAT SG HAS NOT BEEN ISOLATED

  - Any SG - COMPLETELY DEPRESSURIZED AND THAT SG HAS NOT BEEN ISOLATED

- COLD LEG RECIRCULATION SWITCHOVER CRITERIA

  IF RWST level decreases to less than 23.4% (2/4 Low-Low alarm), THEN GO TO EPP-010, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.

- AFW SUPPLY SWITCHOVER CRITERIA

  IF CST level decreases to less than 10%, THEN switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

- RHR RESTART CRITERIA

  IF RCS pressure decreases to less than 230 PSIG in an uncontrolled manner, THEN restart RHR pumps to supply water to the RCS.

- MULTIPLE TUBE RUPTURE CRITERIA

  IF any intact SG level increases in an uncontrolled manner OR any intact SG has abnormal radiation levels, THEN stop RCS depressurization AND cooldown AND RETURN TO Step 1.

71. 2009A NRC RO 071

A Source Check is being performed on the Plant Vent Stack Wide Range Gas Monitor, (RM-3509-1-SA).

The activity measured by the channel is lower than required when the source is exposed.

Which ONE of the following describes the expected indication for this condition?

A. The Check Source (C/S) button on the RM-11 console flashes

B. The Check Source (C/S) button on the RM-23 module flashes

C.✓ The symbol (**) is presented on the RM-11 screen indicating Channel Check Source Failed

D. The symbol (**) is presented on the RM-23 module indicating Channel Check Source Failed

Plausibility and Answer Analysis

A Incorrect. Plausible because the button is backlit during a source check.

B Incorrect. Plausible because this button is also backlit during a source check.

C Correct. This symbol will display to indicate the Channel Check Source has failed.

D Incorrect. Plausible because RM-23 is monitored during source check but RM-23 module does not have the indication of a source check failure. The RM-11 has indication that a source check has failed.

KA Statement - Radiation Control - Ability to use radiation monitoring systems

Importance Rating:	2.9 2.9
Technical Reference:	OST-1058 Rev. 11, page 4
References to be provided:	None
Learning Objective:	RMS, Obj. 5
Question Origin:	2007 NRC, RO #24
Comments:	

Origin:	PREVIOUS NRC	Cog Level:	F
Difficulty:	3	Reference:	OST-1058
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.3.5	K/A 2:	

7.0 PROCEDURE

7.1. Plant Vent Stack Wide Range Gas Monitor

7.1.1. Low Range

1. At the RM-11 Radiation Monitoring Computer Control Console, depress the key labeled GRID 1. _____

2. Select the Plant Vent Stack (RAB WRGM LOW RNG) Channel by:

   a. Keying in 1813. _____

   b. Depressing the SEL key. _____

3. Display the Channel Status Display by depressing the STATUS key. _____

> **NOTE:** If the status symbol (**) is displayed for any of the following the source check cannot be done.

4. Check the RM-11 Poll Status and RM-11 Communications by checking that the status symbol (**) <u>is not</u> displayed for any of the following:

   - Monitor Offline _____
   - Monitor Communications Failure _____
   - Channel Not Responding To Poll _____

5. At the RM-23, select the LOW RANGE Channel for RC-21AV-3509-1-SA, the Plant Vent Stack WRGM. _____

6. At the RM-23, depress the C/S function key. _____

7. At the RM-23, check the C/S function key lit. _____

8. At the RM-11, perform the following:

   a. Check the Check Source indication is lit. _____

   b. Check that the status symbol (**) is displayed to the Channel Check Source energized line. _____

9. When the C/S Function Key Light de-energizes, check that the status symbol (**) <u>is not</u> displayed for the Channel Check Source Test Failed indication. _____

10. If the status symbol (**) is displayed for Channel Check Source Test Failed indication, notify the Unit SCO and declare the channel inoperable. (Otherwise mark this step N/A) _____

72. 2009A NRC RO 072

Given the following plant conditions:
- A Refueling Outage is in progress
- You have been assigned a task in the RCA and have signed on to the Operations RWP
- When approaching the area where the task is to be performed, you note that the area is barricaded and conspicuously posted with a flashing red light

Which ONE of the following describes the classification of this area and the individual who approves entry?

	Area Classification	Individual Approving Entry
A.	Very High Radiation Area	Plant General Manager
B.	Very High Radiation Area	Radiation Protection Manager
C.	Locked High Radiation Area	Plant General Manager
D✓	Locked High Radiation Area	Radiation Protection Manager

Plausibility and Answer Analysis

A *Incorrect. A VHRA is barricaded, and conspicuously posted but use of a flashing red light is not allowed. It must be locked. The Plant General Manager must be notified of entry into a VHRA, but this is a LHRA and the Radiation Protection Manager will approve entry.*

B *Incorrect. A VHRA is barricaded, and conspicuously posted but use of a flashing red light is not allowed. It must be locked. The Plant General Manager must be notified of entry into a VHRA, but this is a LHRA and the Radiation Protection Manager will approve entry.*

C *Incorrect. A LHRA may be conspicuously posted by the use of a flashing red light IAW Tech Spec 6.12.2.f vice locked if it is not feasible to lock it. The Plant General Manager must be notified of entry into a VHRA, but this is a LHRA and the Radiation Protection Manager will approve entry.*

D *Correct. A LHRA may be conspicuously posted by the use of a flashing red light IAW Tech Spec 6.12.2.f vice locked if it is not feasible to lock it. This is a LHRA and the Radiation Protection Manager will approve entry.*

KA Statement - Radiation Control - Ability to comply with radiation work permit requirements during normal or abnormal conditions

Importance Rating:	3.5 3.6
Technical Reference:	Tech Spec 6.12.2.f AP-504 Rev. 28, page 9
References to be provided:	None
Learning Objective:	LP-PP-3.7, Obj. 2
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	F
Difficulty:	3	Reference:	AP-504
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.3.7	K/A 2:	

HIGH RADIATION AREA (Continued)

3. Possess a direct-reading dosimeter and be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device; who is responsible for controlling personnel exposure within the area; or

5. In those cases where the options of Specifications 6.12.2.d.2, 6.12.2.d.3, and 6.12.2.d.4 above, are impractical or determined to be inconsistent with the "As Low As Reasonably Achievable" principle, possess a radiation monitoring and indicating device.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

5.1 General (continued)

17. LHRAs and VHRAs shall be secured with Best lock "C" keyway cores and keys.

    - C keyway series RP1 cores and keys shall be used for VHRAs.
    - C keyway series RP2 cores and keys shall be used for LHRAs.

18. This step is included as an interim corrective action to NCR 261182. After the Containment Incore Sump VHRA lock core is changed to a C keyway, this step is no longer applicable and this procedure can be editorially corrected to reflect the change.

5.2 Entry into LHRAs

1. Use Attachment 1 to document entries into LHRAs.

2. The Lead RC Technician shall:

   a. Assign an RC Technician to cover the entry.

   b. Obtain approval for the entry as follows:

> **NOTE**: The RC Supervisor may authorize entries verbally (via phone, radio, etc.).

1) For LHRAs with dose rates less than 10 rem/h at 30 cm, obtain approval for the entry from the RC Supervisor. Signify approval by entering the supervisor's name in the "Approved By" section of Attachment 1 (or an equivalent form).

> **NOTE**: The RPM may authorize entries verbally (via phone, radio, etc.) as long as the form is signed upon their return to the site.

R2.1.16 2) For areas with dose rates equal to or greater than 10 rem/h at 30 cm, have the Radiation Protection Manager sign the "Approved By" section of Attachment 1. [SOER 01-1, Recommendation 6f]

   c. Ensure that the RC Technician providing LHRA coverage is briefed per AP-110 on the nature of the work and the expected radiological conditions.

73. 2009A NRC RO 073

Given the following plant conditions:

- The crew is performing EPP-009, Post LOCA Cooldown and Depressurization, due to a small break LOCA
- The crew has initiated a cooldown to Cold Shutdown
- The first CSIP has been secured and the crew is evaluating RCS Pressure response

The following is observed:

Time	RCS Temperature	RCS Pressure
1400	435.0°F	462 psig
1402	433.5°F	458 psig
1404	432.0°F	454 psig
1406	430.5°F	450 psig

Which ONE of the following describes RCS Pressure for this event in accordance with the EOP User's Guide?

A. STABLE because it has been demonstrated that it can be controlled

B.✓ STABLE because RCS subcooling is increasing

C. LOWERING even though RCS subcooling is increasing

D. LOWERING because it cannot be demonstrated that it can be controlled

Plausibility and Answer Analysis

EOP-Users Guide section 6.5

A *Incorrect. RCS Pressure should be considered STABLE because RCS subcooling is increasing with an operator controlled cooldown in progress. No attempt to control RCS Pressure is made. Just because a controlled cooldown is in progress doesn't mean RCS Pressure is controlled.*

B *Correct. RCS Pressure should be considered STABLE because RCS subcooling is increasing with an operator controlled cooldown in progress.*

C *Incorrect. RCS Pressure is lowering but with an operator controlled cooldown in progress and subcooling increasing RCS Pressure should be considered stable.*

D *Incorrect. RCS Pressure is lowering but with an operator controlled cooldown in progress and subcooling increasing RCS Pressure should be considered stable. No attempt has been made to control RCS Pressure. Just because RCS pressure is lowering during a controlled cooldown doesn't mean it is uncontrolled.*

QUESTIONS REPORT

for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Emergency Procedures/Plans - Knowledge of EOP terms and definitions.

Importance Rating:	3.9 4.3
Technical Reference:	EOP-Users Guide Rev. 26, Page 35
References to be provided:	None
Learning Objective:	LP-EOP-3.19, Obj. 4d
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	USERS GUIDE
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.4.17	K/A 2:	

6.3 Resetting SI, Phase A, Phase B, and FW Isolation

In most events SI, Phase A, Phase B, and FW Isolation signals are reset just prior to the steps that manipulate components impacted by these signals. There may be cases; however, when the EOPs direct manipulation of components and an isolation signal affecting the component has not been directed to be reset. In this case it is permissible to reset the associated signal; however, the impact of this action should be evaluated. For example, reset of SI and Phase A may restore the torque switch overload protection for limitorque valves that reposition on SI or are required to reposition to establish cold leg recirculation. In this case the status of the limitorque valves should be evaluated prior to resetting the signals. Note that some RAB ventilation will realign when SI is reset and the ventilation portion of SI verification attachment of OMM-004 should be completed or verified using the plant computer prior to resetting SI.

6.4 Sequencer Operation

Any time the sequencer is actuated by an SI or loss of off-site power signal, the operator should monitor the Emergency Safeguards Sequencer for proper equipment loading. Monitoring the sequencer ensures immediate attention is given to major equipment including that which is not verified in the EOPs (e.g., E-6 fans, WC-2 chiller), but would be checked by OMM-004. The operator should allow the sequencer to complete its cycle prior to attempting to start any large electrical loads including loads the sequencer may have failed to start. If any safeguards component has failed to auto start, a manual start attempt should be made when the component status is checked during implementation of the PATH-1 GUIDE, "Safeguards Verification" attachment. This includes the case where the sequencer itself has failed. This restriction prevents overloading a diesel generator and/or its associated emergency bus. It also provides an orderly methodology to address and correct failures of safeguards equipment. (Reference 2.2.3.15)

6.5 Stable Pressure/Temperature

The operator is frequently asked to check RCS and SG pressures and temperature as STABLE (or INCREASING). STABLE does not necessarily imply constant. RCS and/or SG pressure may be decreasing slowly due to an operator-controlled cooldown and still be considered stable. If the operator can control the rate and magnitude of the pressure change, then pressure should be considered stable. The same conventions apply to temperature. Note that when evaluating RCS pressure response for reducing or terminating SI with an RCS cooldown in progress, RCS pressure may be considered STABLE if subcooling is increasing. (References 2.2.2.11, 2.2.2.19 and 2.2.3.25)

SG pressure may also trend downward during a LOCA if RCS temperature (as measured by core exit thermocouples or T-hot) drops below that of the water in the secondary side of the SGs. This response is caused by the cooling of the secondary water by the circulating RCS fluid.

74. 2009A NRC RO 074

Given the following plant conditions:

- The plant is operating at 100% power
- A loss of MCB annunciators has occurred and the crew has entered AOP-037, Loss of Main Control Room Annunciators
- The USCO has determined that the following ALBs are lost
  - ALB-001, Containment Spray & Accumulator System
  - ALB-002, Emergency Service Normal Service Water System
  - ALB-003, Miscellaneous Systems
  - ALB-004, RHR/RWST System

Which ONE of the following describes the Tech Spec implications and required action for loss of these ALBs?

A.✓ The CNMT Sump Level and Flow Monitoring Alarm is Inoperable;
Begin logging CNMT Sump Level and determining Flow manually within 10 minutes

B. The CNMT Sump Level and Flow Monitoring Alarm is Inoperable;
Begin logging CNMT Sump Level and determining Flow manually within 1 hour

C. The Main and Aux Reservoir Level and Temperature Alarms are Inoperable;
Begin logging Temperature and Level for both Reservoirs within 10 minutes

D. The Main and Aux Reservoir Level and Temperature Alarms are Inoperable;
Begin logging Temperature and Level for both Reservoirs within 1 hour

Plausibility and Answer Analysis

A Correct. ALB-01 Window 6-1 meets the requirement for T.S. 3.4.6.1.b. With this alarm lost, AOP-16 attachment 16 provides for manual determination and the first sump level must be recorded within 10 minutes of the original loss.

B Incorrect. ALB-01 Window 6-1 meets the requirement for T.S. 3.4.6.1.b. With this alarm lost, AOP-16 attachment 16 provides for manual determination but the first sump level must be recorded within 10 minutes of the original loss. The 1 hour time requirement is incorrect but many Tech Specs have 1 hour actions and ROs are responsible for actions of 1 hour or less.

C Incorrect. The Main and Aux Reservoir alarms have been lost on ALB-02 Window 7-5 but this surveillance is met by the normal logging in OST-1021 every 12 hours. No additional actions are required.

D Incorrect. The Main and Aux Reservoir alarms have been lost on ALB-02 Window 7-5 but this surveillance is met by the normal logging in OST-1021 every 12 hours. No additional actions are required.

KA Statement - Emergency Procedures/Plans - Knowledge of operator response to loss of all annunciators.

Importance Rating:	3.6 4.0
Technical Reference:	T.S. 3.4.6.1.b APP-ALB-001 Rev. 18 Window 6-1 APP-ALB-002 Rev. 41, Window 7-5 AOP-016 Re. 39, Attachment 16, pages 101-107 AOP-037 Rev. 12
References to be provided:	None
Learning Objective:	LP-AOP-3.37, Obj. 2
Question Origin:	NEW
Comments:	The RO's are not responsible for classifying events in the E Plan (PEP-110). This question is written toward the AOP (Emergency Procedure) that would be implemented along with the Tech Specs that are implemented during loss of annunciators.

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-016
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.4.32	K/A 2:	

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Airborne Gaseous Radioactivity Monitoring System,
b. The Reactor Cavity Sump Level and Flow Monitoring System, and
c. The Containment Airborne Particulate Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1. Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2. Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3. Perform a Reactor Coolant System water inventory balance at least once per 8 hours*.

   Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

*Not required to be performed until 12 hours after establishment of steady-state operation. Not applicable to primary-to-secondary leakage.

6-1

DEVICES: ERFIS Points	**SETPOINT:**	**CONTAINMENT UNIDENTIFIED LEAKAGE OR TROUBLE**
Leakage: URE9001 or URE9002	>0.76 gpm above leak rate baseline	
Trouble: URE9003 or URE9004	>0.76 gpm difference from previous hour leak rate	
REFLASH:	**YES**	

OPERATOR ACTIONS:

1. **CONFIRM** alarm using:
   - LIT-7161A, Cont Reactor Cav Sump Hi Lvl Alarm (Narrow Range)
   - LIT-7161B, Cont Reactor Cav Sump Hi Lvl Alarm (Narrow Range)

2. **VERIFY** Automatic Functions: None

3. **PERFORM** Corrective Actions:
   a. **IF** the alarm is inoperable,
      THEN PERFORM AOP-016, Excessive Primary Plant Leakage, Attachment 16
      AND CONTINUE with this procedure.
   b. **DETERMINE** if the alarm is a result of increased LEAKAGE or TROUBLE with the program as follows:
      (1) **CHECK** the computer points which input to this alarm are functional using OP-163, Plant Computer.
      (2) **DETERMINE** if there has been an increase in containment sump pump-downs.
      (3) **REFER TO** Tech Spec 3.4.6.1.
   c. **IF** leakage to the sump is indicated,
      THEN GO TO AOP-016, Excessive Primary Plant Leakage, Attachment 16.
   d. **IF** excessive leakage is evident by alarm
      AND increased frequency of containment sump pump-downs,
      THEN MAKE a containment entry when permissible to locate, identify, and isolate the leak.
   e. **IF** an evolution known to cause sump inleakage is in progress,
      AND there are NO other indications of RCS leakage,
      THEN:
      (1) **MONITOR** the evolution **AND STOP** if necessary.
      (2) **PERFORM** OST-1026.
   f. **IF** maintenance is to be performed,
      THEN REFER TO OWP-ERFIS, ERFIS Computer.

CAUSES:

1. Excessive leakage into the containment sump
2. Computer malfunction
3. Alarm circuit or instrumentation malfunction

REFERENCES:

1. Tech Specs 3.4.6.1 and 3.4.6.2
2. AOP-016, Excessive Primary Plant Leakage
3. OP-163, Plant Computer
4. OST-1026, Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4
5. OWP-ERFIS, ERFIS Computer
6. 2166-B-401 1047

DEVICES: See below **SETPOINT**: See below

COMPUTER ALARM SERVICE WATER

REFLASH: (SEE NOTE BELOW)

Computer Point	Warning	Alarm
FSW9205A(B), ESW Flow from RAB A(B) Chiller WC-2 (Power Operation)	2750 gpm	2800 gpm
TSW9160A(B), ESW PUMP A(B) Mtr Winding Temp	225°F	240°F
TSW9114, Aux Reservoir Temperature (Power Operation)	90°F	91°F
TSW9115, Main Reservoir Temperature (Power Operation)	90°F	91°F
LSC8752A(B), Auxiliary Reservoir Level	251 ft	250.5 ft
LSC8750A(B), Main Reservoir Level	216 ft	215.5 ft
TSW9150A(B), NSW Pump A(B) Mtr Winding Temp	240°F	266°F
TSW9151A(B), NSW Pump A(B) Upper Brg Temp	175°F	185°F
TSW9152A(B), NSW Pump A(B) Mtr Low Brg Temp	175°F	185°F
TSW9153A(B), NSW Pump A(B) Thrust Brg Temp	175°F	185°F
TSW9301, NSW Header Temp	95°F	102°F
LSC8755A, ESW Mn Rsvr A-SA Scrn Diff Lvl	N/A	HI-HI
LSC8755B, ESW Mn Rsvr B-SB Scrn Diff Lvl	N/A	HI-HI
LSC8756A, ESW Aux Rsvr A-SA Scrn Diff Lvl	N/A	HI-HI
LSC8756B, ESW Aux Rsvr B-SB Scrn Diff Lvl	N/A	HI-HI

OPERATOR ACTIONS:

NOTE

ERFIS alarms will not re-flash on the Annunciator Panel when elevating from a Warning to Alarm on the same point. The only indication that the alarm has changed state will be a color change from yellow to red on the alarm screen. New alarm points that come in subsequently will initiate a single re-flash of the Annunciator window and will follow the same process.

1. **CONFIRM** alarm using:
   a. Associated ERFIS alarm screen or Group Display ALB 02
2. **VERIFY** Automatic Functions: None
3. **PERFORM** Corrective Actions:
   a. **CHECK** instrumentation on MCB associated with alarming point.
   b. **IF** the alarm is for Reservoir temperature **OR** low level,
      THEN DISPATCH an Operator to check local indications associated with alarming point(s).
   c. **IF** the alarm is for Main Reservoir low level,
      THEN:
      (1) **IF** ESW pump(s) are aligned to the Main Reservoir,
          THEN REALIGN the affected ESW Pump(s) suction to the Auxiliary Reservoir.
      (2) **VERIFY** spillway valves and sluice gates are shut per OP-141, Cooling Tower and Reservoir Complex.
      (3) **IF** Main Reservoir level decreases below 215',
          THEN:
          (a) **DECLARE** the Main Reservoir inoperable.
          (b) **REFER TO** Tech Spec 3.7.5 (Modes 1-4).
          (c) **GO TO** AOP-022, Loss of Service Water.

(Continued on Next Page)

Attachment 16

Sheet 1 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED

CAUTION

- Calculation and evaluation of manual sump log leak rate shall be completed at a frequency such that any unidentified leakage can be detected in less than one hour.
- The first determination of sump level must be obtained **within 10 minutes** of alarm inoperability.

☐ 1. **INITIATE** Attachment 17, Containment Sump Inleakage Log, AND **RECORD** the time of inoperability in comments section.

2. **DETERMINE** an initial sump leak rate limit as follows:

NOTE

These values may require use of AutoLog archive files.

☐ a. **IF** ERFIS points URE9001 and URE9002, Sump Leak Rate, are functional per OP-163, **THEN OBSERVE** values AND **USE** the lowest to perform the calculation below.

RESPONSE NOT OBTAINED: ☐ a. **OBSERVE** CO Log for the value of leak rate recorded for ERFIS points URE9001 and URE9002 after the last completion of OST-1026 (OST-1226), AND **USE** the lowest value to perform the calculation below.

Leak Rate obtained from above (gpm) + 0.76 gpm = _________ Leak Rate Limit (gpm)

☐ b. **RECORD** the Leak Rate Limit on Attachment 17 in the 'Leak Rate Limit' column.

Attachment 16
Sheet 2 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- Manual determination per Attachment 17, Containment Sump Inleakage Log, may be performed if at least one level channel is properly functioning.
- Sump pumpdowns should be anticipated AND starting and stopping levels **are required**. (Approximate Start: 2.91 ft - Stop: 0.82 ft)

3. **OBTAIN** an initial sump level as follows:

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ a. **IF** ERFIS point LCT7161A and/or LCT7161B, Key Way Sump Level, channel check satisfactorily, **THEN RECORD** present values on Attachment 17 in 'Level' column (Level in feet).	☐ a. **DETERMINE** sump level on LIT-7161A and/or LIT-7161B, Cont Reactor Cav Sump Hi Lvl Alarm (Narrow Range), AND **RECORD** present values on Attachment 17 in 'Level' column (Level in inches).
☐ b. **RECORD** the time this level was obtained in the Attachment 17 'Time' column.	

NOTE

The second determination of sump level must be obtained **within 30 minutes** of alarm inoperability.

4. **OBTAIN** AND **RECORD** the second reading of sump level as follows:

☐ a. **PERFORM** Step 3.

b. **RECORD** the following for ANY sump pumpdown:

☐ • maximum sump level (sump pumpdown starts)

_____________ max level

☐ • minimum sump level (sump pumpdown stops):

_____________ min level

Attachment 16

Sheet 3 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED
5. **IF** previous sump leak rate is less than 6 gpm, **THEN CALCULATE** a present leak rate using the applicable following equation:	☐ **5.** **GO TO** Step 6.
☐ **a.** **CHECK** NO sump pumpdown has occurred.	☐ **a.** **GO TO** Step 5.d.
☐ **b.** **IF** using ERFIS points LCT7161A and/or LCT7161B, **THEN CALCULATE** the leak rate as follows:	☐ **b.** **IF** using local level indication on LIT-7161A and/or LIT-7161B, **THEN CALCULATE** the leak rate as follows:
$\frac{[\Delta \text{ level (ft)}]\ [157 \text{ (gal/ft)}]}{\Delta \text{ Time (min)}} = \_\_\_\_\_\_\_\_ \text{(gpm)}$	$\frac{[\Delta \text{ level (in)}]\ [13.083 \text{ (gal/in)}]}{\Delta \text{ Time (min)}} = \_\_\_\_\_\_\_\_ \text{(gpm)}$
☐ **c.** **GO TO** Step 7.	

NOTE

The Δ level used in the following calculation should be the sum of the level change before and after the sump pumpdown. Max and min levels are recorded in Step 4.b.

☐ **d.** **CALCULATE** Δ level as follows:

Δ level = (max level - initial level) + (final level - min level)

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ **e.** **IF** using ERFIS points LCT7161A and/or LCT7161B, **AND** sump pumpdown has occurred, but a leak rate was NOT calculated using levels immediately prior to pumpdown, **THEN CALCULATE** the leak rate as follows:	☐ **e.** **IF** using local level indication on LIT-7161A and/or LIT-7161B, **AND** sump pumpdown has occurred, but a leak rate was NOT calculated using levels immediately prior to pumpdown, **THEN CALCULATE** the leak rate as follows:
$\frac{[\Delta \text{ level (ft)}]\ [157 \text{ (gal/ft)}] + [\text{pump run time (min)}]\ [6 \text{ gpm}]}{\Delta \text{ Time (min)}}$	$\frac{[\Delta \text{ level (in)}]\ [13.083 \text{ (gal/in)}] + [\text{pump run time (min)}][6 \text{ gpm}]}{\Delta \text{ Time (min)}}$
$= \_\_\_\_\_\_\_ \text{(gpm)}$	$= \_\_\_\_\_\_\_ \text{(gpm)}$
☐ **f.** **GO TO** Step 7.	

Attachment 16
Sheet 4 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED

6. **IF** previous sump leak rate is between 6 and 13.5 gpm, **THEN**:

   a. **OBTAIN** time of sump pump starts and stops by performing one of the following:

   ☐ (1) **DIRECT** Radwaste Control Room to record time of sump pump starts and stops.

   ☐ (2) **OBSERVE** ERFIS time for minimum and maximum sump levels.

NOTE

- A sump pump down is assumed to remove 344 gallons from the sump.
- Pump Stop = time pumpdown stops
- Pump Start = time pump starts for the following pumpdown

☐ b. **CALCULATE** a Leak Rate using the following equation:

$$\frac{\text{344 gallons}}{\Delta\ \text{Time [min] (Pump Start - Pump Stop)}} = \underline{\hspace{3cm}}\ \text{(gpm)}$$

NOTE

The following step will allow calculating sump leak rate per the above step so the less than 60 minute time requirement can be met prior to exceeding 6.0 gpm.

☐ 7. **IF** sump leak rate is between 5.80 and 6.0 gpm, **THEN CALCULATE** a leak rate by performing BOTH Steps 5 and 6 above.

Attachment 16

Sheet 5 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ 8. **RECORD** the highest calculated leak rate for each channel on Attachment 17 in the Leak Rate column.	

NOTE

- When recording Δ Leak Rate, include + and - signs.
- The determination of each successive leakrate must be completed **before 60 minutes** has elapsed such that Regulatory Guide 1.45 requirement of detecting leakrate "in less than one hour" can be met.

☐ 9. **CALCULATE** the Δ Leak Rate for each channel before an hour (60 min) has elapsed using the following equation AND **RECORD** on Attachment 17 in the 'Δ Leak Rate' column:

Present Leak Rate - Previous Leak Rate = ____________________ *Δ Leak Rate*

☐ 10. **PERFORM** Independent Verification of all calculations and initial on Attachment 17 in the 'Verify Init" column.

INSTRUCTIONS	RESPONSE NOT OBTAINED
11. **CHECK** BOTH of the following conditions exist: ☐ • BOTH channels of sump Leak Rate rises less than the sump Leak Rate Limit ☐ • Δ Leak Rate changes less than ± 0.76 gpm	11. **PERFORM** the following: ☐ a. **INVESTIGATE** the cause. ☐ b. **PERFORM** OST-1026 OR OST-1226 to determine unidentified leakage rate. ☐ c. **LOG** occurrence, actions taken and results in the CO Log.

Attachment 16
Sheet 6 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ **12. RECALCULATE** a new sump Leak Rate Limit each time OST-1026 (OST-1226) leak rate calculation is performed and note in comments section.	
☐ **13. RECORD** readings as follows: • More frequently than hourly • Immediately prior to and after sump pumpdowns	
14. RECORD the following events at a minimum on Attachment 17 Comments Section: ☐ • Each completed OST-1026 (OST-1226) ☐ • Any sump pump downs ☐ • Any change in Leak Rate Limit and the basis ☐ • Any Leak Rate and/or Leak Rate Limit exceeded	
☐ **15. CHECK** keyway inleakage less than 13.5 gpm.	**15. PERFORM** the following: ☐ a. **APPLY** Action b of Tech Spec 3.4.6.1. ☐ b. **PERFORM** a leakrate calculation (refer to OST-1026 or OST-1226).
☐ **16. OBTAIN** a Unit USCO review of Attachment 17 for each shift and have the Unit USCO initial on Attachment 17.	

Attachment 16
Sheet 7 of 7

Containment Unidentified Leakage Alarm Inoperability

INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Manual determination may be stopped when the Sump Leak Rate program is declared operable.

INSTRUCTIONS	RESPONSE NOT OBTAINED
☐ **17. CHECK** leak rate program inoperability ends prior to 2400 (midnight).	**17. PERFORM** the following: ☐ a. **START** a new Attachment 17 for the new day. ☐ b. **FORWARD** the previous Attachment 17 for review.
☐ **18. EXIT** this attachment.	

-- END OF ATTACHMENT 16 --

75. 2009A NRC RO 075

Given the following plant conditions:
- The plant is operating in Mode 5
- The RCS is in solid plant operation
- Both Trains of RHR are aligned in the Shutdown Cooling Mode
- A large RCS leak has developed
- The crew has aligned flow through the BIT with 'A' CSIP in service as directed by AOP-020, Loss Of RCS Inventory Or Residual Heat Removal While Shutdown
- Core Exit Thermocouples continue to rise
- RCS water level continues to lower

Which ONE of the following is the action required by AOP-020 to mitigate the event?

A. Start the 'B' CSIP with flow through 1SI-3 and 1SI-4, BIT Outlet Valves

B. Start the 'B' CSIP with flow through 1SI-52, Alternate High Head SI to Cold Leg Valve

C.✓ Align 'A' RHR Pump for Low Head SI with flow through 1SI-340, Low Head SI Train A to Cold Leg Valve

D. Align 'A' RHR Pump for Low Head SI with flow through 1SI-359, Low Head SI Train to Hot Leg Valve

Plausibility and Answer Analysis

A *Incorrect. Starting the second CSIP with flow through 1SI-3 and 1SI-4, BIT Outlet Valves would provide additional flow but only one CSIP is Operable in this mode.*

B *Incorrect. Start the second CSIP with flow through 1SI-52, Alternate High Head SI to Cold Leg Valve would provide additional flow and this alignment is directed in EPP-010 with two CSIPs but only one CSIP is Operable in this mode.*

C *Correct. Align one train of RHR for Low Head SI with flow through 1SI-340, Low Head SI Train A to Cold Leg Valve is directed in AOP-020.*

D *Incorrect. Align one train of RHR for Low Head SI is directed in AOP-020 but flow is through 1SI-340, Low Head SI Train A to Cold Leg Valve not 1SI-359. This alignment is used in EPP-011.*

QUESTIONS REPORT
for 2009A NRC RO ONLY QUESTIONS REV1

KA Statement - Emergency Procedures/Plans - Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies

Importance Rating:	3.8 4.2
Technical Reference:	AOP-020 Rev. 31 Section 3.6, pages 61-64
References to be provided:	None
Learning Objective:	LP-AOP-3.20, Obj. 2
Question Origin:	NEW
Comments:	

Origin:	NEW	Cog Level:	H
Difficulty:	3	Reference:	AOP-020
Ref. Provided?:	N	Key Words:	2009A NRC RO
K/A 1:	G2.4.9	K/A 2:	

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.6 Establishing SI Following a Major RCS Leak	
☐ **1. VERIFY** one CSIP RUNNING.	☐ **1. GO TO** Step 4.
2. ALIGN CSIPs to the BIT:	
a. VERIFY OPEN Suctions From RWST: ☐ • 1CS-291 (LCV-115B), CSIP Suction From RWST ☐ • 1CS-292 (LCV-115D), CSIP Suction From RWST	
b. VERIFY SHUT VCT Outlet valves: ☐ • 1CS-165 (LCV-115C), VCT Outlet ☐ • 1CS-166 (LCV-115E), VCT Outlet	

NOTE

Low VCT level is a precursor to gas binding the CSIPs. **[C.3]**

INSTRUCTIONS	RESPONSE NOT OBTAINED
✱☐ **c. CHECK** VCT level MAINTAINED ON SCALE.	**c. Locally TORQUE SHUT** the following valves: **[C.3]** ☐ • 1CS-165 (LCV-115C), VCT Outlet ☐ • 1CS-166 (LCV-115E), VCT Outlet
d. SHUT Charging Line Isolation valves: ☐ • 1CS-235, Charging Line Isolation ☐ • 1CS-238, Charging Line Isolation	

(Continued on Next Page)

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.6 Establishing SI Following a Major RCS Leak

2. (continued)

e. **OPEN** Boron Injection Tank Outlet valves:

- ☐ 1SI-3, BIT Outlet Isolation
- ☐ 1SI-4, BIT Outlet Isolation

f. **SHUT** Charging/SI Pumps Normal Miniflow Isolation valves:

- ☐ 1CS-182, Charging SI Pump A-SA Miniflow Isolation
- ☐ 1CS-196, Charging SI Pump B-SB Miniflow Isolation
- ☐ 1CS-210, Charging SI Pump C-SAB Miniflow Isolation
- ☐ 1CS-214, Charging SI Pumps Miniflow Isolation

☐**3. CHECK** for ANY of the following:

- RCS level dropping
- Core exit thermocouple temperature rising

RESPONSE NOT OBTAINED:

☐**3. GO TO** Step 5.

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.6 Establishing SI Following a Major RCS Leak

CAUTION

If liquid in the RHR suction line is sufficiently hot, depressurizing the line by opening RWST suctions in the next step may cause some flashing to steam and/or water hammer to occur.

4. **ALIGN** Low Head SI:

   a. **OPEN** the following:

   ☐ • 1SI-322, RWST to RHR Pump A-SA

   ☐ • 1SI-323, RWST to RHR Pump B-SB

☐ b. **START** one RHR Pump.

   c. **OPEN** the discharge associated with the running RHR Pump:

   A RHR Pump

   ☐ • 1SI-340, Low Head SI Train A to Cold Leg

   B RHR Pump

   ☐ • 1SI-341, Low Head SI Train B to Cold Leg

☐ d. **CHECK** for ANY of the following:

   • RCS level dropping

   • Core exit thermocouple temperature rising

   RESPONSE NOT OBTAINED: ☐ d. **GO TO** Step 5.

☐ e. **START** the remaining RHR Pump.

   RESPONSE NOT OBTAINED: ☐ e. **GO TO** Step 5.

(Continued on Next Page)

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.6 Establishing SI Following a Major RCS Leak	
4. (continued)	
f. **OPEN** the discharge valve for the RHR Pump just started:	
A RHR Pump	
☐ • 1SI-340, Low Head SI Train A to Cold Leg	
B RHR Pump	
☐ • 1SI-341, Low Head SI Train B to Cold Leg	
5. **MONITOR** RWST level:	
✱☐ a. GREATER THAN 23.4%	a. **WHEN** CNMT Wide Range Sump level is GREATER THAN 142 in., **THEN:**
	☐ (1) **REFER TO** EPP-010, Transfer to Cold Leg Recirculation.
	☐ (2) **TRANSFER** to cold leg recirculation.
	☐ (3) **CONSULT** with Operations Management for further actions.
	☐ (4) **EXIT** this procedure.
✱☐ b. GREATER THAN 3%	☐ b. **STOP** ALL CSIP and RHR Pumps taking suction solely from RWST.