Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 4	Group	1	
	K/A	007 EK1.02	
Level of Difficulty: 2	Importance Rating	3.4	

Reactor Trip-Stabilization-Recovery: Knowledge of the operational implications of the following concepts as they apply to the Reactor Trip: Shutdown margin

Proposed Question: 1

Why are Control Rod insertion limits established for power operation?

A. Minimizes the worth of a postulated dropped Control Rod.

B. Maintains a negative Moderator Temperature Coefficient.

C. Provides adequate shutdown margin after a Reactor Trip.

D. Ensures sufficient positive reactivity to offset the Power Defect.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that if RILs are high enough the effect of a dropped rod would be minimized by the remaining rods being above the RIL.

B. Incorrect. Plausible because a positive moderator temperature coefficient will reduce shutdown margin due to the positive reactivity added during a heatup. So it could be thought that RILs provide a mechanism for ensuring a negative MTC is maintained.

C. Correct. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

D. Incorrect. Plausible because it could be thought that RILs ensure power defect is offset, however rods are essentially fully withdrawn and boron concentration is reduced to offset the power defect.

Technical Reference(s):
- Technical Specification LCO 3.1.1 Bases
- Technical Specification LCO 3.1.6 Bases
- LO21.GFR.PHY, Page 4

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** reactor response to a control rod insertion.
DESCRIBE the basic design of the Rod Insertion Limit (RIL) Monitor System.

Question Source:	Bank	ILOT1413	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	1	
	55.43		

Comments / Reference: Technical Specification LCO 3.1.1 Bases	Revision: 68

SDM
B 3.1.1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are satisfied by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control System, together with the Chemical and Volume Control System (CVCS), provides the SDM during power operation and is capable of making the core subcritical, assuming that the rod of highest reactivity worth remains fully withdrawn. The CVCS can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Comments / Reference: Technical Specification LCO 3.1.6 Bases	Revision: 68

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and five shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

Comments / Reference: LO21.GFR.PHY, Page 4	Revision: 12/17/07

will occur if he completes withdrawal of the shutdown rods.

Once the shutdown rods have been fully withdrawn, the heatup of the RCS can commence. The reactor coolant pumps, pressurizer heaters, and reactor decay heat are the sources of heat. Heatup procedures typically limit the heatup rate of the RCS to 50°F per hour.

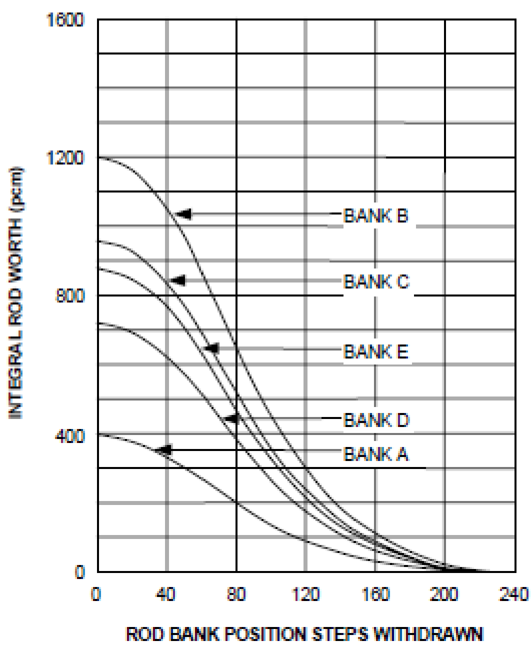


Figure 8-1 Integral Rod Worth vs. Steps Withdrawn

Under certain conditions, the possibility exists that the moderator temperature coefficient is positive. This may cause the plant to lose some of its required shutdown margin due to the addition of positive reactivity during the heatup. Plant technical specifications require that the shutdown margin be maintained greater than 1.3% Δk/k. It is the operator's responsibility to ensure that the minimum shutdown margin requirements of the plant technical specifications are met at all times.

Once the RCS temperature has been increased to normal operating temperature (NOT) and the pressure increased to normal operating pressure (NOP), the reactor startup may begin.

Before withdrawing the control rods, the source range count rate is 150 cps. If the reactivity addition produces a subcritical multiplication factor of 2.5, what is the final steady-state count rate following this evolution?

Example 8-2

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 5	Group	1	
	K/A	008 AA1.08	
Level of Difficulty: 3	Importance Rating	3.8	

Pressurizer Vapor Space Accident: Ability to operate and/or monitor the following as they apply to the Pressurizer Vapor Space Accident: PRT level, pressure, and temperature

Proposed Question: 2

Given the following conditions:

- Unit 1 is in MODE 1.
- 1-8010B, Pressurizer Safety Relief Valve B, is partially open.
- The following parameters are observed:
  - Reactor power is 99.8%.
  - Pressurizer pressure is 2185 psig and slowly lowering.
  - Pressure Relief Tank (PRT) pressure is 1 psig and slowly rising.

Which of the following lists the approximate temperature and phase of the fluid flowing to the PRT from the open relief valve?

A. 102°F, saturated

B. 102°F, superheated

C. 216°F, saturated

D. 216°F, superheated

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 102°F is the saturation temperature for the downstream pressure if not converted to psia. Fluid phase is correct.

B. Incorrect. Plausible because 102°F is the saturation temperature for the downstream pressure if not converted to psia. Additionally, if the downstream pressure was not converted to psia, fluid phase would appear to be superheated.

C. Correct. The steam tables are used to determine the enthalpy of the downstream fluid. Enthalpy is determined at absolute pressure. Since the ideal throttling process in isenthalpic, enthalpy does not change across the relief valve. Enthalpy at 2200 psia (2185 psig + ~15 psi) = ~1122 Btu/lbm. With downstream pressure at 16 psia, the enthalpy of saturated vapor is approximately 1152 Btu/lbm. Fluid at this pressure that is below this value is saturated. Therefore the fluid on the downstream side of the relief valve is a saturated, or wet, vapor. Temperature of the fluid is equal to the saturation temperature of the fluid pressure (~16 psia), which is approximately 216°F.

D. Incorrect. Plausible because fluid temperature is correct. However, if the downstream pressure was not converted to psia, fluid phase would appear to be superheated.

Technical Reference(s) Steam Tables Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: **SOLVE** throttling process problems, applying the General Energy Equation.

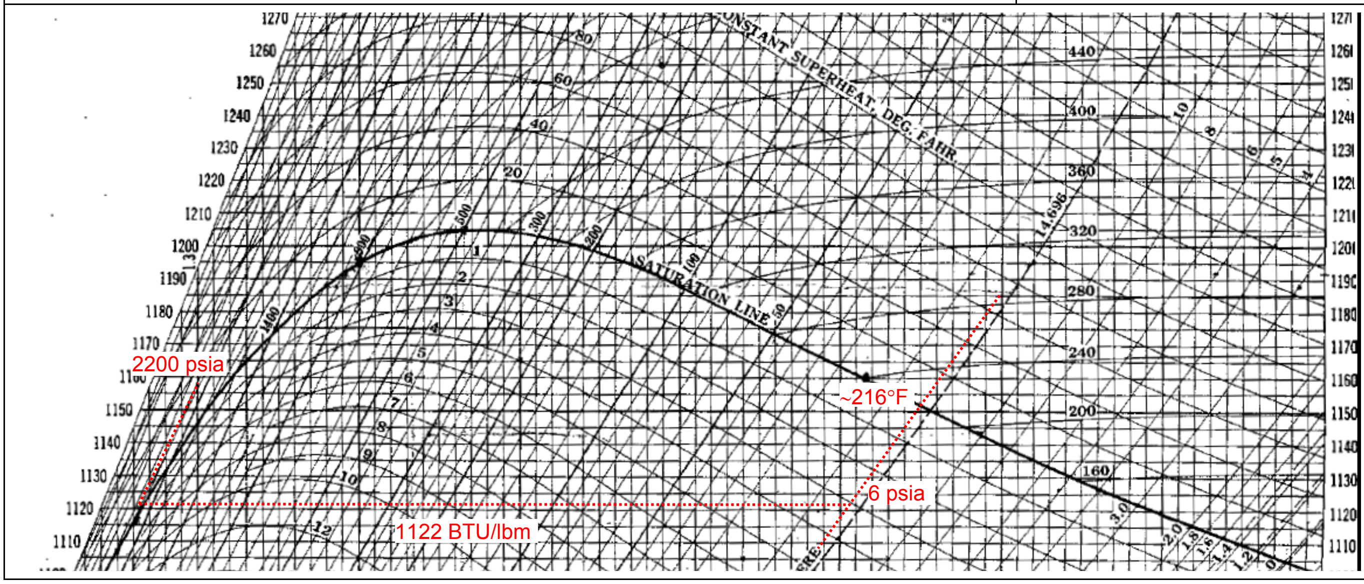
Question Source: Bank ______
Modified Bank ILOT2314 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 14
55.43 ______

Comments / Reference: From Mollier Diagram	Revision: N/A



Original Question: CPNPP Exam Bank ILOT2314

What is the approximate temperature and phase of the fluid downstream of the pressurizer relief valve if it sticks partially open with 2,200 psia in the pressurizer and a 50 psia backpressure?

A. 281ºF, saturated

B. 281ºF, superheated

C. 332ºF, saturated

D. 332ºF, superheated

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 5	Group	1	
	K/A	009 EA2.14	
Level of Difficulty: 2	Importance Rating	3.8	

Small Break LOCA: Ability to determine and interpret the following as they apply to the small break LOCA: Actions to be taken if PTS limits are violated

Proposed Question: 3

Given the following conditions:

- Unit 1 is responding to a Small Break Loss of Coolant Accident.
- A RED path on INTEGRITY is being addressed in accordance with FRP-0.1A, Response to Imminent Pressurized Thermal Shock Condition.
- Reactor Coolant System (RCS) temperature is stable.
- RCS pressure is stable with only the Control Group C Pressurizer Heaters energized.
- A Reactor Coolant System temperature soak has just been initiated.

Which of the following evolutions may be performed during the soak required by FRP-0.1A?

A. Energize additional Pressurizer Heaters.

B. Place Auxiliary Pressurizer Spray in service.

C. Raise Auxiliary Feedwater flow to 300 gpm to each Steam Generator to raise levels to 50%.

D. Commence cooldown after recirculating Residual Heat Removal to equalize boron concentration.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because energizing additional heaters would raise temperature which does not violate soak restrictions, however, energizing heaters also causes pressure to increase which does violate soak restrictions.
B. Correct. Any action that will not cause either an RCS cooldown or RCS pressure increase is permitted during the soak.
C. Incorrect. Plausible because raising steam generator level to between 50% and 60% is procedurally directed, however at 1200 gpm total SG flow a cooldown would be unavoidable.
D. Incorrect. Plausible because recirculation of RHR in preparation for placing in service would be acceptable as cooldown at less than 50°F per hour would follow the soak, however commencing the cooldown before the soak is complete is not acceptable.

Technical Reference(s) FRP-0.1A, Step 28 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRP-0.1, Response to Imminent Pressurized Thermal Shock Condition.

Question Source: Bank ILOT0947
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: FRP-0.1A, Step 28	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRP-0.1A
RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	REVISION NO. 8	PAGE 18 OF 53

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*28	Determine If RCS Temperature Soak Is Required:	
	a. Cooldown rate in RCS cold legs - GREATER THAN 100°F IN ANY 60 MINUTE PERIOD.	a. Go to Step 29.
	b. Perform all of the following:	
	1) Do not cool down RCS until temperature has been stable for 1 hour.	
	2) Do not increase RCS pressure during soak period.	
	3) Perform actions of other procedures in effect which do not cool down or increase RCS pressure until the RCS temperature soak has been completed.	
	4) RCS cool down is permitted after 1 hour.	
	5) Maintain RCS pressure and cold leg temperatures within the limits of Attachment 3.	
	6) Maintain cooldown rate in RCS cold legs less than 50°F in any 60 minute period.	
29	Return To Procedure And Step In Effect.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 4	Group	1	
	K/A	011 EA2.03	
Level of Difficulty: 3	Importance Rating	3.7	

Large Break LOCA: Ability to determine and interpret the following as they apply to the large break LOCA: Consequences of managing LOCA with loss of CCW

Proposed Question: 4

Given the following conditions:

- A Large Break Loss of Coolant Accident has occurred inside Unit 1 Containment.
- Reactor Coolant System and Containment pressure are both approximately 30 psig.
- Component Cooling Water (CCW) Pump 1-01 has tripped.
- Unit 1 Refueling Water Storage Tank level is 47% and lowering.
- CCW trains are split and CANNOT be cross-tied.

With the current status of CCW, the Residual Heat Removal (RHR) system can be operated in what Modes of Emergency Core Cooling?

A. Train A RHR can be operated in the Injection Mode ONLY.
Train B RHR can be operated in BOTH the Injection and Recirculation Modes.

B. Train A RHR can be operated in the Injection Mode ONLY.
Train B RHR can be operated in the Injection Mode and can also be operated in the Recirculation Mode ONLY if the RHR trains are cross-tied.

C. Train A RHR can be operated in the Injection and Recirculation Modes if the RHR trains are cross-tied.
Train B RHR can be operated in BOTH the Injection and Recirculation Modes.

D. Train A RHR can be operated in the Injection and Recirculation Modes if the RHR trains are cross-tied.
Train B RHR can be operated in the Injection and Recirculation Modes if the RHR trains are cross-tied.

Proposed Answer: A

Explanation:

A. Correct. When an RHR train is operating in Injection Mode, it pumps water from the Refueling Water Storage Tank (RWST). The temperature of the water from the RWST is < 120°F, which is the limit for using a RHR pump (see EOS-1.3A, Transfer to Cold Leg Recirculation, Attachment 3, Bases for Step 2). Therefore, Train A RHR may only operate in the Injection Mode. Train B may be operated in any mode since CCW is still available to the Train B RHR system.

B. Incorrect. Plausible because Train A RHR can only be operated in the Injection Mode, but Train B can be operated in injection or recirculation mode since it has CCW available.

C. Incorrect. Plausible because Train B RHR can be operated in the Injection or Recirculation Mode since it has CCW available, but Train A can only be operated in the Injection Mode without CCW available.

D. Incorrect. Plausible because both Trains of RHR can be operated in the Injection Mode, but only Train B RHR can be operated in the Recirculation Mode.

Technical Reference(s): EOS-1.3A, Attachment 3, Step 2 Bases; FRC-0.1A, Step 1 CAUTION — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from EOS-1.3, Transfer to Cold Leg Recirculation **STATE** the purpose/basis for the step(s).

Question Source: Bank ILOT5905
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8
55.43 ______

Comments / Reference: EOS-1.3A, Attachment 3, Step 2 Bases	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 37 OF 54

ATTACHMENT 3
PAGE 1 OF 18

BASES

CAUTION: Since the amount of water in the RWST between the switchover setpoint and the empty point is limited, the realignment of ECCS to cold leg recirculation must be done as quickly as possible.

A suction source of water for the ECCS pumps must be maintained to provide for core cooling. The actions of these first three steps must be completed even if challenges to a Critical Safety Function or Foldout Page criteria occur at this time, since these steps relate to the maintenance of core cooling.

If cold leg recirculation cannot be established or maintained, the operator is instructed to transition to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, before the completion of these steps. If a transition out of EOS-1.3A to ECA-1.1A is made, the Status Trees should be monitored and the caution no longer applies. A transition to ECA-1.1A is only permitted if neither a RED nor an ORANGE condition is detected on the Status Trees. The order of priority in this case is the switchover steps in EOS-1.3A identified in the caution, RED or ORANGE path FRGs if a transition out of EOS-1.3A occurs before the completion of these steps, then ECA-1.1A.

STEP 1: In order to realign or stop safeguards equipment, a deliberate action must be taken to reset the SI signal.

STEP 2: The RHR and CS heat exchangers are used for heat removal during the post accident recirculation phase and CCW flow should have already been established to the RHR and Containment Spray heat exchangers. If CCW flow has not previously been established, then it should be established at this time.

If CCW cannot be established to one heat exchanger, the remaining procedure steps can be performed as listed provided that the uncooled recirculation fluid temperature and pressure do not exceed equipment design conditions. RHR pumps should not pump water greater than 120°F without CCW to the RHR System.

Comments / Reference: FRC-0.1A, Step 1 CAUTION	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 8	PAGE 3 OF 45

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: RHR pumps should not pump water greater than 120°F without CCW to the RHR system.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 4	Group	1	
	K/A	015/017 G 2.2.22	
Level of Difficulty: 3	Importance Rating	4.0	

RCP Malfunctions: Equipment Control: Knowledge of limiting conditions for operations and safety limits

Proposed Question: 5

Given the following conditions:

- Unit 1 is in MODE 3 following a Refueling outage.
- Both Control Rod Drive Motor Generators are energized and Reactor Trip Breakers are CLOSED.

Which of the following identifies the Technical Specification Limiting Condition for Operation requirements for the Reactor Coolant Pumps in the current plant conditions?

A. All four RCPs shall be OPERABLE with four in operation.

B. Two RCPs shall be OPERABLE with two in operation.

C. Two RCPs shall be OPERABLE with one in operation.

D. One RCP shall be OPERABLE with one in operation.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because in MODES 1 and 2 all four loops shall be OPERABLE and in operation IAW TS 3.4.4 LCO.
B. Correct. In MODE 3 with rod withdrawal capable, 2 loops shall be OPERABLE with 2 loops in operation IAW TS 3.4.5 LCO.
C. Incorrect. Plausible because in MODE 3 with rod withdrawal NOT capable 2 loops shall be OPERABLE with 1 loop in operation IAW TS 3.4.5 LCO.
D. Incorrect. Plausible because it could be thought that 1 loop shall be OPERABLE with 1 loop in operation with the given conditions.

Technical Reference(s) Technical Specification LCO 3.4.4 / Technical Specification LCO 3.4.5 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Reactor Coolant system including Technical Specifications, TRM and ODCM.

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: Technical Specification LCO 3.4.4	Amendment: 161

RCS Loops -- MODES 1 and 2
3.4.4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops -- MODES 1 and 2

LCO 3.4.4 Four RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

Comments / Reference: Technical Specification LCO 3.4.5	Amendment: 161

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops -- MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

---NOTE---

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 3	Group	1	
	K/A	025 AK2.01	
Level of Difficulty: 3	Importance Rating	2.9	

Loss of RHR System: Knowledge of the interrelations between the Loss of RHR System and the following: RHR heat exchangers

Proposed Question: 6

Which of the following would prevent Residual Heat Removal Heat Exchanger 1-01 from performing its design function with Cold Leg Recirculation in progress?

A. A loss of Instrument Air to 1-HV-606, U1 RHR HX 1-01 FLO CTRL VLV.

B. Closing 1-HV-4572, RHR HX 1-01 CCW RET VLV.

C. A loss of Instrument Air to 1-FCV-618, RHR HX 1-01 BYP FLO CTRL VLV.

D. Closing 1CC-0109, RHR HX 1-01 CCW SPLY ISO VLV.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that the RHR Heat Exchanger Flow Control Valve will close on a loss of instrument air. However, this valve fails open on a loss of air.
B. Correct. As the plant is in Cold Leg Recirculation, SI has been reset and closing the RHR Heat Exchanger CCW Return Valve will secure CCW flow to the HX. This would not be the case if SI had not been reset.
C. Incorrect. Plausible if thought that the RHR Heat Exchanger Bypass Flow Control Valve will open on a loss of instrument air. However, this valve fails closed on a loss of air.
D. Incorrect. Plausible because a typical isolation valve would terminate flow when closed. However, the CCW Inlet Isolation valve for the RHR Heat Exchangers is normally closed and has an orifice in the disc that allows sufficient flow for the RHR system to meet design basis accident criteria when closed.

Technical Reference(s) ABN-301, Attachment 1
EOS-1.3A, Attachment 3, Step 2 & 15 Bases
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Residual Heat Removal system including interrelations with other systems to include interlocks and control loops.

Question Source:	Bank	ILOT8002	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: From EOS-1.3A, Attachment 3, Step 2 Bases	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 37 OF 54

ATTACHMENT 3
PAGE 1 OF 18

BASES

CAUTION: Since the amount of water in the RWST between the switchover setpoint and the empty point is limited, the realignment of ECCS to cold leg recirculation must be done as quickly as possible.

A suction source of water for the ECCS pumps must be maintained to provide for core cooling. The actions of these first three steps must be completed even if challenges to a Critical Safety Function or Foldout Page criteria occur at this time, since these steps relate to the maintenance of core cooling.

If cold leg recirculation cannot be established or maintained, the operator is instructed to transition to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, before the completion of these steps. If a transition out of EOS-1.3A to ECA-1.1A is made, the Status Trees should be monitored and the caution no longer applies. A transition to ECA-1.1A is only permitted if neither a RED nor an ORANGE condition is detected on the Status Trees. The order of priority in this case is the switchover steps in EOS-1.3A identified in the caution, RED or ORANGE path FRGs if a transition out of EOS-1.3A occurs before the completion of these steps, then ECA-1.1A.

STEP 1: In order to realign or stop safeguards equipment, a deliberate action must be taken to reset the SI signal.

STEP 2: The RHR and CS heat exchangers are used for heat removal during the post accident recirculation phase and CCW flow should have already been established to the RHR and Containment Spray heat exchangers. If CCW flow has not previously been established, then it should be established at this time.

If CCW cannot be established to one heat exchanger, the remaining procedure steps can be performed as listed provided that the uncooled recirculation fluid temperature and pressure do not exceed equipment design conditions. RHR pumps should not pump water greater than 120°F without CCW to the RHR System.

Comments / Reference: EOS-1.3A, Attachment 3, Step 15 Bases	Revision: 8

1CC-0109, RHR HX 1-01 CCW SPLY ISO VLV is a valve as described below.

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 50 OF 54

ATTACHMENT 3
PAGE 14 OF 18

BASES

The RHR and Containment Spray Heat Exchanger Supply Valves are manual operated valves with an orifice drilled in the valve disc which functions to restrict flow. With these valves in the closed position (normally closed in MODES 1, 2 and 3), the orifice is aligned directly in the flow path to provide an acceptable CCW flow value. The RHR and Containment Spray Heat Exchanger Return Valves are motor operated valves which function to automatically throttle flow to the required value following a design basis accident. The RHR and Containment Spray Heat Exchanger Supply and Return Valves collectively regulate CCW flow to the RHR and Containment Spray Heat Exchangers to remove the required heat from the RCS and the containment, while limiting the heat addition to the CCW System.

The manually operated RHR and Containment Spray Heat Exchanger Supply Valves are NOT required to be open to mitigate a design basis accident (e.g., LOCA). Likewise, the motor operated RHR and Containment Spray Heat Exchanger Return Valves are not required to be open past their throttled position to support accident analyses. However, the manually operated and motor operated valves may be opened as desired to accelerate cooldown provided the accident heat loads have sufficiently decayed. The radiological and environmental conditions of Safeguards Building 790' corridor must be considered when opening the manual valves.

Comments / Reference: ABN-301, Attachment 1	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 41 OF 122

ATTACHMENT 1
PAGE 4 OF 15

CONTROL BOARD AIR OPERATED VALVE FAILURE POSITIONS

LOCATION	COMPONENT	NOMENCLATURE	FAILURE POSITION
CB-04	1/u-8879C	RHR TO CL 3 TEST VLV	F.C.
CB-04	1/u-8879D	RHR TO CL 4 TEST VLV	F.C.
CB-04	1/u-8880	SI/PORV ACCUM N2 ISOL VLV	F.C.
CB-04	1/u-8882	CCP SI TEST VLV	F.C.
CB-04	1/u-8890A	RHR TO CL 1 & 2 TEST VLV	F.C.
CB-04	1/u-8890B	RHR TO CL 3 & 4 TEST VLV	F.C.
CB-04	u-FK-618	RHR HX 1 BYP FLO CTRL	F.C.
CB-04	u-FK-619	RHR HX 2 BYP FLO CTRL	F.C.
CB-04	u-HC-606	RHR HX 1 FLO CTRL	F.O.
CB-04	u-HC-607	RHR HX 2 FLO CTRL	F.O.
CB-04	u-HC-943	ACCUM 1-4 VENT CTRL	F.C.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 3	Group	1	
	K/A	026 AA1.06	
Level of Difficulty: 3	Importance Rating	2.9	

Loss of Component Cooling Water: Ability to operate and/or monitor the following as they apply to a Loss of Component Cooling Water: Control of flow rates to components cooled by the CCWS

Proposed Question: 7

Given the following conditions:

- Unit 1 is in MODE 3 with Train A Residual Heat Removal System operating in the Shutdown Cooling Mode.
- 1-ALB-3B, Window 4.5 – CCW HX 1/2 SPLY FLO LO is LIT.
- While performing actions in accordance with ALM-0032A, Alarm Procedure 1-ALB-3B, the crew observes the following:
  - CCW Pump 1-01 Discharge Pressure (1-PI-4520): 140 psig.
  - CCW HX 1-01 Outlet Flow (1-FI-4536A): 8000 gpm.
  - CCW HX 1-01 Recirculation Flow (1-FI-4536B): 8000 gpm.
  - CCW HX 1-01 Outlet Temperature (1-TI-4530): 100°F.
  - CCW Surge Tank Level (1-LI-4500): 68%.

Which of the following lists the action that should be performed in response to the CCW parameters in accordance with ALM-0032A?

A. Start SSWP 1-02 and CCWP 1-02 to share heat load in both trains.

B. Start CCWP 1-02 and place CCWP 1-01 in standby.

C. Transition to ABN-502, Component Cooling Water System Malfunction.

D. Close 1-HS-4536, CCWP 1-01 Recirculation Valve.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that 100°F at the heat exchanger outlet is too high, however, the alarm for this action is at 118°F.
B. Incorrect. Plausible because if thought that 100°F would require swapping CCWPs this action is directed at ≥ 120°F.
C. Incorrect. Plausible because if total flow was less than 10,000 gpm action required is a transition to ABN-502 for either a pump trip or loss of flow.
D. Correct. IAW ALM-0032A with total flow > 15,500 gpm with the Recirc valve open the expected action is to close the Recirc valve.

Technical Reference(s): ALM-0032A, Window 4.5, Logic Diagram; ALM-0032A, Window 4.5, Steps 2, 3, & 4; ALM-0032A, Window 1.5, Logic Diagram; ALM-0032A, Window 1.5, Step 1 & 1.B

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Component Cooling Water system.

ANALYZE the response to Loss of All CCW Flow in accordance with ABN-502, Component Cooling Water System Malfunction.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

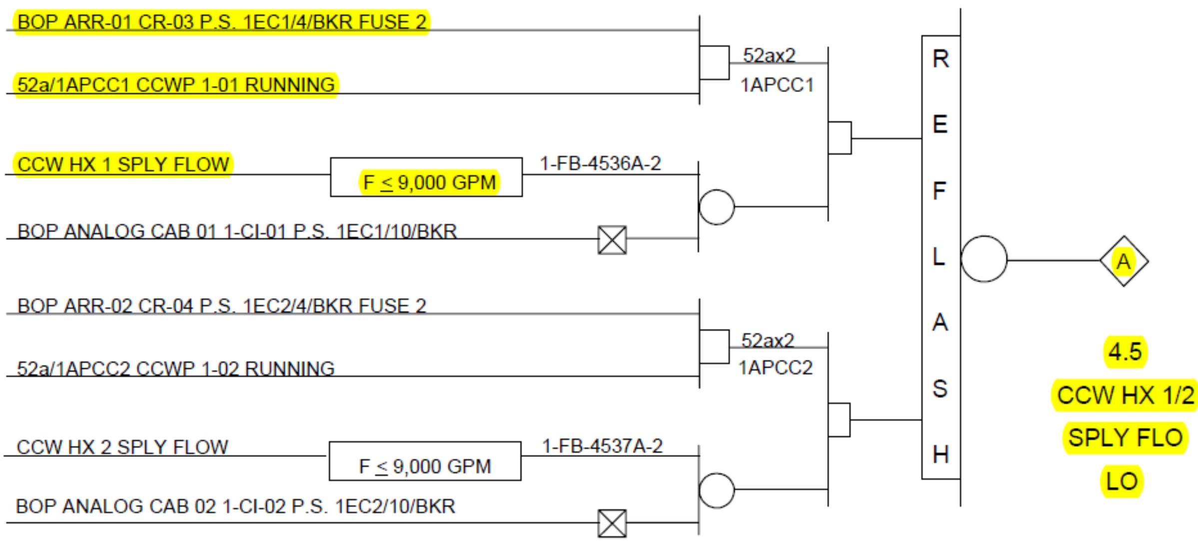
10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments / Reference: ALM-0032A, Window 4.5, Logic Diagram	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 142 OF 169

ANNUNCIATOR NO.: 4.5

LOGIC:



Comments / Reference: ALM-0032A, Window 4.5, Steps 2, 3, & 4	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 143 OF 169

ANNUNCIATOR NOM./NO.: **CCW HX 1/2 SPLY FLO LO** **4.5**

PROBABLE CAUSE:

CCW Pump 1-01 or 1-02 malfunction
Low system demand
Recirc control valve malfunction

CAUTION: Low system demand can result from low CCW supply temperature or when plant conditions require a minimum number of CCW loads to be in service. In these situations, recirc valve will be open and this will be a normal alarm.

AUTOMATIC ACTIONS: None

NOTE: CCW recirc control valve opens at approximately 8,200 gpm CCW heat exchanger outlet flow.

OPERATOR ACTIONS:

1. Determine affected train.
   - 1-HS-4518A, CCWP 1
   - 1-HS-4519A, CCWP 2

2. Monitor affected CCW heat exchanger outlet flow and recirculation flow.
   - 1-FI-4536A, CCW HX 1 OUT FLO
   - 1-FI-4536B, CCW HX 1 RECIRC FLO
   - 1-FI-4537A, CCW HX 2 OUT FLO
   - 1-FI-4537B, CCW HX 2 RECIRC FLO

   A. IF total flow is <10,000 gpm, THEN refer to ABN-502.

CAUTION: Do NOT exceed 17,500 gpm flow through a CCW pump.

3. Determine if recirc control valve is open.
   - 1-HS-4536, CCWP 1 RECIRC VLV
   - 1-HS-4537, CCWP 2 RECIRC VLV

NOTE: If CCW Heat Exchanger outlet flow is restored to >8,200 gpm with recirc valve open, a recirc control valve malfunction is indicated.

   A. If recirc valve is open and total flow is >15,500 gpm, attempt to close valve.

4. Monitor affected CCW heat exchanger outlet temperature.
   - 1-TI-4530, CCW HX 1 OUT TEMP
   - 1-TI-4534, CCW HX 2 OUT TEMP

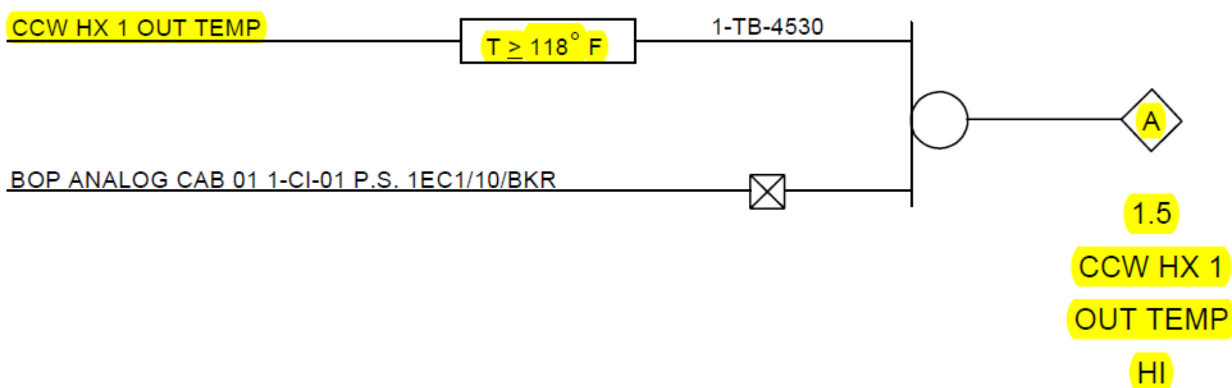
   A. If temperature is ≥120°F, start standby CCW pump and shutdown affected pump per SOP-502A.

Comments / Reference: ALM-0032A, Window 1.5, Logic Diagram	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 18 OF 169

ANNUNCIATOR NO.: 1.5

LOGIC:



Comments / Reference: ALM-0032A, Window 1.5, Step 1 & 1.B	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 19 OF 169

ANNUNCIATOR NOM./NO.: **CCW HX 1 OUT TEMP HI** 1.5

PROBABLE CAUSE:

Inadequate SSW flow through heat exchanger
Excessive CCW heat load
High SSW supply temperature

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

1. Verify SSW flow through CCW heat exchanger is between 14,900 and 15,500 gpm on Plant Computer.

CAUTION: Increasing discharge flow through CCW heat exchanger will decrease flow to other components in associated SSW header.

[IV] A. If flow is $<$ 14,900 gpm, dispatch an operator to slowly increase flow to 14,900 gpm using 1SW-0023, CCW HX 1-01 SSW OUT THROT VLV.

B. If flow is $\geq$ 14,900 gpm, indicating excessive CCW heat load, start standby SSW pump per SOP-501A and CCW pump per SOP-502A to share heat load between trains.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 3	Group	1	
	K/A	027 AK2.03	
Level of Difficulty: 3	Importance Rating	2.6	

Pressurizer Pressure Control Malfunction: Knowledge of the interrelations between Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Proposed Question: 8

Given the following conditions:

- Unit 2 is in MODE 1.
- All Pressurizer Backup Heaters have energized due to a Pressurizer level deviation high and CANNOT be secured.

What is the plant response?

A. Both Pressurizer Spray Valves open to restore pressure between 2220 psig and 2250 psig.

B. Only one Pressurizer Spray Valve opens to restore pressure between 2220 psig and 2250 psig.

C. Both Pressurizer Power Operated Relief Valves open to maintain pressure below 2335 psig.

D. Only one Pressurizer Power Operated Relief Valve opens to maintain pressure below 2335 psig.

Proposed Answer: A

Explanation:

A. Correct. With all backup heaters energized, both pressurizer spray valves will open. Pressurizer pressure will initially peak at approximately 2270 psig and then lower to the normal control band based on the response of 2-PK-455A, Pressurizer Master Pressure Controller which will reposition the spray valves as necessary.
B. Incorrect. Plausible if it is believed that a single pressurizer spray valve is sufficient to overcome the energy added by all backup heaters.
C. Incorrect. Plausible if it believed that both pressurizer spray valves are not sufficient to mitigate the energy added by all backup heaters.
D. Incorrect. Plausible if it believed that both pressurizer spray valves are not sufficient to mitigate the energy added by all backup heaters and both PORVs open at the same setpoint.

Technical Reference(s)	ABN-705, Section 3.2	Attached w/ Revision: See Comments / Reference
	LO21.SYS.PP1, Page 5	
	DBD-ME-250, Reactor Coolant System	

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Pressurizer Pressure Instrument Malfunction in accordance with ABN-705, Pressurizer Pressure Malfunction.

Question Source: Bank NRC 2013 44
Modified Bank ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam 2013

Question Cognitive Level: Memory or Fundamental Knowledge ________
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: ABN-705, Section 3.2	Revision: 12

This reference is included to indicate that spray flow will overcome heater input if a spray valve were to fail open. So that indicates that if heaters were to all come on that spray flow would overcome heater input and restore pressure to the control band.

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 12	PAGE 8 OF 26

3.0 Pressurizer Spray Valve Failure

3.1 Symptoms

a. Annunciator Alarms

- PRZR 1 OF 4 PRESS LO (5B-3.4)
- PRZR 1 OF 4 SI PRESS LO (5B-4.4)
- PRZR LO PRESS PORV 456 BLK (5B-1.6)
- PRZR LO PRESS PORV 455A BLK (5B-2.6)

b. Plant Indication

- Spray valve indicated open when not called for by master controller.

3.2 Automatic Actions

a. Control response for failed open spray valve(s)

1) Control and backup heaters come on.

- 1/u-PCPR, PRZR CTRL HTR GROUP C
- 1/u-PCPR1, PRZR BACKUP HTR GROUP A
- 1/u-PCPR2, PRZR BACKUP HTR GROUP B
- 1/u-PCPR3, PRZR BACKUP HTR GROUP D

NOTE: A reactor trip at high power and low pressure may result in an SI due to AFW flow.

2) Reactor trip at 1880 psig.

3) Safety Injection at 1820 psig.

b. Control response for failed close spray valves.

- PRZR PORV may open on a pressure transient.

Comments / Reference: From LO21.SYS.PP1, Page 5	Revision 05/05/11

During normal plant operations, the pressurizer is filled with boiling water and steam. The temperature of the boiling (or saturated) water determines the pressure inside the entire Reactor Coolant System. Pressurizer temperature is controlled to regulate RCS pressure by energizing electric heaters in the bottom of the pressurizer to raise pressure, and by spraying the steam space (or steam bubble) in the top of the pressurizer with cooler water to reduce pressure.

Under normal operating conditions, the Pressurizer Pressure Control System will automatically maintain the plant at 2235 psig. Heaters maintain a saturated condition in the pressurizer and spray valves throttle open to hold pressure at the 2235 psig setpoint. Backup banks of heaters energize on decreasing RCS pressure. On increasing pressure, spray valves open automatically to cause partial steam bubble condensation.

Comments / Reference: From DBD-ME-250, Reactor Coolant System	Revision 41

The pressurizer spray valves are required to pass the maximum cold leg spray flow sufficient to maintain the system pressure below the power operated relief valve setpoint for a 12 percent load rejection at 100 percent power.

The pressurizer spray line is required to deliver 900 gpm to the pressurizer.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to reduce pressure variations due to contraction and expansion of the reactor coolant.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	1	
	K/A	029 EK1.01	
Level of Difficulty: 3	Importance Rating	2.8	

ATWS: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermal-hydraulics behavior

Proposed Question: 9

Given the following conditions:

- Unit 1 was operating at 100% power, MOL.
- Both Main Feedwater Pumps tripped.
- An automatic Reactor Trip did NOT occur.
- Attempts to manually trip the Reactor have NOT been successful.
- The Turbine has been manually tripped.
- Both Motor Driven Auxiliary Feedwater Pumps are feeding all four Steam Generators at 1200 gpm.
- The Turbine Driven Auxiliary Feedwater Pump has tripped and CANNOT be reset.

Which of the following describes the expected Reactor core and Pressurizer pressure response prior to locally tripping the Reactor?

A. Total core power LOWERS due to Moderator Temperature Coefficient.
Pressurizer pressure LOWERS due to Pressurizer Spray Valves opening.

B. Total core power LOWERS due to Moderator Temperature Coefficient.
Pressurizer pressure RISES due to available heat removal capability.

C. Total core power RISES due to Pressure Coefficient.
Pressurizer pressure LOWERS due to Pressurizer Spray Valves opening.

D. Total core power RISES due to Pressure Coefficient.
Pressurizer pressure RISES due to available heat removal capability.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because as RCS density lowers due to decreased heat removal the MTC adds negative reactivity to the core causing total core power to lower and the effect from pressurizer spray on pressurizer pressure is negligible compared to the pressure rise from the heat imbalance.

B. Correct. As RCS density lowers due to decreased heat removal the MTC adds negative reactivity to the core causing total core power to lower. Pressurizer pressure rises due to the imbalance between the heat source (reactor) and the heat sink (steam generators) that are being fed at well below the capability of the MFPs.

C. Incorrect. Plausible because the pressure coefficient will add positive reactivity to the core, however, the effect from the negative reactivity from MTC cause total core power to lower and the effect from pressurizer spray on pressurizer pressure is negligible compared to the pressure rise from the heat imbalance.

D. Incorrect. Plausible because the pressure coefficient will add positive reactivity to the core, however, the effect from the negative reactivity from MTC cause total core power to lower. Pressurizer pressure rises due to the imbalance between the heat source (reactor) and the heat sink (steam generators) that are being fed at well below the capability of the MFPs.

Technical Reference(s) LO21.MCO.MI5, Pages 12 & 15; LO21.GFR.COF, Pages 4 & 8 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the Anticipated Transient Without Trip analysis.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 5
55.43 ____

Comments / Reference: LO21.GFR.COF, Page 4	Revision: 11/29/07

Decreasing moderator density increases the migration length of the neutrons, which leads to an increase in the fraction of neutrons that leak out of the core, therefore decreasing the non-leakage factors. For large commercial power reactors neutron leakage is insignificant.

The fast fission factor increases slightly due to increased slowing down length, but the effect is very small.

The reproduction factor is not dependent on moderator density, so it does not change significantly as moderator temperature changes.

As Figure 4-2 shows, moderator temperature changes result in essentially two competing processes, the resonance escape probability (p) and the thermal utilization factor (f). The resonance escape probability has the dominant effect, causing k_{eff} and reactor power to decrease as moderator temperature increases. Since increasing moderator temperature (decreasing the moderator to fuel ratio) decreases k_{eff}, the moderator temperature coefficient is negative.

The region to the left of the maximum effective neutron multiplication factor is called the under-moderated region. Note that in this region, an increase in temperature results in a reduction of the effective neutron multiplication factor. This results in a negative moderator temperature coefficient. Operating in the under-moderated region is very important in terms of reactor control. If reactor power suddenly increases, the moderator temperature will rise, inserting negative reactivity into the system and thus limiting the power excursion. Commercial reactors are designed with a moderator-to-fuel ratio such that the moderator temperature coefficient is negative.

The region to the right of the maximum effective neutron multiplication factor is called the over-moderated region. In the over-moderated region the reduction in moderator density has a greater effect on the thermal utilization factor than the resonance escape probability. The increased thermal utilization causes a positive reactivity addition with increasing moderator temperature. If the reactor were allowed to operate on the over-moderated side of the curve, any increase in power would cause an increase in moderator temperature, adding positive reactivity and accelerating the power increase.

Also, at higher temperatures, the moderator temperature coefficient becomes more negative due to a larger change in density for the same change in temperature.

CHANGE IN MODERATOR TEMPERATURE COEFFICIENT WITH BORON CONCENTRATION

The moderator temperature coefficient has been discussed under the implicit assumption that the moderator is "pure" water. In a PWR, however, boron, in the form of boric acid, is added to the moderator/coolant. Boron has a high thermal neutron cross section and acts as a soluble "control rod." As the reactor operates and fuel is consumed, the boron is removed (diluted) to compensate for the decrease in reactivity due to fuel depletion. The presence of boron in the moderator alters the value of the moderator temperature coefficient, depending on the boron concentration.

Comments / Reference: LO21.GFR.COF, Page 8	Revision: 11/29/07

PRESSURE COEFFICIENT (α_{PSI})

Since the moderator is a fluid, its density depends, to a very small extent, on reactor pressure. Increases in reactor pressure result in an increase of moderator density, an effect which is directly opposite to that of increased temperature. Therefore, the pressure coefficient α_{psi} will be positive and is approximately 1×10^{-6} $\Delta k/k/psi$. As a rule of thumb, a 100 psi increase in pressure introduces the same reactivity as a one degree decrease in temperature. Since pressure is maintained constant, this coefficient has little effect.

DOPPLER COEFFICIENT (α_d)

THE DOPPLER EFFECT

In order to understand what the Doppler Effect is and how it affects the neutron absorption process, a brief review of the general principle behind this effect will first be presented.

The Doppler Effect was originally associated with the physics of sound and light. Most people have observed this phenomenon without realizing its cause. For example, when a source of sound, or a listener, or both, are in motion relative to the air, the pitch of the sound, as heard by the listener, is in general not the same as when the source and the listener are at rest.

The most common example is the sudden drop in the pitch of the sound from an automobile horn as one meets and passes a car traveling in the opposite direction. This phenomenon can be explained by considering the velocity of the sound waves. Sound has a velocity of about 730 miles per hour in air. If a racecar approached an observer at 140 mph, the relative speed of the sound wave to the stationary observer is 870 mph. However, if the racecar is going away from the observer, the speed relative to the observer is 590 mph. The difference in the velocity of the waves results in a change in the pitch of the sound.

The probability of resonant neutron absorption as a function of fuel temperature can be explained in terms of the Doppler Effect. As previously described, neutrons give up energy in step changes through collisions with nuclei. The microscopic cross section for absorption (σ_a) for U-238 is 5500 barns for neutrons at an energy level of 21 eV. But the σ_a is only 15 to 20 barns for a neutron with energy levels of 20 or 22 eV.

Comments / Reference: LO21.MCO.MI5, Page 12	Revision: 01/09/12

LOSS OF FEEDWATER ATWT

CAUSES

Loss of normal feedwater could result from a malfunction in the condensate system or its control system from such causes as a trip of a condensate pump, simultaneous trip of both main feedwater pumps (or closure of their discharge valves), or simultaneous closure of all feedwater control valves. The vast majority of these cases would cause only a partial loss of feedwater flow. The most likely cause of a complete loss of feedwater would be loss of station power.

TRANSIENT DESCRIPTION

The loss of feedwater produces a large imbalance in the heat source/ heat sink relationship. When feedwater flow to the SGs is terminated, the secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising RCS temperature and pressure, and by increasing pressurizer water level, which is due to the insurge of expanding reactor coolant. Water level in the SGs drops as the remaining water in the secondary system is boiled off. When SG level falls to the point where the SG tubes are exposed and the primary-to-secondary heat transfer is reduced, the reactor coolant temperature and pressure begin to increase at a greater rate. This greater rate of increase is maintained as the pressurizer fills and releases water through the PORVs and safety valves. (The PORVs and safety valves have a smaller volumetric relief capacity for water than for steam.) Negative reactivity feedback, from the moderator temperature coefficient as primary temperature rises, reduces core power. Pressure begins to decrease and a steam bubble is again formed in the pressurizer. The event time line is shown in Table 1.

Comments / Reference: LO21.MCO.MI5, Page 15	Revision: 01/09/12

SENSITIVITY STUDIES (PARAMETRIC VARIATIONS)

The Loss of Feedwater ATWT was also subjected to sensitivity studies which analyzed changes in significant assumptions and parameters to determine their effect on RCS over pressure. The results of these studies are discussed below.

- Effect of Not Tripping the Turbine

  Failure to trip the turbine permitted a higher steam release from the SGs. In addition, more heat was removed from the primary system early in the transient. The core power level stayed relatively high and the primary pressure attained a higher maximum value than for the case in which the turbine was tripped.

- Effect of Not Opening the PORVs

  With only the three pressurizer safety valves available for steam and water relief, the peak pressures attained increased by about 9 percent.

- Effect of Not Using Pressurizer Spray

  Addition of spray water into the pressurizer steam space decreased early in the transient for the base case. Thus, the use of no pressurizer spray did not significantly affect the peak pressure reached during the transient.

- Effect of using Automatic Rod Control

  Allow the automatic insertion of control rods to compensate for rising reactor coolant temperature reduced the coolant expansion rate to the point that the pressurizer safety valves easily relieved the coolant insurge without even reaching the full open position (at 2590 psia). Also, the temperature and pressure transients were controlled to the extent that the reactor coolant pumps (RCPs) did not cavitate and the PORVs were able to limit the pressurizer pressure to about 2350 psia. When the pressurizer filled and water was released through the valves, pressure rose rapidly.

- Effect of Variation in Initial Average Coolant Temperature

  For the purpose of determining the effect of initial average coolant temperature on the Loss of Feedwater ATWT, and in order to encompass the average coolant temperatures of a variety of Westinghouse plants, analyses were conducted assuming +8°F and -20°F variation in initial average coolant temperature.

- Effect of Variation in Initial Pressurizer Level

  Variation of ± 10 percent in initial pressurizer level was considered. The higher water level meant that the pressurizer filled to capacity earlier in the transient when the core power level was still relatively high. A lower than normal water level delayed the filling of the pressurizer and provided more steam for volumetric relief through the valves and resulted in a lower pressurizer pressure.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 3	Group	1	
	K/A	040 AK3.06	
Level of Difficulty: 2	Importance Rating	3.4	

Steam Line Rupture: Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Containment temperature and pressure considerations

Proposed Question: 10

Following a Steam Line Break accident inside Containment, the operator is expected to stop Auxiliary Feedwater flow to the faulted Steam Generator within 10 minutes in order to prevent exceeding...

A. ...Motor Driven Auxiliary Feedwater Pump capacity.

B. ...Containment design temperature and pressure.

C. ...Steam Generator tubesheet and U-tube stress limits.

D. ...Containment Building designed flooding level.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that isolating AFW flow would prevent exceeding the capacity of the AFW pumps due to the reduced resistance to flow from a faulted SG; however, orifices in the AFW feed lines to the SGs limit flow.
B. Correct. STI-214.01 lists isolating AFW flow to a faulted SG inside CNTMT as a timed operator action that is taken within 10 minutes to ensure CNTMT design temperature and pressure is not exceeded.
C. Incorrect. Plausible because it could be thought that stopping AFW flow is to limit stress on the SG tubesheet and U-tubes; however, the Steam Generator flow restrictor at the top of the SG is designed to limit stress on the tubesheet and U-tubes.
D. Incorrect. Plausible because FRZ-0.2A considers AFW as a possible source of CNTMT flooding.

Technical Reference(s):
- STI-214.01, Attachment 8.A
- LO21.SYS.AF1, Page 12
- LO21.SYS.MR1, Page 8
- FRZ-0.2A, Step 1

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Auxiliary Feedwater system including Technical Specifications, TRM and ODCM.

Question Source:	Bank		
	Modified Bank	LORT0662	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: STI-214.01, Attachment 8.A	Revision: 0

CPNPP STATION INSTRUCTIONS MANUAL		PROCEDURE NO. STI-214.01
CONTROL OF TIMED OPERATOR ACTIONS	REVISION NO. 0 INFORMATION USE	PAGE 17 OF 45

ATTACHMENT 8.A
PAGE 1 OF 6

CPNPP Timed Operator Actions WITH ≤ 30 MINUTE RESPONSE TIME

TCA# / Val Meth	Time Critical Action(s)	Procedure	LBD/DBD Reference	Time Considerations for Procedure Development	Task Objective for Training Observation
1.0 Time Critical Actions (TCA)					
TCA-1.1 / Sim	Isolate AFW Flow to Faulted SG Following a Feed Line Break	EOP-0.0A/B, EOP-2.0A/B	FSAR 15.2.8, II.E.1.1; DBD-ME-206	Within 30 minutes after event initiation Reactor trip on SG low-low level.	Limit/terminate the excessive cooldown associated with faulted SG and terminate break flow from faulted SG. Ensure within 10CFR100 dose limits & containment design pressure not exceeded.
TCA-1.2 / Sim	Isolate AFW Flow to Faulted SG Following a Steam Line Break Inside Containment	EOP-0.0A/B, EOP-2.0A/B	FSAR 15.1.5; FSAR II.E.1.1; DBD-ME-008; DBD-ME-206	Within 10 minutes after event initiation.	Ensure containment design temperature and pressure not exceeded, and terminate excessive cooldown.

Comments / Reference: LO21.SYS.AF1, Page 12	Revision: 05/11/11

LESSON PLAN

NOTES	LESSON OUTLINE

1. Each MDAFW pump discharge line branches into individual lines feeding its two associated SGs.
   a. Each SG AFW line is provided with a normally open, pneumatically operated flow control valve.
   b. The flow control valves fail open on loss of air or electrical power.
2. Each valve is provided with a safety class air accumulator sized for five full cycles, plus leakage and steady state consumption for 30 minutes.
   a. This allows the valve to regulate AFW flow following a loss of Instrument Air coincident with a plant condition which requires AFW operation, or to isolate a faulted SG when the normal motor operated isolation valves are not available.
   b. The flow control valves are located near the pumps to allow for manual operation.
3. Manual/auto (M/A) controllers on the Main Control Board enable the operator to control flow manually from the Control Room.
   a. Upon automatic start of the MDAFW pumps, the flow control valves will automatically trip from manual to automatic control and position full open to ensure flow to the SGs.
   b. After a 10 second time delay the flow control valves can be manually positioned by the operator to adjust flow to the SGs.
   c. M/A controllers for these valves on the RSP enable the operator to control flow from the RSP when the RSP controllers are placed in manual.
   d. When in automatic, these controllers allow feed control to be accomplished at the Main Control Board.
4. A flow restricting orifice is provided downstream of each flow control valve. The orifice is designed to limit the maximum flow to a faulted SG to 700 gpm and prevent a pump runout condition.

Comments / Reference: LO21.SYS.MR1, Page 8	Revision: 09/04/13

Steam Generator Flow Restrictor

Each SG outlet nozzle contains an integral flow restrictor made up of seven (7) venturi type flow nozzles having a total flow area of 1.388 ft^2. During normal operation, the flow restrictor provides a 2 to 3 psi ΔP for steam flow measurement.

Under accident conditions, a double ended rupture (DER) of the main steam line would theoretically expose a 4.6 ft^2 break area to be available for blowdown flow based upon main steam line piping size. However, the integral flow restrictor limits the maximum break area per SG to 1.388 ft^2. From a safety analysis perspective, limiting break size and thus break flow provides several protective advantages:

- Protects fuel integrity by limiting the primary system cooldown rate thereby reducing the reactivity addition rate ensuring a departure from nucleate boiling (DNB) condition does not occur during the reactors return to power.
- Protects containment integrity by limiting the containment temperature and pressure rise.
- Reduces thrust forces on main steam line piping.
- Limits stresses on internal SG components, particularly the tubesheet and U-tubes.

Comments / Reference: FRZ-0.2A, Step 1	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRZ-0.2A
RESPONSE TO CONTAINMENT FLOODING	REVISION NO. 8	PAGE 3 OF 9

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check The Following Systems For Indication Of Possible Source Of Water To The Containment Sump (i.e., Pressure, Surge Tank Level, Flow, etc.): • RMUW • Demineralized water • CCW • Chemical and Volume Control System • Main Feedwater • AFW • Ventilation Chilled Water • Fire protection water	

Original Question: CPNPP Exam Bank LORT0662

For a steam line break accident inside Containment, the operator is expected to take manual action to stop AFW flow to the faulted SG within (1) after the initiation of the break in order to prevent exceeding (2) .

	(1)	(2)
A.	60 seconds	pump flow capacity
B.	600 seconds	Containment internal pressure
C.	60 minutes	SG tubesheet delta T
D.	600 minutes	Containment flooding level

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	1	
	K/A	055 EK3.02	
Level of Difficulty: 3	Importance Rating	4.3	

Station Blackout: Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power

Proposed Question: 11

Which of the following describes the reasons for depressurizing the Steam Generators to 270 psig in accordance with ECA-0.0A, Loss of All AC Power?

A. Initiates Safety Injection System Accumulator discharge and minimizes Reactor Coolant Pump seal leakage.

B. Establishes Natural Circulation conditions and initiates Safety Injection System Accumulator discharge.

C. Establishes Natural Circulation conditions and minimizes secondary heat sink requirements if Auxiliary Feedwater inventory is limited.

D. Minimizes secondary heat sink requirements if Auxiliary Feed inventory is limited and minimizes RCP seal leakage.

Proposed Answer: A

Explanation:

A. Correct. Lowering RCS pressure and restoring lost inventory is the reason for depressurizing.
B. Incorrect. Plausible because RCS depressurization will assist Natural Circulation, but is not the reason for depressurization to 270 psig.
C. Incorrect. Plausible because Natural Circulation will be established as a byproduct of rapid depressurization. Rapid cooldown and depressurization due to limited AFW is an action that could be taken in E-3 series procedures.
D. Incorrect. Plausible because in E-3 series procedures, rapid secondary depressurizations may be performed when there is limited makeup availability.

Technical Reference(s): ECA-0.0A, Step 18; ECA-0.0A, Attachment 7, Step 18 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-0.0, Loss of All AC Power.

Question Source:	Bank	ILOT6035	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	4	
	55.43		

Comments / Reference: ECA-0.0A, Step 18	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 18 OF 88

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		2) Continue with Step 19. <u>WHEN</u> narrow range level greater than 43% (50% FOR ADVERSE CONTAINMENT) in at least one intact SG, <u>THEN</u> do Steps 18b, 18c, 18d and 18e.
	b. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR	
	c. Manually dump steam using SG atmospheric(s).	c. Locally dump steam using SG atmospheric(s).
	d. Check SG pressures - LESS THAN 270 PSIG	d. Continue with Step 19. <u>WHEN</u> SG pressures decreased to less than 270 psig, <u>THEN</u> do Step 18e.
	e. Manually control SG atmospheric(s) to maintain SG pressures at 270 psig.	e. Locally control SG atmospheric(s) to maintain SG pressure at 270 psig.

Comments / Reference: ECA-0.0A, Attachment 7, Step 18 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 73 OF 88

ATTACHMENT 7
PAGE 15 OF 30

BASES

NOTE: Loss of pressurizer level and reactor vessel upper head voiding may result from the rapid depressurization of the intact steam generators. Such a condition is anticipated and should not interfere with operator actions in this step to depressurize the steam generators to reduce RCS pressure and temperature and to minimize RCS inventory loss out of the RCP seals.

Step 18: This step depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss.

During SG depressurization, SG level must be maintained above the top of the SG U-tubes, in at least one SG. Maintaining the U-tubes covered in at least one SG will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. This step requires that SG level be in narrow range in at least one SG before SG depressurization is initiated. If level is not in narrow range in at least one SG, the RNO instructs the operator to maintain maximum AFW flow until narrow range level is established in one SG. When narrow range level is established, SG depressurization can be started or continued.

This step instructs the operator to reduce SG pressures by depressurizing the intact SGs. Depressurization should be accomplished by opening the ARVs on the intact SGs to establish a steam dump rate such that RCS cooldown of 100°F/hr is not exceeded. At the onset of this event,high temperature RCS fluid will be leaking past the RCP seals causing a heat-up of the seal area. A RCS cooldown rate of 100°F/hr provides a rapid RCS temperature reduction while maintaining an acceptable cooldown across the RCP seal as the RCS fluid leaking past the seal is cooled as part of the RCS cooldown. The SG ARVs are air-operated with reserve accumulators and have DC control power; therefore, control is available from the Control Room.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 4	Group	1	
	K/A	056 AA2.17	
Level of Difficulty: 4	Importance Rating	3.4	

Loss of Offsite Power: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of PZR backup heaters

Proposed Question: 12

Given the following conditions:

- A Loss of Offsite Power has occurred on Unit 1.
- During plant recovery, the crew resets the Blackout Sequencer.
- Current Pressurizer pressure is 2200 psig and slowly lowering.
- The output of Pressurizer Master Pressure Controller, 1-PK-455A, indicates 41%.

Which of the following describes how Pressurizer Heater control is restored under these conditions?

The breakers for Pressurizer Heater Banks…

A. …A, B and D must be closed locally; Control Bank C power is restored with NO additional operator action.

B. …A, B and D must be closed locally; Control Bank C must be momentarily placed to ON.

C. …A, B and D will close if left in AUTO or ON; Control Bank C power is restored with NO additional operator action.

D. …A, B and D will close if left in AUTO or ON; Control Bank C must be momentarily placed to ON.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that automatic control is lost when the Blackout Sequencer (BOS) actuates, however, automatic control is restored when the BOS is reset. Additionally, without a demand signal present Control Bank C must be momentarily placed in ON.

B. Incorrect. Plausible if thought that automatic control is lost when the Blackout Sequencer (BOS) actuates, however, automatic control is restored when the BOS is reset. Placing Control Bank C momentarily in ON is the correct action.

C. Incorrect. Plausible when the BOS is reset, the backup heaters will operate as stated. Additionally, without a demand signal present. Control Bank C must be momentarily placed in ON.

D. Correct. When the BOS is reset, the backup heaters will operate as stated. Without a demand signal present, Control Bank C must be momentarily placed to ON.

Technical Reference(s) LO21.SYS.PP1, Pages 7 & 8
ABN-602, Step 10.3.9 NOTE
ABN-602, Attachment 1

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Pressurizer Pressure and Level Control System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank ILOT1967
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments / Reference: LO21.SYS.PP1, Page 7	Revision: 05/05/11

The compensated error signal from the master pressure controller controls spray valves, heaters, and PORV u-PCV-0455A in order to drive pressure to the 2235 psig setpoint. The actual pressure at which the spray valves, heaters or PORV operate is based upon the amount of time that actual pressure is off setpoint. Time passing with pressure off setpoint causes the compensated error signal to continue to increase even if the actual pressure deviation is constant. Controller output automatically changes, operating equipment as necessary to bring pressure back to 2235 psig. The master pressure controller is manually operated by setting the output signal to operate the desired equipment. The following table equates equipment operation with controller output and uncompensated error.

ACTION	**OUTPUT %**	**ERROR SIGNAL**	**NOMINAL VALUE**
PORV opens	81.3	+100 psig	2335 psig
PORV closes	75.0	+80 psig	2315 psig
Spray valves fully open	73.4	+75 psig	2310 psig
Spray valves start to open	57.8	+25 psig	2260 psig
Variable heaters off	54.7	+15 psig	2250 psig
Normal operating pressure	50.0	- 0 -	2235 psig
Variable heaters fully on	45.3	–15 psig	2220 psig
Backup heaters off	44.7	–17 psig	2218 psig
Backup heaters on	42.2	–25 psig	2210 psig

Comments / Reference: LO21.SYS.PP1, Page 8	Revision: 05/05/11

Pressurizer Heaters

The PRZR heaters consist of 78 elements mounted vertically in the bottom of the PRZR. The PRZR heaters are divided into four groups identified as A, B, C, and D. Groups A and B each have 21 heater elements and a heat capacity of 485 KW. Groups C and D each have 18 heater elements and a heat capacity of 416 KW. The total heater capacity is 1802 KW.

Groups A, B, and D are called "backup heaters." The backup heaters are energized by closing their power supply breakers in switchgear uEB2, uEB3 and uEB4. Each group has a 3-position maintained (OFF-AUTO-ON) handswitch located on u-CB-05. The backup heater power supply breaker is closed by placing the handswitch in ON or by a low pressure signal from the master pressure controller when the handswitch is in AUTO. Backup heaters in AUTO will also be energized by pressurizer level deviation of 5% above program level. Groups A and B may be operated from the Remote Shutdown Panel.

Group C is the "control heaters," also called variable or proportional heaters. These heaters operate with variable output controlled by the master pressure controller. A 3-position (OFF-neutral-ON) spring-return to center handswitch operates the control heaters from u-CB-05. During normal operation, the handswitch is taken to the ON position, closing the power supply breaker in switchgear uEB1, and released to the center position. A silicon controlled rectifier (SCR) circuit supplies power to the heater elements using a time-proportioned average output voltage based on the control signal from the master pressure controller. This means that a full 480 VAC is supplied to the heaters in pulses such that the average voltage supplied over time is proportional to the pressure controller output. The control heater power supply breaker will not close automatically.

Comments / Reference: ABN-602, Step 10.3.9 NOTE	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 59 OF 107

10.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Automatic Back-up Heater Control returns when Blackout Sequencer is reset.

☐ 9 Manually control Pressurizer Heaters as needed to maintain RCS pressure

Comments / Reference: ABN-602, Attachment 1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 66 OF 107

ATTACHMENT 1
PAGE 2 OF 12

6900/480 V SWITCHGEAR UNDERVOLTAGE LOAD SHEDDING

1. b. Bus 1EB1

1) PDP

2) CNTMT FN CLR FN 1

* 3) MCC XEB1-3

* 4) MCC 1EB1-2

* 5) MCC 1EB1-3

** 6) PRZR CTRL HTR GROUP C

* Supply breakers only trip if the following occur: 1) Train A DG is ready to load; 2) Breakers 1EA1-1 AND 1EA1-2 are open; 3) An undervoltage condition exists on 1EB1 OR Train A DG is supplying Train A power.

** Supply breaker must be manually closed following restoration of power to Bus 1EB1. IF Blackout Sequencer has actuated, THEN supply breaker can NOT be closed until approximately 120 seconds after power restored.

c. Bus 1EB3

1) CRDM VENT FN 1

2) CNTMT FN CLR FN 3

3) EMER FILL/FIRE BRGD TRNG PUMP

** 4) PRZR BACKUP HTR GROUP A

* 5) MCC XEB3-1

NOTE:
- The conditions of * above will also trip XEB3-3/2M/BKR, 480 VAC MCC XEB3-3 INNER BUS TIE BREAKER, if XEB3-3 is powered by affected unit.
- IF MCC XEB3-1 is de-energized, THEN ensure Security has been notified to implement STA-919 controls due to loss of panel XF-AF-1 (AAP Ltg. Panel AP-1) and partial detection equipment including AAP pop up barrier deployment.

* Supply breaker only trips if the following occur: 1) Train A DG is ready to load: 2) Breakers 1EA1-1 AND 1EA1-2 are open; 3) An undervoltage condition exists on 1EB3 OR Train A DG is supplying Train A power.

** IF Blackout Sequencer has actuated, THEN after approximately 120 seconds, backup heaters can be manually controlled. Automatic control returns when sequencer is reset.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/25/2014	Tier	1	
Change: 3	Group	1	
	K/A	058 AA1.03	
Level of Difficulty: 3	Importance Rating	3.1	

Loss of DC Power: Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Vital and battery bus components

Proposed Question: 13

Given the following conditions:

- Unit 2 is at 100% power with all systems in their normal alignment.
- A loss of 2ED1, 125 VDC Switch Panel occurs.

Which of the following are the correct component responses?

The Turbine Driven Auxiliary Feedwater Pumps starts…

A. …WITH speed indication.
The Feedwater Isolation Valves fail close.

B. …WITH speed indication.
The Feedwater Control Valves fail close.

C. …WITHOUT speed indication.
The Feedwater Isolation Valves fail close.

D. …WITHOUT speed indication.
The Feedwater Control Valves fail close.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the TDAFWP does start, but speed indication is also lost. The Feedwater Isolation Valves do not fail close on the loss of 2ED1, but are plausible as they fail close on the loss of 2EC1.
B. Incorrect. Plausible because the TDAFWP does start, but speed indication is also lost. The Feedwater Control Valves which fail closed on a loss of 2ED1 and result in a reactor trip occurring.
C. Incorrect. Plausible because the TDAFWP which auto starts when the Train A steam admission valve fails open on a loss of 2ED1. Speed indication is lost when the bus is lost. The Feedwater Isolation Valves do not fail close on the loss of 2ED1, but are plausible as they fail close on the loss of 2EC1.
D. Correct. The first vital component listed is the TDAFWP which auto starts when the Train A steam admission valve fails open on a loss of 2ED1. Speed indication is lost when the bus is lost. The second vital components are the Feedwater Control Valves which fail closed on a loss of 2ED1 and result in a reactor trip occurring.

Technical Reference(s): ALM-0102B, Window 1.13; ABN-603, Section 3.1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Solid State Protection System including interrelations with other systems to include interlocks and control loops.

Question Source:

Bank		
Modified Bank		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	6
55.43	

Comments / Reference: ALM-0102B, Window 1.13	Revision: 6

CPNPP ALARM PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. ALM-0102B
ALARM PROCEDURE 2-ALB-10B	REVISION NO. 6	PAGE 68 OF 354

ANNUNCIATOR NOM./NO.: **125VDC SWITCH PNL 2ED1 TRBL** **1.13**

PROBABLE CAUSE:

Blown fuse
Battery charger malfunction
Bus ground
Battery charger in equalize

NOTE: An undervoltage condition OR blown fuse will be indicated by an associated 125VDC alarm on 2-SSII-1 TRN A.

AUTOMATIC ACTIONS: None

NOTE: A loss of power to switch panel 2ED1 will cause a loss of the following equipment: Steam Dumps, TDAFWP Speed Indication, Feedwater Valves, Train A DG, Pressurizer Auxiliary Spray, Pressurizer PORV 455A, VCT level control, Letdown, Trip AND Close Power to Train A Switchgear, Steam Generator Blowdown AND Sampling. The following Valves will fail open: TDAFWP Steam Supply 2-HV-2452-1 AND Train A MDAFWP flow control AND recirc valves.

Comments / Reference: ABN-603, Section 3.1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 16 OF 34

3.0 LOSS OF INSTRUMENT BUS

3.1 Symptoms

a. The affected inverter trouble alarm:

- 118V INV IVuEC1 TRBL (10B-1.15)
- 118V INV IVuEC2 TRBL (10B-2.15)
- 118V INV IVuEC3 TRBL (10B-2.18)
- 118V INV IVuEC4 TRBL (10B-3.18)
- 118V INV IVuEC1/3 TRBL (10B-1.18)
- 118V INV IVuEC2/4 TRBL (10B-4.18)

b. The associated bus instruments alarming or failing (see Attachments 3 and 4):

- uEC1 from IVuEC1
- uEC2 from IVuEC2
- uEC5 from IVuEC3
- uEC6 from IVuEC4

NOTE: **(Unit 2 only)** On loss of 2EC1 or 2EC2, the FWIVs close due to loss of water hammer interlocks and the FPBVs open. During low power operations, this could cause overheating of the containment penetrations. 2-ALB-8A, 1.5, 2.5, 3.5 and 4.5 contain actions should this occur.

c. **(Unit 2 only)** A feed isolation will occur, FWIVs close (loss of 2EC1 or 2EC2).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 3	Group	1	
	K/A	062 AK3.02	
Level of Difficulty: 4	Importance Rating	3.6	

Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS

Proposed Question: 14

Given the following conditions:

- Unit 1 has experienced a complete Loss of Offsite and Onsite AC power.
- The crew is performing ECA-0.0A, Loss of All AC Power.
- Station Service Water (SSW) is NOT available from Unit 2, resulting in a complete loss of Unit 1 SSW.
- While performing ECA-0.0A, Safety Injection actuated and was RESET.
- The Safety Injection Sequencers were also RESET.
- Power is being restored from the Alternate Power Generators to Bus 1EA1.
    - ABN-601, Response to a 138/345 KV System Malfunction has been performed to restore power to Train A Safeguards 6.9 KV Bus 1EA1.
    - SOP-614A, Alternate Power Generator Operation has been performed to align system components for Alternate Power Generator operation.

Which of the following indicates the status of the SSW system when Bus 1EA1 is energized?

SSW Pump 1-01…

A. …STARTS with 1-HS-4393, DG 1 CLR SSW RET VLV OPEN.

B. …is in PULLOUT with 1-HS-4393, DG 1 CLR SSW RET VLV OPEN.

C. …STARTS with 1-HS-4393, DG 1 CLR SSW RET VLV stroking OPEN.

D. …is in PULLOUT with 1-HS-4393, DG 1 CLR SSW RET VLV stroking OPEN.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the SSW pump is normally left in AUTO during performance of ECA-0.0A, so that the pump will auto start when the bus is resupplied from the Diesel Generator. When power is restored to the bus from the APGs, the SSW pump is placed in PULL-OUT as the Component Cooling Water Pump is loaded on the bus prior to the SSW pump. 1-HS-4393 is a normally open valve which receives a confirmatory open signal from Safety Injection if closed. It is plausible to believe that the valve would have been closed at some point during the event to preclude pump run out when restarted. However, this valve is not closed and will have remained open.

B. Correct. SSW Pump 1-01 would have been placed in PULL-OUT during performance of SOP-614A prior to energizing the bus from the APGs. 1-HS-4393 would have remained open throughout the event.

C. Incorrect. Plausible because the SSW pump is normally left in AUTO during performance of ECA-0.0A, so that the pump will auto start when the bus is resupplied from the Diesel Generator. When power is restored to the bus from the APGs, the SSW pump is placed in PULLOUT as the Component Cooling Water Pump is loaded on the bus prior to the SSW pump. 1-HS-4393 is a normally open valve which receives a confirmatory open signal from Safety Injection if closed. It is plausible to believe that the valve would have been closed at some point during the event to preclude pump run out when restarted, thus the valve would currently be stroking.

D. Incorrect. Plausible because SSW Pump 1-01 would have been placed in PULL-OUT during performance of SOP-614A prior to energizing the bus from the APGs. 1-HS-4393 is a normally open valve which receives a confirmatory open signal from Safety Injection if closed. It is plausible to believe that the valve would have been closed at some point during the event to preclude pump run out when restarted, thus the valve would currently be stroking.

Technical Reference(s) SOP-614A, Step 5.1.C, 5.1.J
ECA-0.0A, Attachment 7, Step 27 Bases
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Loss of All Unit u Station Service Water in accordance with ABN-501, Station Service Water System Malfunction.

DISCUSS plant response, operator actions, and the reasons for the actions contained in ECA-0.0, Loss of All AC Power.

Question Source: Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: SOP-614A, Step 5.1.C	Revision: 13

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-614A
ALTERNATE POWER GENERATOR OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 10 OF 105

5.1 C. ENSURE the following Hand Switches are in PULLOUT OR as otherwise noted: **[Control Room Operator]**

- ☐ CS-1EG1, DG 1 BKR 1EG1
- ☐ CS-1EA1-1, INCOMING BKR 1EA1-1
- ☐ CS-1EA1-2, INCOMING BKR 1EA1-2
- ☐ 1-HS-2450A, MD AFWP 1
- ☐ 1/1-APCH1, CCP1
- ☐ 1/1-APPD, PDP (STOP)
- ☐ 1/1-PCPR1, PRZR BACKUP HTR GROUP A (OFF)
- ☐ 1/1-PCPR, PRZR CTRL HTR GROUP C (OFF)
- ☐ 1/1-APRH1, RHRP 1
- ☐ 1-HS-6700, RECIRC PMP 5
- ☐ 1-HS-4518A, CCWP 1
- ☐ 1-HS-5421 CRDM VENT FN 1
- ☐ 1-HS-5405A, CNTMT FN CLR FN 1
- ☐ 1-HS-5413A, CNTMT FN CLR FN 3
- ☐ 1-HS-4764, CSP 1
- ☐ 1-HS-4765, CSP 3
- ☐ 1/1-APSI1, SIP 1
- ☐ 1-HS-4250A, SSWP 1

Comments / Reference: SOP-614A, Step 5.1.J	Revision: 13

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-614A
ALTERNATE POWER GENERATOR OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 13 OF 105

5.1 J. The priorities are isolating any inventory loss from the RCS while restoring the following systems to service by starting the following components:

> NOTE: CCW Pump, SSW Pump AND Recirc Pump 5 will auto start when handswitch is positioned from PULLOUT to AUTO.

☐ 1) 1-HS-4518A, CCWP 1 (≈789 kW)

☐ 2) 1-HS-4250A, SSWP 1 (≈643 kW)

Comments / Reference: ECA-0.0A, Attachment 7, Step 27 Bases	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 80 OF 88

ATTACHMENT 7
PAGE 22 OF 30

BASES

Step 26: Following restoration of power to one AC safeguards bus, the operator should stabilize plant conditions while selecting the appropriate recovery procedure. If a steam generator depressurization is in progress when AC power is restored, Step directs the operator to stabilize steam generator pressures at the values existing when AC power is restored.

If local control of the SG atmospheric(s) has been established, SG pressures must be stabilized using local control, until automatic control is restored during subsequent actions (e.g., ECA-0.1A, Step 10). Stabilizing SG pressure can also be accomplished by placing the SG atmospheric(s) in auto, if air pressure is available and the handwheel is in the correct position locally.

SG pressure is controlled as directed by this step during subsequent procedure performance; therefore, this step is identified as a Continuous Action Step.

Step 27: The APGs may be used to restore power to a safeguard bus. When the APG is supplying the safeguard bus, then SOP-614A will be used to start the SSWP and CCWP. Due to the starting current required by the CCWP, the CCWP is loaded first and then the SSWP is started. If the operator arrives at this step before SOP-614A has directed starting the CCWP and SSWP, then the SSWP handswitch will be in PULLOUT. Verification of SSW flow is not required and ECA-TP-11-001A will start both of these pumps in the correct sequence.

If the operator arrives at this step after SOP-614A has started SSWPs, then automatic loading of the SSW pump should be verified. Flow verification to inservice equipment is critical to ensure cooling. Flow should be verified to ensure cooling flow to the diesel generator (to provide cooling for the diesel generator should it have been started as a result of local actions to restore AC emergency power) and other equipment necessary for recovery. If flow is not verified, the system should be aligned to establish required cooling. If SSW can not be aligned, the DG is stopped and the Operator is directed to return to Step 7. Some actions after Step 7 may have already been performed. However, when power is restored the Operator is directed to Step 26, thus in the event power was temporarily restored prior to completing all actions after Step 7, the Operator is directed to return to Step 7.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 3	Group	1	
	K/A	065 G 2.4.11	
Level of Difficulty: 3	Importance Rating	4.0	

Loss of Instrument Air: Emergency Procedures/Plan: Knowledge of abnormal condition procedures

Proposed Question: 15

Given the following conditions:

- Unit 1 is at 100% power.
- A major break in the Instrument Air System has occurred.
- ABN-301, Instrument Air System Malfunction is being implemented.
- All available Air Compressors are running.
- Instrument Air header pressure is 33 psig.

Which of the following describes the required actions for this condition in accordance with ABN-301?

A. Continue efforts to restore Instrument Air pressure and simultaneously initiate a Manual Turbine Runback per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction.

B. Trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection. Continue in ABN-301, Instrument Air System Malfunction, and take actions to control Charging flow locally.

C. Align the Unit Instrument Air Cross-tie to assist in pressure recovery. If loss of air operated valve control is observed, trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

D. Continue efforts to restore Instrument Air pressure per ABN-301, Instrument Air System Malfunction, and when pressure reaches 25 psig, trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because loss of Instrument Air has a significant impact on the secondary plant and it could be thought that rapidly reducing power would minimize the effects, however, ABN-301 requires a Unit trip when pressure reaches 35 psig.
B. Correct. ABN-301, Instrument Air Malfunction, requires a Unit trip when pressure reaches 35 psig. The EOP is entered but the actions of ABN-301, Instrument Air Malfunction are still performed which includes local control of Charging flow due to failed open valves.
C. Incorrect. Plausible if thought that the only trip criteria was loss of air operated valve control and that would be the proper actions until loss of control was observed, however, ABN-301 requires a Unit trip when pressure reaches 35 psig.
D. Incorrect. Plausible if thought that the pressure criteria for tripping were 25 psig and the actions to trip at that point would be correct.

Technical Reference(s) ABN-301, Steps 2.3.3, 2.3.5, & 2.3.7 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Instrument Air Compressor Trip or Header Pressure Low in accordance with ABN-301 Instrument Air System Malfunction.

Question Source: Bank ILOT8252
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ABN-301, Step 2.3.3	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 6 OF 122

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify Instrument Air Header Pressure - INCREASING OR STABLE.	Monitor Instrument Air Header Pressure continuously. a. IF header pressure continues to decrease, THEN perform the following: 1) Consult with opposite unit Control Room to crosstie Unit 1 and Unit 2 Instrument Air Headers per SOP-509A.

Comments / Reference: ABN-301, Step 2.3.5	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 9 OF 122

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: • Equipment controlled by instrument air will commence to fluctuate or drift to its failed position when instrument air pressure decreases to a range of 35 psig to 45 psig.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5 Check status of Instrument Air:	Perform the following:
☐ a. Verify instrument air malfunction - REPAIRED OR ISOLATED	IF in MODE 1, 2, 3, OR 4 AND Instrument Air Header pressure decreases to 35 psig OR control of system(s) is lost, THEN manually trip the reactor AND GO TO EOP-0.0A/B while other operator(s) continue this procedure.
☐ b. Verify Instrument Air Header pressure - GREATER THAN 45 psig AND INCREASING	
☐ c. GO TO Section 3.0, this procedure.	IF RHR operation is affected during this procedure, THEN perform ABN-104 while continuing this procedure.

Comments / Reference: ABN-301, Step 2.3.7	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 11 OF 122

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7 Establish manual control of RCS Charging.	
[C] ☐ a. Open RWST to charging pump suction: • 1/u-LCV-112D, RWST TO CHRG PMP SUCT VLV • 1/u-LCV-112E, RWST TO CHRG PMP SUCT VLV	a. Locally open valves: (AB 810' X-207) • u-LCV-0112D, RWST u-01 TO CHRG PMP SUCT VLV 0112D • u-LCV-0112E, RWST u-01 TO CHRG PMP SUCT VLV 0112E
[C] ☐ b. Close VCT to charging pump suction: • 1/u-LCV-112B, VCT TO CHRG PMP SUCT VLV • 1/u-LCV-112C, VCT TO CHRG PMP SUCT VLV	b. Locally close valves: [AB 810' X-203(X-202)] • u-LCV-0112B, VCT u-01 TO CHRG PMP UPSTRM LVL CTRL VLV 0112B • u-LCV-0112C, VCT u-01 TO CHRG PMP DNSTRM LVL CTRL VLV 0112C

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	1	
	K/A	W/E11 G 2.4.20	
Level of Difficulty: 3	Importance Rating	3.8	

Loss of Emergency Coolant Recirculation: Emergency Procedures/Plan: Knowledge of the operational implications of EOP warnings, cautions, and notes

Proposed Question: 16

Given the following conditions:

- Unit 2 has experienced a Loss of Coolant Accident.
- During performance of EOP-1.0B, Loss of Reactor or Secondary Coolant, the crew determined that Residual Heat Removal Pump 2-01 was NOT available and that Bus 2EA2 was de-energized.
- The Unit Supervisor transitioned to ECA-1.1B, Loss of Emergency Coolant Recirculation.
- Prior to performing Step 3 to RESET Safety Injection, the Unit Supervisor read the following CAUTION:
    - “If offsite power is lost after SI reset, manual action may be required to restore safeguards equipment to desired status.”
- RWST level is currently 43% and lowering.
- Containment pressure is 12 psig and lowering.

While performing ECA-1.1B, Step 10, “Determine Containment Spray Requirements (Suction From RWST)”, Offsite Power is lost to 2EA1 and the bus is re-energized by Diesel Generator 2-01.

The primary operational concern, in accordance with ECA-1.1B, Loss of Emergency Coolant Recirculation is that…

A. …Centrifugal Charging Pump 2-01 must be manually re-started.

B. …Containment Spray Pumps 2-01 and 2-03 would NOT restart thereby challenging the Containment Critical Safety Function.

C. …Safety Injection Pump 2-01 would NOT restart to provide core cooling.

D. …Containment Isolation will be lost as Phase A Isolation Valves reposition.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because a concern is always that cooling to the RCP seals is occurring. If the Centrifugal Charging Pump did not restart, manually restarting would be required. However, the CCP receives an auto start from the BOS.
B. Incorrect. Plausible because the Containment Spray Pumps would not restart after Step 6 is performed and Containment has not reached the pressure which would normally allow all Containment Spray flow to be stopped (3.0 psig). However, in ECA-1.1B, Step 10, all Containment Spray Pumps are stopped when Containment pressure is less than 18.0 psig to conserve RWST water.
C. Correct. The only ECCS Pumps available are Centrifugal Charging Pump 2-01 which would restart and Safety Injection Pump 2-01 which would need to be manually started.
D. Incorrect. Plausible because Containment Isolation Phase A would have been RESET in Step 5, but the valves should not reposition as a result of the Blackout Sequencer operation.

Technical Reference(s): EOS-1.3B, Step 5, CAUTION
EOS-1.3B, Attachment 3, Step 5 CAUTION
EOP-0.0B, Attachment 9, Item 10
ECA-1.1B, Step 3 CAUTION
ECA-1.1B, Attachment 7, Step 3 CAUTION
ECA-1.1B, Attachment 7, Step 9 Bases
ECA-1.1B, Step 10

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in ECA-1.1, Transfer to Cold Leg Recirculation.

Question Source: Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: EOS-1.3B, Step 5, CAUTION	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-1.3B
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 10 OF 54

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: ECCS recirculation flow to RCS must be maintained at all times.

CAUTION: If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment (CCP will be running with no suction).

5 Perform The Following To Complete Recirculation Alignment:

Comments / Reference: EOS-1.3B, Attachment 3, Step 5 CAUTION	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-1.3B
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 42 OF 54

ATTACHMENT 3
PAGE 6 OF 18

BASES

The symbol [1H] has been utilized to identify that Attachment 1.H exists, which allows the actions for aligning Containment Spray for cold recirculation to be delegated to a Reactor Operator by handing off the attachment. Since the action involves multiple specific actions to accomplish this evolution, having the RO perform the evolution using the attachment in a step-wise manner may benefit the overall ERG performance (e.g., minimize communications, permit SRO directing response and recovery activities to maintain higher level view of effort, provide termination criteria to RO in a written format).

CAUTION: The operator should ensure that flow is being maintained to the RCS so that core cooling is maintained. Maintaining core cooling will minimize or prevent fuel damage.

Operator is expected to restart secured ECCS & Containment Spray pumps when adequate suction source has been reestablished.

CAUTION: With the SI signal reset, no further automatic signal will be generated to restart safeguards equipment. Normal sequencing of safeguards loads onto the safeguards bus after diesel-generator startup will not occur. However, a "blackout" sequencer actuation is possible. This may result in a charging pump restarting without an adequate suction source since the RHR pump will not be automatically restarted. It should be noted that the RHR pumps should be started before the SI pumps to provide a suction source for them.

Comments / Reference: EOP-0.0B, Attachment 9, Item 10	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-0.0B
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 73 OF 117

ATTACHMENT 9
PAGE 14 OF 20

POST EVENT SYSTEM REALIGNMENT

7. IF cooldown of the RCS is to be performed, THEN perform the following:

☐ a. Cooldown rates should be logged per OPT-407, RCS TEMPERATURE AND PRESSURE VERIFICATION.

☐ b. Place the PROTECTION SWITCH on RCP breaker cubicles to "COLD LOOP" position. (locally)

☐

8. Shutdown unnecessary plant equipment per IPO-009B, PLANT EQUIPMENT SHUTDOWN FOLLOWING A TRIP while continuing with this procedure.

> NOTE: The Containment Hydrogen Microprocessor should be calibrated within 24 hours following a LOCA and every 30 days thereafter.

☐

9. IF a LOCA has occurred, THEN notify I & C to calibrate the Containment Hydrogen Microprocessors.

10. WHEN Containment pressure is less than 3.0 psig, THEN place the Containment Spray System in standby by performing the following:

☐ a. Stop all four containment spray pumps.

☐ b. Ensure Containment Spray Heat Exchanger Outlet Valves are closed.

☐ c. Containment Spray System aligned in standby as required, per SOP-204B, CONTAINMENT SPRAY SYSTEM.

Comments / Reference: ECA-1.1B, Step 3 CAUTION	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-1.1B
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 3 OF 79

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: If emergency coolant recirculation capability is restored during this procedure, further recovery actions should continue by returning to procedure and step in effect.

CAUTION: If suction source is lost to any ECCS or Containment Spray pump, the pump should be stopped and the Plant Staff should be notified of the condition.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check If Emergency Coolant Recirculation Equipment - AVAILABLE PER ATTACHMENT 2.	Restore at least one train.
2	IF The Diesels Are Running, THEN Place Both DG EMER STOP/START Handswitches in START	

CAUTION: If offsite power is lost after SI reset, manual action may be required to restore safeguards equipment to desired status.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Reset SI If Necessary.	Reset SI per EOP-0.0B, REACTOR TRIP OR SAFETY INJECTION, Attachment 9.

Comments / Reference: ECA-1.1B, Attachment 7, Step 3 CAUTION Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-1.1B
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 54 OF 79

ATTACHMENT 7
PAGE 2 OF 27

BASES

SI pumps and CCPs may experience permanent damage within a short time of loss of suction. Operators should immediately stop these pumps if they observe symptoms of loss of suction. RHR and Containment Spray pumps can withstand effects of cavitation for a longer time, but not indefinitely without pump damage. If operators observe indication of cavitation on a RHR pump, action should be taken to stop the SI pump and CCP taking suction from the RHR pump discharge. The reason for stopping SI pump and CCP first is twofold: 1) Protect the vulnerable SI pump or CCP from damage, and 2) Stopping the SI pump and CCP reduces total flow through the RHR pump, which improves NPSH conditions for the RHR pump. The Plant Staff is notified so that potential compensatory actions for responding to a degraded sump can be evaluated.

STEP 1: Failures or unavailability of the RHR pumps and recirculation sump isolation valves are the most common reason for loss of ECR capability. This step instructs the operator to attempt to restore the equipment needed for emergency coolant recirculation. Equipment can be restored by various methods, such as racking in pump breakers, closing breakers for motor operated valves, local operation of valves or equipment repair.

This step is not pertinent for recovering from only a LOCA outside containment event since emergency coolant recirculation equipment would be available. However, this step should still be performed.

STEP 2: To ensure the diesel generators remain available in an emergency mode the operator is instructed to insert a manual emergency start signal prior to resetting the automatic emergency start from the SI signal if the diesels are running.

CAUTION: With the SI signal reset, no further automatic signal will be generated to restart safeguards equipment. Normal sequencing of safeguards loads onto the Safeguards bus after diesel-generator startup will not occur. However, a "blackout" sequencer actuation is possible.

Comments / Reference: ECA-1.1B, Attachment 7, Step 9 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-1.1B
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 56 OF 79

ATTACHMENT 7
PAGE 4 OF 27

BASES

The intent of this step is to provide containment heat removal capability by using containment fan coolers when conditions inside containment are NOT adverse (i.e., LOCA outside containment). The containment fan coolers and its associated ventilation chilled water cooling are not qualified for post accident operation. High temperature conditions inside containment can cause flashing/waterhammer to occur and chilled water to containment should not be aligned under these conditions, and consequently, the containment fan coolers should not be started under these conditions. A check that containment pressure has remained less than 5 psig ensures that the chilled water inside the containment fan coolers cooling supply will not be under saturated conditions when cooling water is realigned. If containment pressure increased above 5 psig to reaching this step, the containment fan coolers cooling supply may be under saturated conditions and the Plant Staff is notified to evaluate the potential for the cooling water flashing to steam, and the subsequent waterhammer that may be experienced if cooling flow is realigned.

A check of the alarm for the Seal Water Heat Exchanger is to verify flow in the Non-Safeguards loop of the CCW system to ensure cooling flow is available to the HVAC Centrifugal Water Chillers. If this component is not available, any indication of flow in the Non-Safeguards loop is sufficient to satisfy the substep.

For the reset of the Containment isolation signals, this part of the automatic logic requires a deliberate operator action to remove the "close" signal. No valve will reposition upon actuation of the resets, but subsequent control actions will open the valves. These valves should remain closed, unless necessary process streams are being established, until the cause of the SI is determined or corrected.

The maximum CCW pump flow is 17,500 GPM to prevent pump runout. If the CCW pump runs out during performance of this alignment, non-essential CCW loads will need to be isolated to prevent CCW pump damage.

STEP 9: If the RWST is not empty, the operator proceeds with Steps 10 through 31, which are concerned with minimizing the RWST outflow and, therefore, extending time that fluid for core cooling is providing by the RWST. This is accomplished by stopping the containment spray pumps and decreasing the ECCS pumps flow rates. However, if the RWST is empty, the operator is instructed to skip to Step 32 to stop pumps taking suction from the empty RWST.

Comments / Reference: ECA-1.1B, Step 10	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-1.1B
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 5 OF 79

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Determine Containment Spray Requirements (Suction From RWST):	
	a. Containment spray pump suction - ALIGNED TO RWST	a. IF containment spray pump suction aligned to sump, THEN go to Step 12.
	b. Determine number of containment spray pumps required from Table 1.	

TABLE 1

RWST LEVEL	CONTAINMENT PRESSURE	SPRAY PUMPS REQUIRED
GREATER THAN RWST EMPTY	GREATER THAN 50 PSIG	4
	BETWEEN 18.0 PSIG AND 50 PSIG	2
	LESS THAN 18.0 PSIG	0
LESS THAN RWST EMPTY	-	0

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	c. Containment spray pumps running - EQUAL TO NUMBER REQUIRED	c. Manually operate containment spray pumps as necessary.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier	1	
Change: 3	Group	1	
	K/A	W/E05 EK1.3	
Level of Difficulty: 4	Importance Rating	3.9	

Loss of Secondary Heat Sink: Knowledge of the operational implications of the following concepts as they apply to the Loss of Secondary Heat Sink: Annunciators and conditions indicating signals, and remedial actions associated with the Loss of Secondary Heat Sink

Proposed Question: 17

Given the following conditions:

- FRH-0.1A, Response to Loss of Secondary Heat Sink, is in progress on Unit 1.
- Both Centrifugal Charging Pumps are available.
- All Reactor Coolant Pumps were stopped when Auxiliary Feedwater could NOT be established to at least one Steam Generator (SG).
- Efforts are in progress to establish Main Feedwater to at least one SG.
- Containment Pressure is 0 psig and stable.
- In accordance with FRH-0.1A, Attachment 2, all SG Wide Range Levels are between 38% and 40%.
- Pressurizer pressure has steadily risen over the last two minutes and both Pressurizer Power Operated Relief Valves have opened.
- Reactor Coolant loop temperature differential between hot and cold legs is less than 5 ºF on all loops.

Which of the following indicates the actions required in accordance with FRH-0.1A?

A. Initiate Reactor Coolant System (RCS) bleed and feed.

B. Depressurize at least one SG to less than 500 psig.

C. Depressurize the RCS to less than 1910 psig.

D. Depressurize at least one SG to atmospheric pressure.

Proposed Answer: A

Explanation:

A. Correct. In accordance with FRH-0.1A, an alternate indication of SG inventory reducing to critical levels where bleed and feed must be initiated is based on RCS pressure of 2335 psig. With both PORVs opening, an RCS pressure of at least 2335 is indicated.

B. Incorrect. Plausible because if Main Feedwater cannot be established, FRH-0.1A Step 9 requires that one SG be depressurized to less than 500 psig in order to feed the SG with condensate.

C. Incorrect. Plausible because if Main Feedwater cannot be established, FRH-0.1A Step 9 requires that the RCS be depressurized to 1910 in order to block Safety Injection and proceed with attempts to feed one SG with condensate.

D. Incorrect. Plausible because FRH-0.1A, Step 21 RNO has the operator depressurize at least one SG to atmospheric pressure to inject a low pressure water source if there is not an adequate RCS bleed path.

Technical Reference(s): FRH-0.1A, Steps 2, 3, 9, 11, 12 & 21 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1, Response to Loss of Secondary Heat Sink.

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: FRH-0.1A, Steps 2 & 3	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 4 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Check CCP Status - BOTH AVAILABLE	Immediately perform the following: a. STOP ALL RCPs. b. Verify power to PRZR PORV block valves - AVAILABLE Locally restore power to block valve(s). c. Go to Step 12. OBSERVE CAUTION PRIOR TO STEP 12.
* 3	Check Bleed And Feed - REQUIRED:	
	a. Check the following: • Actual wide range level (per Attachment 2) in at least 3 SGs - LESS THAN 35% (40% FOR ADVERSE CONTAINMENT) -OR- • PRZR pressure - GREATER THAN OR EQUAL TO 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK	a. Go to Step 4.
	b. Trip all RCPs.	
	c. Verify power to PRZR PORV block valves - AVAILABLE	c. Locally restore power to block valve(s).
	d. Go to Step 12 AND perform Steps 12 through 21 without delay.	

Comments / Reference: FRH-0.1A, Step 9	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 13 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Following block of automatic SI actuation, manual SI actuation may be required if conditions degrade.

NOTE: After the low steamline pressure SI signal is blocked, main steam isolation will occur if the high steam pressure rate setpoint is exceeded.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	Establish Feed Flow From Condensate System:	
	a. Depressurize RCS to less than 1910 psig:	
	1) Turn off all PRZR heaters.	
	2) Check letdown - IN SERVICE	2) Use one PRZR PORV. **IF NOT**, **THEN** use auxiliary spray. Go to Step 9b.
	3) Use auxiliary spray.	3) Use one PRZR PORV.
	b. Block SI signals: • Low steamline pressure SI • Low PRZR pressure SI	
	c. Depressurize at least one SG to less than 500 psig:	

Comments / Reference: FRH-0.1A, Steps 11 & 12	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 17 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Check SG Levels:	
	a. Narrow range level in at least one SG - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)	a. IF feed flow to at least one SG verified, THEN maintain flow to restore narrow range level to greater than 43% (50% FOR ADVERSE CONTAINMENT). IF NOT verified, THEN go to Step 11.
	b. Return to procedure and step in effect.	
11	Check Bleed And Feed - REQUIRED	
	a. Check the following: • Actual wide range level (per Attachment 2) in at least 3 SGs - LESS THAN 35% (40% FOR ADVERSE CONTAINMENT) -OR- • PRZR pressure - GREATER THAN OR EQUAL TO 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK	a. Return to Step 1.

CAUTION: Steps 12 through 21 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

12	Actuate SI.	

Comments / Reference: FRH-0.1A, Step 21	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 19 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1D] 19	Establish Instrument Air And Nitrogen To Containment:	
	a. Establish instrument air:	
	1) Verify air compressor running. -AND-	1) Manually start air compressor and align valve as appropriate.
	2) Establish instrument air to containment.	
	b. Establish nitrogen:	
	1) Verify ACCUM 1•4 VENT CTRL, 1-HC-943 - CLOSED	1) Manually close valve.
	2) Open SI/PORV ACCUM N_2 ISOL VLV, 1/1-8880	
20	Establish RCS Bleed Path:	
	a. Verify power to PRZR PORV block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. Verify PRZR PORV block valves - BOTH OPEN	b. Manually open both block valve(s).
	c. Open PRZR PORVs.	
21	Verify Adequate RCS Bleed Path:	Perform the following:
	• PRZR PORVs - BOTH OPEN	a. Open vents on reactor vessel head and on the PRZR to containment.
	• PRZR PORV block valves- BOTH OPEN	b. Align any available low pressure water source to the SG(s). IF no low pressure water source can be aligned, THEN go to Step 22.
		c. Depressurize at least one intact SG to atmospheric pressure using SG atmospheric to inject low pressure water source.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/25/2014	Tier	1	
Change: 3	Group	1	
	K/A	077 AK3.01	
Level of Difficulty: 4	Importance Rating	3.9	

Generator Voltage and Electric Grid Disturbances: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Reactor and turbine trip criteria

Proposed Question: 18

Given the following conditions:

- Unit 1 is at 45% Reactor Power.
- Generator Load Target is 530 MWe.
- North Texas is experiencing a winter ice storm.
- Grid Frequency has just lowered from 58.7 Hz to 58.3 Hz.

Which of the following describes the required actions in accordance with ABN-601, Response to a 138/345 KV System Malfunction?

If grid frequency does NOT increase to greater than 58.4 Hz in…

A. …2 seconds, trip the turbine.

B. …30 seconds, trip the turbine.

C. …2 seconds, trip the reactor.

D. …30 seconds, trip the reactor.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because if the grid frequency were to lower to less than or equal to 58.0 Hz then 2 seconds would be allowed prior to tripping the reactor. Tripping the turbine is plausible as ABN-402, Main Generator Malfunction has four sections which require a Turbine Trip rather than a Reactor Trip when reactor power is less than 50%.

B. Incorrect. Plausible because the action listed is the correct value and timing. Tripping the turbine is plausible as ABN-402, Main Generator Malfunction has four sections which require a Turbine Trip rather than a Reactor Trip when reactor power is less than 50%.

C. Incorrect. Plausible because if the grid frequency were to lower to less than or equal to 58.0 Hz then 2 seconds would be allowed prior to tripping the reactor. .

D. Correct. In accordance with ABN-601 Step 9.3, with grid frequency between 58.0 Hz and 58.4 Hz a delay of 30 seconds is allowed for frequency recovery prior to tripping the reactor.

Technical Reference(s) ABN-402, Section 2, 3, 6 & 7 Attached w/ Revision: See Comments / Reference

ABN-601, Section 9.3

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of Main Turbine and its support systems.

Question Source: Bank

Modified Bank (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments / Reference: ABN-601, Step 9.3.3	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 124 OF 229

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Generator load will decrease with decreasing frequency (due to reduction in generator efficiency) and automatic load control will restore the load to the set load. This will result in increasing steam flow and increased reactor power. Maintain reactor power less than 100%.
- Steps 1 through 3 should be considered continuous action steps during periods of grid instability.

☐ 1 Maintain Reactor Power - LESS THAN OR EQUAL TO 100%.

☐ 2 Verify QSE Generation Controller communications - AVAILABLE

Control frequency as necessary per Attachment 23

CAUTION: When CPNPP trips both Reactors, it is highly probable that the grid will be lost and a loss of all offsite power will occur.

☐ 3 Perform the following as appropriate.

FREQUENCY	ACTION
>60.6 Hz (1818 rpm)	• Maintain contact with QSE • Stabilize plant power for load reduction • If immediate recovery is not evident after 9 minutes, coordinate with QSE to trip the reactor, if necessary and GO TO EOP 0.0A/B.
≤57.5 Hz (1725 rpm)	**IMMEDIATELY** Trip reactor and GO TO EOP.0.0A/B
≤58.0 Hz (1740 rpm)	**AFTER 2 sec** Trip reactor and GO TO EOP.0.0A/B
≤58.4 Hz (1752 rpm)	**AFTER 30 sec** Trip reactor and GO TO EOP.0.0A/B
≤59.4 Hz (1782 rpm)	**GO TO STEP 4**
>59.4 Hz (1782 rpm)	Continuous operation allowed.

Comments / Reference: ABN-402, Step 2.3.5 RNO	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-402
MAIN GENERATOR MALFUNCTION	REVISION NO. 9	PAGE 5 OF 83

2.3 Operator Actions

		ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		CAUTION: Manual voltage control is provided for emergency operation only. While in MANUAL the generator will not respond to changing load characteristics, therefore frequent operator attention is necessary.	
☐	4	Verify voltage regulator transferred to - MANUAL • On the TG Control Display in the "Voltage Control" Section, "Manual" Bar is red and Auto/Man Subloop Controller is Green.	IF necessary for generator control and stability, THEN perform the following: a. Manually transfer voltage regulator as follows: 1) On the TG Control Display verify Exciter Current Target is matched with Exciter Amps 2) In the "Voltage Control" Section shift Voltage Control to manual using the Auto/Man Subloop Controller. (manual is green)
☐	5	Verify main generator capability limits using the "Gen Capability Curve" Display - NOT EXCEEDED	Perform the following: a. Restore main generator parameters to within limits. b. IF main generator can NOT be restored to within limits, THEN perform the following: 1) IF Reactor Power is greater than 50% (P-9), THEN manually trip the reactor AND GO TO EOP-0.0A/B. 2) IF Reactor Power is less than 50% (P-9), THEN trip the turbine AND GO TO ABN-403.

Comments / Reference: ABN-402, Step 3.3.1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-402
MAIN GENERATOR MALFUNCTION	REVISION NO. 9	PAGE 10 OF 83

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- This ABN gives guidance if specific parameters are exceeded or local actions are unsuccessful in restoring system to normal.
- System recovery and alarm response will be performed locally using ALM-1201 and SOP-407A/B.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Check generator hydrogen pressure >45 psi. • Control Room: P2800A / u-PI-6557 OR • Dispatch an Operator to the TB-778 Gas Rack to monitor local indicator: u-ST11P505	Perform the following: 1) IF Reactor Power is greater than or equal to 50%, THEN trip the Reactor AND GO TO EOP-0.0A/B while other qualified operator(s) continue with this procedure, Section 3 & 4. 2) IF Reactor Power is less than 50%, THEN trip the turbine AND perform ABN-403 while continuing with this procedure, Section 3 & 4.

Comments / Reference: ABN-402, Step 6.3.5	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-402
MAIN GENERATOR MALFUNCTION	REVISION NO. 9	PAGE 25 OF 83

6.3 Operator Actions

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5	Drain the liquid level detector(s) per Attachment 3.	IF in-leakage is faster than can be drained, THEN perform the following: • IF Reactor Power is greater than 50% (P-9), THEN manually trip the Reactor AND GO TO EOP-0.0A/B. • IF Reactor Power is less than or equal to 50% (P-9), THEN trip the Turbine AND GO TO ABN-403.

Comments / Reference: ABN-402, Step 7.3.7	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-402
MAIN GENERATOR MALFUNCTION	REVISION NO. 9	PAGE 35 OF 83

7.3 Operator Actions

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6		c. IF H_2 Cold Gas temperature is greater than or equal to 55°C (131°F), THEN perform following: • Initiate a load reduction per IPO-003A/B until temperatures within limits. OR • IF control of temperature can not be re-established immediately, THEN trip the Turbine per Shift Manager.
☐ 7	On the Gen Temp/Leak Water Display in the Primary Water TCV Section, verify PW Supply header temperature less than 140°F.	IF Reactor Power is greater than 50% (P-9), THEN trip the Reactor AND GO TO EOP-0.0A/B. IF Reactor Power is less than or equal to 50% (P-9), THEN trip the Turbine AND GO TO ABN-403.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	2	
	K/A	001 AA2.03	
Level of Difficulty: 3	Importance Rating	4.5	

Continuous Rod Withdrawal: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place

Proposed Question: 19

Given the following conditions:

- Unit 1 is performing a Reactor Startup per IPO-002A, Plant Startup from Hot Standby twelve hours after a Reactor Trip.
- During the last rod withdrawal when the operator released 1/1-FLRM, CONTROL ROD MOTION CTRL the control rods continued to withdraw.
- The operator identified that the continuous rod withdrawal was a result of inadvertently placing 1/1-RBSS, CONTROL ROD BANK SELECT in AUTO and has now placed 1/1-RBSS in MAN and the control rods stopped moving.
- Current indications are as follows:
  - 1-NI-31B, SR COUNT RATE CHAN I = 1.2 x 10^{5} CPS and increasing.
  - 1-NI-32B, SR COUNT RATE CHAN II = 8.4 x 10^{4} CPS and increasing.
  - 1-NI-35B, IR CURRENT CHAN I = 9 x 10^{-10} amps and increasing.
  - 1-NI-36B, IR CURRENT CHAN II = 1.1 x 10^{-10} amps and increasing.

Which of the following actions are required following the inadvertent Continuous Rod Withdrawal?

A. Place 1/1-RTC, RX TRIP BKR in TRIP and perform the immediate actions of EOP-0.0A, Reactor Trip or Safety Injection.

B. Level Reactor Power at approximately 1 x 10^{-8} amps by adjusting the control rods as necessary to establish a 0 decade per minute startup rate.

C. Place 1/1-FLRM, CONTROL ROD MOTION CTRL in IN and drive rods to the desired control bank position for the 1/M plot data collection.

D. When 1-PCIP, Window 2.5 – SR RX TRIP BLK PERM P-6 is ON, place both SR RX TRIP RESET/BLK switches in BLOCK.

Proposed Answer: A

Explanation:

A. Correct. The operator should recognize that the Reactor Trip logic of 1/2 Source Range channels greater than 10^5 CPS is satisfied and trip the Reactor.

B. Incorrect. Plausible because the startup concludes with the operator leveling Reactor Power at 1 x 10^{-8} amps. If the operator did not recognize by the indications that a Reactor Trip should have already occurred, this could be expected.

C. Incorrect. Plausible because the operator could believe that returning to the desired rod withdrawal endpoint is conservative as operator error introduced the initial problem.

D. Incorrect. Plausible because performing this action would be normal once by Intermediate Range Channels were greater than 1 x 10^{-10} amps and the P-6 permissive was received, therefore, if the operator did not recognize that the Source Range Reactor Trip should have already occurred, this could be expected.

Technical Reference(s) IPO-002A, Step 5.2.22 & 5.2.23 Attached w/ Revision: See Comments / Reference
EOP-0.0A, Section C.1
ALM-0065A, 1-PCIP, Window 2.5

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the symptoms or entry conditions for EOP-0.0, Reactor Trip or Safety Injection.

Question Source: Bank
Modified Bank (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6
55.43

Comments / Reference: IPO-002A, Step 5.2.22 & 5.2.23	Revision: 20

CPSES INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-002A
PLANT STARTUP FROM HOT STANDBY	REVISION NO. 20	PAGE 38 OF 85

NOTE: The Source Range and Intermediate Range overlap region should be passed through quickly with a steady startup rate of approximately 0.5 DPM. Expeditiously proceeding through this region should prevent inadvertent Source Range Reactor Trip caused by Intermediate Range perturbations around the P-6 setpoint.

5.2.20 Establish a startup rate of approximately 0.5 DPM.

/
Initials Date

NOTE: The minimum required overlap between the Source Range and Intermediate Range channels is ONE decade.

5.2.21 Verify the Intermediate Range channels begin to respond when the Source Range channels are between 10^3 cps and 10^4 cps.

/
Initials Date

CAUTION: There is only approximately ½ decade of Source Range counts between the P-6 interlock setpoint and the Source Range Reactor Trip setpoint.

5.2.22 WHEN 1-PCIP, 2.5, SR RX TRIP BLK PERM P-6 is ON, THEN perform the following:

A. Place both SR RX TRIP RESET/BLK switches in BLOCK:

- ☐ • 1/1-N-33A, SR RX TRIP RESET/BLK
- ☐ • 1/1-N-33B, SR RX TRIP RESET/BLK

/
Initials Date

B. Verify the following are ON:

- ☐ • 1-PCIP, 1.1, SR TRN A RX TRIP BLK
- ☐ • 1-PCIP, 2.1, SR TRN B RX TRIP BLK
- ☐ • 1-ALB-6D, 1.1, SR HI VOLT FAIL
- ☐ • 1-TSLB-9, 1.6, IR SR BLK PERM NC-35D
- ☐ • 1-TSLB-9, 2.6, IR SR BLK PERM NC-36D

/
Initials Date

5.2.23 Level Reactor power at approximately 1 X 10^{-8} amps on the highest reading Intermediate Range channel by adjusting the control rods as necessary to establish a 0 DPM startup rate.

/
Initials Date

Comments / Reference: EOP-0.0A, Section C.1	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 2 OF 117

A. PURPOSE

This procedure provides actions to verify proper response of automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery procedure.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2 and 3 GREATER THAN OR EQUAL TO 1000 PSIG. Using this procedure when not in these modes requires a step by step evaluation to determine if the required action is still applicable to current plant conditions.

C. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip:

- 2/4 Neutron Flux power ranges greater than 109%
- 2/4 Neutron Flux power ranges greater than 25% (Below P-10 permissive)
- 2/4 Neutron Flux rate trip lights as indicated on NIS cabinets (POSITIVE RATE TRIP)
- 1/2 Neutron Flux source ranges greater than 10^5 CPS (Below P-6 permissive)

Comments / Reference: ALM-0065A, 1-PCIP, Window 2.5	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 31 OF 73

ANNUNCIATOR NOM./NO.: **SR RX TRIP BLK PERM P-6** **2.5**

PROBABLE CAUSE:

Reactor shutdown

NOTE: This window is normally illuminated when reactor power is $>10^{-10}$ amps.

AUTOMATIC ACTIONS:

Provides backup block to source range flux doubling actuation
P-6 permits manual block of source range reactor trip

NOTE: When both intermediate range detectors indicates $\leq 10^{-10}$ amps, the source range detectors will automatically reenergize and source range flux doubling is reenabled.

OPERATOR ACTIONS:

None

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 3	Group	2	
	K/A	003 AK1.03	
Level of Difficulty: 2	Importance Rating	3.5	

Dropped Control Rod: Knowledge of the operational implications of the following concepts as they apply to the Dropped Control Rod: Relationship of reactivity and reactor power to rod movement

Proposed Question: 20

Given the following conditions:

- Unit 1 is operating at 20% power with all systems in automatic.
- Control Bank D rods are at 120 steps.
- Control Bank C rod K-6 drops to the bottom of the core.
- NO Rod Control Urgent Failure alarms occur.

Where will thermal power and Reactor Coolant System (RCS) T_{AVE} stabilize in response to the dropped rod?

Reactor thermal power will be…

A. …approximately the same as prior to the dropped rod; RCS T_{AVE} will be within 1ºF of the temperature prior to the dropped rod.

B. …approximately the same as prior to the dropped rod; RCS T_{AVE} will be more than 5ºF lower than the temperature prior to the dropped rod.

C. …several percent lower than prior to the dropped rod; RCS T_{AVE} will be more than 5ºF lower than the temperature prior to the dropped rod.

D. …several percent lower than prior to the dropped rod; RCS T_{AVE} will be within 1ºF of the temperature prior to the dropped rod.

Proposed Answer: A

Explanation:

A. Correct. Power will initially decrease due to the dropped rod. As power decreases, temperature will decrease. As temperature decreases, positive reactivity is added to restore power. Bank D rods in auto will cause rods to step out. Rods stepping out will restore power and temperature to the original value.
B. Incorrect. Plausible since power will be restored due to the previous decrease in temperature, but rods will step out to restore temperature.
C. Incorrect. Plausible since power and temperature will initially decrease on the dropped rod, but power will be restored by the decreasing temperature and rods will step out to restore temperature.
D. Incorrect. Plausible since rods will step out to restore temperature, but the decreased temperature adds positive reactivity to restore power.

Technical Reference(s) ABN-712, Section 3.2 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** overall operation of the Rod Control and Digital Rod Position Indication systems.

COMPREHEND the normal, abnormal and emergency operation of the Rod Control System.

Question Source: Bank ILOT6393
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 1
55.43

Comments / Reference: ABN-712, Section 3.2	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 9 OF 52

3.0 DROPPED OR MISALIGNED ROD IN MODE 1 OR 2

3.1 Symptoms

a. Annunciator Alarms

- PR CHAN DEV (6D-3.4)
- DRPI ROD DEV (6D-3.5)
- ANY ROD AT BOT (6D-3.7)
- ≥2 ROD AT BOT (6D-4.7)
- QUADRANT PWR TILT (6D-4.10)

b. Plant Indications

- Plant parameters changing abnormally during rod position changes

NOTE:
- A dropped rod will distort the symmetrical flux distribution of the reactor core. This distortion will be reflected as a deviation in the power range and N16 indications monitored by OPT-102A/B (SR 3.3.1.1.2.a; 3.3.1.1.2.b.;3.3.1.1.6; 3.3.1.1.7). The power range and N16 instrumentation need not be declared inoperable if indications were within the required deviation prior to the event and no other influence has occurred. (SMF-2007-003427)
- For the 12 hour shifty surveillance while in the abnormal condition of a dropped rod, an assessment should be performed that the channels are indicating as expected for the condition of an asymmetrical flux pattern. Since the dropped rod may cause the channels to deviate beyond the normal Channel Check criteria, an assessment is required that the channels are as expected for the plant condition. If required, additional resources (e.g. Core Performance Engineering) may be consulted to assist with the assessment. (SMF-2007-003427)

- NIS Power Range instruments power or AFD indications disagree
- DRPI Rod Bottom Light(s) lit for rods which should be withdrawn
- DRPIs in a bank disagree by greater than 12 steps
- DRPI disagrees with its group step counter by greater than 12 steps

3.2 Automatic Actions

- Possible Reactor trip
- Automatic control rod motion

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	2	
	K/A	024 AK3.02	
Level of Difficulty: 3	Importance Rating	4.2	

Emergency Boration: Knowledge of the reasons for the following responses as they apply to the emergency Boration: Actions contained in EOP for emergency boration

Proposed Question: 21

Given the following conditions:

- Unit 2 just tripped from 100% power.
- The Unit was two weeks from a scheduled Refueling Outage.
- Three Control Rods do NOT indicate fully inserted.
- An Emergency Boration is commenced.

In accordance with the EOP-0.0B, Reactor Trip or Safety Injection Foldout Page, which of the following describes the required amount of boric acid to be injected and the reason for the boration?

A. 3600 gallons of boric acid. Accounts for the maximum reactivity worth of the additional two rods not assumed to stick out to ensure proper shutdown margin on the most limiting accident which is the Main Steam Line Break.

B. 3600 gallons of Boric Acid. Accounts for the maximum reactivity worth of the additional two rods not assumed to stick out to ensure proper shutdown margin on the most limiting accident which is the Large Break Loss of Coolant Accident.

C. 5400 gallons of Boric Acid. Accounts for the maximum reactivity worth of each rod stuck out to ensure proper shutdown margin on the most limiting accident which is the Main Steam Line Break.

D. 5400 gallons of Boric Acid. Accounts for the maximum reactivity worth of each rod stuck out to ensure proper shutdown margin on the most limiting accident which is the Large Break Loss of Coolant Accident.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because shutdown margin is the concern, however, the required boration is 1800 gallons for each rod stuck out, not only for the two not assumed in the accident analysis. The proper accident for the analysis is listed.

B. Incorrect. Plausible because shutdown margin is the concern, however, the required boration is 1800 gallons for each rod stuck out, not only for the two not assumed in the accident analysis. The Main Steam Line Break not the LBLOCA is the proper accident for the analysis listed.

C. Correct. A boration of 1800 gallons per rod is required to ensure shutdown margin in the event of a Main Steam Line Break per EOP-0.0A, Attachment 1.A.

D. Incorrect. Plausible because a boration of 1800 gallons per rod is required to ensure shutdown margin in the event of a Main Steam Line Break per EOP-0.0A, Attachment 1.A. The Main Steam Line Break not the LBLOCA is the proper accident for the analysis listed.

Technical Reference(s) EOP-0.0B, Attachment 1.A; Technical Specification LCO 3.1.1 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** EOP-0.0, Reactor Trip or Safety Injection including the Purpose, Applicability, Symptoms/Entry Conditions, Operator Actions, Bases, Foldout Pages and Attachments.

Question Source:
- Bank: ILOT8310
- Modified Bank: ____ (Note changes or attach parent)
- New: ____

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge: ____
- Comprehension or Analysis: X

10 CFR Part 55 Content:
- 55.41: 10
- 55.43: ____

Comments / Reference: EOP-0.0B, Attachment 1.A	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-0.0B
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 18 OF 117

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOP-0.0B REACTOR TRIP OR SAFETY INJECTION

1. RCP TRIP CRITERIA

NOTE: ABN-101, REACTOR COOLANT PUMP TRIP/MALFUNCTION criteria for tripping an RCP is applicable during use of the Emergency Procedures.

Trip all RCPs if BOTH conditions listed below occur:

a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
b. CCP or SI pump - AT LEAST ONE RUNNING

2. SHUTDOWN MARGIN CRITERIA
Emergency borate per ABN-107 if either of the following conditions below occur:

- Two or more control rods NOT fully inserted (1800 gallons of 7000 ppm boric acid for each control rod not fully inserted).
- Control rod position indication is NOT available (3600 gallons of 7000 ppm boric acid).

Comments / Reference: Technical Specification LCO 3.1.1 Bases	Revision: 68

SDM
B 3.1.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm average fuel pellet enthalpy at the hot spot for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements area main steam line break (MSLB) and boron dilution accidents, as described in the accident analysis (Ref. 2).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	2	
	K/A	037 AA1.01	
Level of Difficulty: 4	Importance Rating	3.7	

Steam Generator Tube Leak: Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak: Maximum controlled depressurization rate for affected SG

Proposed Question: 22

Given the following conditions:

- Unit 2 has experienced a three gpm Steam Generator Tube Leak.
- The Unit was shutdown to MODE 3 in accordance with ABN-106, High Secondary Activity.
- EOP-0.0B, Reactor Trip or Safety Injection was completed through Step 4.
- EOS-0.1B, Reactor Trip Response was completed with a transition to IPO-005B, Plant Cooldown from Hot Standby to Cold Shutdown.
- The Reactor Operator is ready to commence a Reactor Coolant System cooldown in accordance with ABN-106.

Which of the following details the ABN-106, High Secondary Activity requirements for the Steam Generator (SG) depressurization prior to the Safety Injection Block at less than 1960 psig?

Dump steam to the Main Condenser…

A. …at the maximum rate while ensuring Main Steam isolation is avoided.

B. …at the maximum rate while controlling Pressurizer level 30% to 70%.

C. …at less than or equal to 100ºF/hr cooldown rate while controlling Pressurizer level 17% to 30%.

D. …at less than or equal to 100ºF/hr cooldown rate while maintaining sub-cooling 60ºF to 70ºF.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because in EOP-3.0B, Steam Generator Tube Rupture, the SG depressurization guidance is at maximum rate to avoid main steam isolation.
B. Incorrect. Plausible because maximum rate is specified in EOP-3.0B and the Pressurizer level band is used in ABN-106 once SI has been blocked. The lower band of 17% to 30% must be used until SI is blocked.
C. Correct. These are the correct values per ABN-106 until SI is blocked.
D. Incorrect. Plausible because the cooldown rate is correct but the sub-cooling limitation cannot be applied until SI has been blocked.

Technical Reference(s): EOP-3.0B, Step 6.c; ABN-106, Steps 3.3.17 & 3.3.21; ABN-106, Attachments 1 & 2 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Steam Generator Tube Leakage greater than or equal to 75 gpd in accordance with ABN-106, High Secondary Activity.

Question Source:
Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments / Reference: EOP-3.0B, Step 6.c	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-3.0B
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 9 OF 101

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	c. Dump steam to condenser from intact SG(s) at maximum rate and avoid main steam isolation. 1) Transfer Steam Dump to steam pressure mode. 2) Place the steam pressure controller in manual and increase demand. 3) When P-12 (553°F TAVG) is reached, select bypass interlock on Steam Dumps and continue cooldown.	c. Dump steam at maximum rate from intact SG(s) using SG atmospheric(s). 1) Make plant announcement and notify Plant Staff of steam release. 2) Place SG(s) atmospheric(s) controller(s) in manual and increase demand or take local control of SG(s) atmospheric(s) and open valve. IF no intact SG available THEN perform the following: • Use faulted SG. -OR- • Go to ECA-3.1B, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, Step 1.

Comments / Reference: ABN-106, Step 3.3.17	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 25 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Control cooldown to maintain pressurizer level greater than 17%.
PZR level shall not be raised to >30% until SI is blocked.
The RCS shall not be cooled down below 510 deg prior to SI block.
SI shall not be blocked until adequate SDM for 350 deg, xenon free, has been verified.

(Provide Attachment 2 to RO to track cooldown requirements)

NOTE:
- An initial cooldown rate of 30-60 deg/hr is reccomended to enhance PZR level control and allow time to adjust AFW flow.
- Just prior to commencing cooldown, raise charging flow approximately 1gpm for each deg/hr of cooldown rate, to offset RCS contraction.

17 Cooldown the RCS:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a. Initiate monitoring of the RCS pressure AND temperature per OPT-407.	IF Steam Dumps to condenser can NOT be used, THEN cool down the RCS using intact Steam Generator Atmospheric Relief(s)
☐ b. Ensure Steam Dumps in STM PRESS mode in manual.	
☐ c. Adjust u-PK-507, STM DMP PRESS CTRL to maintain Cooldown Rate - LESS THAN OR EQUAL TO 100°F/hr	
☐ d. WHEN P-12 (553° F TAVG) is reached, THEN select bypass interlock on Steam Dumps and continue cooldown.	
☐ e. Adjust charging flow as needed to control PZR level 17%-30%.	

Comments / Reference: ABN-106, Step 3.3.21	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 26 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 18 Using Normal pressurizer spray, reduce RCS pressure to approximately 1925 psig. (1900-1950)	
☐ 19 Chemistry confirms that RCS boron concentration is >350 deg, Xenon free prior to blocking SI.	
20 WHEN RCS pressure < 1960 psig, THEN BLOCK: ☐ • MSL ISOL SI ☐ • PRZR PRESS SI	

NOTE: When reducing RCS pressure, do not reduce pressure below 900 psig before addressing the actions to isolate the Accumulators.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 21 Continue cooldown with the following limitations (ref Attachment 2): • Reduce RCS pressure to maintain 60-70 deg subcooling. • Gradually raise the cooldown rate to no more than 100 deg/Hr. • Maintain PZR level 30-70%.	

Comments / Reference: ABN-106, Attachment 1	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 30 OF 31

ATTACHMENT 1
PAGE 1 OF 1

COOLDOWN BRIEFING

These points are added to the items covered by the IPO-003A/B pre-trip brief (Attachment 9).

Emphasize the overall goals of:

- Establish emergency boration to achieve the needed boration ASAP and allow blocking of SI and raising Pzr level
- Cooling down in a controlled fashion not to exceed 100°F/Hr.
- Reduce pressure to 1900 psig at step 19 to maximize charging capacity and minimize the leak.
- Block SI to allow continued cooldown using the 60°F subcooling limit.
- Avoid SI unless inventory control cannot be maintained.

Specific items:

- Initiate Emergency Boration per IPO-003A/B as soon as practical, consistent with the verification steps of EOP-0.0A/B. Use an extra RO if available.
- Control Pzr level to 30% until SI blocked.
- Lead the cooldown with inventory. Raise charging flow first, then start the cooldown. If PZR level is already 30%, raise flow as soon as PZR level starts to trend down.

Comments / Reference: ABN-106, Attachment 2	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 31 OF 31

ATTACHMENT 2
PAGE 1 OF 1

COOLDOWN LIMITATIONS

Prior to SI Block

- Maintain PZR level >17% but < 30%
- Maintain PZR pressure 1900 - 1950 psig
- Maintain RCS temperature >510°F
- Control non-affected SG levels 60-75%
- Control leaking SG level > 43% (10% Unit 2)
- Max cooldown rate is 100°F/Hr (but should be less for control)

After SI Block

- Maintain sub-cooling 60-70°F
- Maintain RCS pressure >900 psig until the accumulators are isolated
- Control non-affected SG levels 60-75%
- Control leaking SG level > 43% (10% Unit 2)
- Max cooldown rate is 100°F/Hr

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	2	
	K/A	069 AK2.03	
Level of Difficulty: 4	Importance Rating	2.8	

Loss of Containment Integrity: Knowledge of the interrelations between Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch

Proposed Question: 23

Which of the following would require declaring the Containment inoperable in accordance with Technical Specification LCO 3.6.1, Containment while in MODE 1?

A. The outer door of the Personnel Air Lock is inoperable and closed and the inner door cannot be closed.

B. The Emergency Air Lock interlock mechanism is inoperable and neither door can be locked closed.

C. The Personnel Air Lock electrical interlock mechanism is inoperable and neither door can be locked closed.

D. Air leakage through the Emergency Air Lock has resulted in exceeding the overall Containment leakage rate.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this situation would require entry into LCO 3.6.2, Containment Air Locks, Condition D which would require a shutdown to MODE 3 in 6 hours.
B. Incorrect. Plausible because this situation would require entry into LCO 3.6.2, Containment Air Locks, Condition D which would require a shutdown to MODE 3 in 6 hours.
C. Incorrect. Plausible because this situation would require entry into LCO 3.6.2, Containment Air Locks, Condition D which would require a shutdown to MODE 3 in 6 hours.
D. Correct. In accordance with LCO 3.6.2 Containment Air Locks NOTE 3, this condition requires declaring the Containment inoperable.

Technical Reference(s): Technical Specification LCO 3.6.2; Technical Specification LCO 3.6.1 Bases; LO21SYSCY1 Study Guide — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Containment system including Technical Specifications, TRM and ODCM.

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	9	
	55.43		

Comments / Reference: Technical Specification LCO 3.6.2	Amendment: 161

Containment Air Locks
3.6.2

3.6 CONTAINMENT SYSTEM

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

--NOTES--

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

Comments / Reference: Technical Specification LCO 3.6.2	Amendment: 161

Containment Air Locks
3.6.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. -------------------------------	
	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND	
	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours

Comments / Reference: Technical Specification LCO 3.6.2	Amendment: 161

Containment Air Locks
3.6.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-----------------------NOTES--------------------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ---	
	B.1 Verify an OPERABLE door is closed in the affected air lock.	1 hour
	AND	
	B.2 Lock an OPERABLE door closed in the affected air lock.	24 hours

Comments / Reference: Technical Specification LCO 3.6.2	Amendment: 161

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND	
	C.2 Verify a door is closed in the affected air lock.	1 hour
	AND	
	C.3 Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

Comments / Reference: Technical Specification LCO 3.6.1 Bases	Revision: 68

Containment
B 3.6.1

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break and a rod ejection accident (REA) (Ref. 2 and 3). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for these DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2 and 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% of containment air weight per day in the safety analysis at P_a = 48.3 psig. The calculated peak pressure for LOCAs is less than 48.3 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and containment purge, hydrogen purge, and containment pressure relief valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

Comments / Reference: LO21SYSCY1 Lesson Notes	Revision: 5/2/2011
For some water tests, an air-driven hydro pump is used as the pressure source. The hydro pump is usually a self-contained rig that consists of the following: Air-driven pump -- provides the pressure source for a hydrostatic test. Air regulator -- determines the pressure output of the pump. Discharge manifold -- allows installation of a calibrated discharge gauge (Hiese or Perma-cal type) and also allows controlled venting of the pressurized volume. The higher of the two valve leakage values is considered to be the leakage from Containment through that particular penetration (most conservative assumption.) That value is added to the leakage value from all other penetrations and the total Containment Leakage sum, for Technical Specification compliance purposes, is derived.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	1	
Change: 2	Group	2	
	K/A	074 EA1.06	
Level of Difficulty: 3	Importance Rating	3.6	

Inadequate Core Cooling: Ability to operate and/or monitor the following as they apply to an Inadequate Core Cooling: RCPs

Proposed Question: 24

Given the following conditions:

- FRC-0.1A, Response to Inadequate Core Cooling, is in progress on Unit 1.
- Efforts to depressurize the Steam Generators have been ineffective.

Which of the following is required in accordance with FRC-0.1A, Response to Inadequate Core Cooling?

Start…

A. …only one RCP. If core cooling is insufficient PORVs are opened and the remaining RCPs are started sequentially.

B. …no more than two RCPs to allow for spray flow and depressurize the RCS enough to permit RHR injection flow.

C. …no more than three RCPs and reserve one RCP for future use, while the other three cool the core.

D. …RCPs one at a time until core exit TCs are less than 1200ºF in order to force two phase flow through the core for cooling.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because in FRC-0.2A, Response to Degraded Core Cooling, Step 5, at least one RCP is verified running. At Step 7, the RCP in loop 4 is stopped to preserve it for future use. Opening the PORVs is directed in FRC-0.1A at Step 20.b RNO once it is determined that running all RCPs has been ineffective at lowering core exit temperature below 1200°F.

B. Incorrect. Plausible because depressurizing the RCS to permit RHR injection flow is attempted earlier in FRC-0.1A at Step 13 in conjunction with the steam generator depressurization (see Step 13 bases). However, at 1200°F all RCPs are started.

C. Incorrect. Plausible because in FRC-0.2A, Response to Degraded Core Cooling RCP 4 is secured at Step 7 to preserve it for future depressurization spray flow. However, with core exit thermocouple temperatures greater than 1200°F all RCPs would be started.

D. Correct. As described in Step 20 of FRC-0.1A, starting the RCPs when core exit temperatures (CET) are greater than 1200°F will result in clearing of the water inventory in the RCS intermediate leg and permit circulation of hot gases from the overheated core into the steam generators. As stated in Steps 20.a through 20.d, idle RCPs are continuously started until CET temperatures are less than 1200°F.

Technical Reference(s)	FRC-0.1A, Steps 13, 18, & 20	Attached w/ Revision: See Comments / Reference
	FRC-0.1A, Attachment 5 Bases	
	FRC-0.2A, Steps 5 & 7	

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRC-0.1, Response to Inadequate Core Cooling.

Question Source:	Bank	ILOT0991	
	Modified Bank		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments / Reference: FRC-0.1A, Step 13.c RNO	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 8	PAGE 9 OF 45

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Partial uncovering of SG tubes is acceptable in the following steps.

NOTE: After the low steamline pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*13	Depressurize All Intact SGs To 170 PSIG:	
	a. Dump steam to condenser at maximum rate and avoid main steam isolation.	a. Manually or locally dump steam at maximum rate from intact SG(s) atmospheric.
	b. **WHEN** PRZR pressure is less than 1960 psig, **THEN** block low steamline pressure SI signal.	
	c. Check SG pressures - LESS THAN 170 PSIG	c. **IF** SG pressure decreasing, **THEN** return to Step 11. OBSERVE CAUTION PRIOR TO STEP 11. **IF NOT**, **THEN** go to Step 20. OBSERVE NOTE PRIOR TO STEP 20.

Comments / Reference: FRC-0.1A, Step 18	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 8	PAGE 15 OF 45

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	Check Core Cooling:	
	a. Core exit TCs - LESS THAN 1200°F	a. Go to Step 20. OBSERVE NOTE PRIOR TO STEP 20.
	b. At least two RCS hot leg temperatures - LESS THAN 350°F	b. Return to Step 16.
	c. RVLIS indication - GREATER THAN OR EQUAL TO 11 IN ABOVE CORE PLATE LIGHT LIT	c. Return to Step 16.

Comments / Reference: FRC-0.1A, Step 20	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 8	PAGE 16 OF 45

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Normal support conditions are desired but not required for starting the RCPs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20	Check If RCPs Should Be Started:	
	a. Core exit TCs - GREATER THAN 1200°F	a. Go to Step 21.

Comments / Reference: FRC-0.1A, Step 20	Revision: 8

b. Check if an idle RCS cooling loop is available:

- Narrow range SG level - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)

-AND-

- RCP in associated loop - AVAILABLE AND NOT RUNNING

b. Perform the following:

1) Reset SI.

A) Main Control Board
Train A - 1/1SIRA
Train B - 1/1SIRB

B) Local at SSPS cabinets per EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, Attachment 9.

2) Reset Containment Isolation Phase A and Phase B.

3) Ensure air compressor running and establish instrument air to containment.

4) Ensure ACCUM 1•4 VENT CTRL, 1-HC-943 - CLOSED

5) Open SI/PORV ACCUM N2 ISOL VALVE, 1/1-8880

6) Open all PRZR PORVs and block valves.

7) IF core exit TCs remain greater than 1200°F, THEN open all vent paths to containment:

- Reactor vessel head vents.
- PRZR vents.

8) Go to Step 21.

c. Start RCP in one idle RCS cooling loop.

d. Return to Step 20a.

Comments / Reference: FRC-0.1A, Attachment 5, Step 13 Bases	Revision: 8

ATTACHMENT 5
PAGE 6 OF 16

BASES

STEP 11: The minimum AFW flow requirement of 460 gpm satisfies the feed flow requirement of the Heat Sink Status Tree. Narrow range level is re-established in all intact SGs to maintain symmetric cooling of the RCS. The control range ensures adequate inventory with level readings on span.

If the inadequate core cooling symptoms were caused by a loss of secondary heat sink, i.e., total AFW flow is less than 460 gpm in combination with a loss of high pressure safety injection, then the operator is instructed to go to Step 20. Step 20 will provide temporarily improved core cooling until either feedwater or safety injection is restored.

This step is a Continuous Action Step.

STEP 12: Any open, isolable RCS vent path should be closed to reduce or eliminate the loss of RCS inventory through that path. Therefore, this step particularly checks PRZR PORVs and block valves in addition to other plant specific RCS vent paths.

To ensure operability of the PRZR PORV block valves, it should be verified that power is available to them. PRZR PORVs are closed to preclude the possibility of an undetected stuck open valve. At least one block valve is left open to ensure availability of at least one PORV for pressure excursions in the RCS (due to degraded conditions). Also, it is desirable to have at least one PORV available to preclude the use of PRZR safety valves.

NOTE: Maintaining of SG level during the rapid depressurization will be difficult. Partial uncovery of the SG tubes may occur if the steam mass removal rate exceeds the maximum feedwater mass addition rate. This is an anticipated result of the rapid SG depressurization. The operator should maintain adequate feed flow in an attempt to keep the SG tubes covered since this will maximize primary-to-secondary heat transfer.

NOTE: Alerts the operator to the potential for inadvertent steamline isolation during the subsequent steam generator depressurization. The rapid cooldown should be continued using the Steam Generator ARVS if MSIV closure occurs.

STEP 13: The rapid secondary depressurization has been shown to be the most effective way to reduce RCS pressure. RCS pressure must be reduced in order for the SI accumulators and RHR pumps to inject.

Comments / Reference: FRC-0.1A, Attachment 5, Step 20 Bases	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 8	PAGE 38 OF 45

ATTACHMENT 5
PAGE 9 OF 16

BASES

STEP 20: The operator will enter this step if:

a. The SGs can not be depressurized; or
b. SG depressurization was not effective in restoring adequate core cooling; or
c. Secondary heat sink is lost

The actions of this step may provide temporary core cooling until some form of makeup flow to the RCS is established or one of the above items is restored.

To temporarily restore core cooling, the operator is instructed to start RCPs one at a time until core exit TCs are less than 1200°F. The RCPs should force two phase flow through the core, temporarily keeping it cool. Even single phase forced steam flow will cool the core for some time provided the RCPs can be kept running and a heat sink is available.

An idle loop is a loop without an RCP running in it.

The particular RCPs to be started will be based on availability; however, if possible consideration should be given to not starting RCP 4 due to maximum capability of spray flow which may be required in subsequent recovery actions.

Starting the RCPs in this step when the core exit temperatures are greater than 1200°F will result in the clearing of the water inventory in the RCS intermediate leg (loop seal) and permit the circulation of hot gases from the overheated core to circulate through the steam generators. If the water level in the steam generators is very low at the time the RCPs are started, high steam generator tube temperatures would occur, leading to possible creep failure of the steam generator tubes. Therefore, RCPs are only started in this step if there is sufficient water level in their associated steam generator to protect the steam generator tubes from creep rupture.

If RCP restart is not effective in decreasing core exit TC temperatures below 1200°F, then the PRZR PORVs should be opened. Opening the PRZR PORVs may help reduce RCS pressure enough to cause RHR injection. If core exit TCs remain above 1200°F after all PRZR PORVs and block valves are open, the operator is instructed to open all other RCS vent paths to containment to reduce RCS pressure.

Comments / Reference: FRC-0.2A, Steps 5 & 7	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.2A
RESPONSE TO DEGRADED CORE COOLING	REVISION NO. 8	PAGE 10 OF 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

> CAUTION: RCS letdown or RCP seal return to VCT should not be initiated if core damage is suspected or is imminent unless recommended by Plant Staff.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Check RCP Status:	
	a. At least one RCP - RUNNING	a. Go to Step 6.
	b. Check RCP Support Conditions - AVAILABLE PER ATTACHMENT 3	b. Establish support conditions for the operating RCP(s).
6	Check Core Cooling:	
	a. Core exit TCs - LESS THAN 750°F	a. IF decreasing, THEN return to Step 1. IF NOT, THEN go to Step 7.
	b. RVLIS indication - GREATER THAN OR EQUAL TO 11 IN ABOVE CORE PLATE LIGHT LIT	b. IF core exit TCs stable or decreasing, THEN return to procedure and step in effect. IF NOT, THEN return to Step 1.
	c. Return to procedure and step in effect.	
7	Check If One RCP Should Be Stopped:	
	a. All RCPs - RUNNING	a. Go to Step 8.
	b. Stop RCP in loop 4.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/25/2014	Tier	1	
Change: 3	Group	2	
	K/A	076 AK3.06	
Level of Difficulty: 4	Importance Rating	3.2	

<u>High Reactor Coolant Activity</u>: Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Actions contained in EOP for high reactor coolant activity

Proposed Question: 25

Given the following conditions:

- Unit 2 is at 800 MWe, 48 hours after a Heater Drain Pump 2-01 trip.
- Chemistry reports that Reactor Coolant System (RCS) Cs-137 levels are elevated over the last shift.
- The Positive Displacement Pump is in service with letdown flow at 75 gpm.

Which of the following indicates the actions required in accordance with ABN-102, High Reactor Coolant Activity and reason for those actions?

A. Maintain letdown at 75 gpm to minimize dose rates in Auxiliary Building.
Contact Core Performance Engineering to determine extent of fuel failure.

B. Maintain letdown at 75 gpm to minimize dose rates in Auxiliary Building.
Contact Chemistry to determine if a CRUD burst has occurred.

C. Start a Centrifugal Charging Pump and secure the Positive Displacement Pump to raise letdown to 120-140 gpm to increase ion exchange.
Contact Core Performance Engineering to determine extent of fuel failure.

D. Start a Centrifugal Charging Pump and secure the Positive Displacement Pump to raise letdown to 120-140 gpm to increase ion exchange.
Contact Chemistry to determine if a CRUD burst has occurred.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that minimizing letdown would lower dose rates, however, raising letdown is required to remove more radioactive ions and particulates. Cs-137 is indicative of fuel failure.

B. Incorrect. Plausible because it could be thought that minimizing letdown would lower dose rates, however, raising letdown is required to remove more radioactive ions and particulates. Cs-137 is indicative of fuel failure so checking for a CRUD burst is not the correct action.

C. Correct. Raising letdown flow is the correct action to remove more radioactive ions and particulates and Core Performance Engineering should be contacted to determine the extent of fuel damage as Cs-137 is indicative of fuel failure.

D. Incorrect. Raising letdown flow is the correct action to remove more radioactive ions and particulates. Cs-137 is indicative of fuel failure so checking for a CRUD burst is not the correct action.

Technical Reference(s) ABN-102, Steps 2.3.2, 2.3.3, 2.3.6 & 2.3.7 Attached w/ Revision: See Comments / Reference
ABN-102, Step 2.3.7 NOTE, 2.3.8, & 2.3.9

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to High Reactor Coolant Activity in accordance with ABN-102, High Reactor Coolant Activity.

Question Source: Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ____

Comments / Reference: ABN-102, Step 2.3.2, 2.3.3, & 2.3.6	Revision: 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-102
HIGH REACTOR COOLANT ACTIVITY	REVISION NO. 7	PAGE 4 OF 6

2.3 Operator Actions

NOTE:
- Reactor Coolant System transients such as power changes, temperature changes, pressure changes, and starting and stopping RCPs can cause temporary increases in RCS activity.
- Monitor spiking and return to normal is not a real indication of failed fuel and as such does not require sampling. A steady or sustained increase over time would be a real indication of failed fuel/RCS activity problems.

☐ 1. Request additional reactor coolant specific activity samples be taken in accordance with CHM-111 for isotopic content analysis per Technical Specification 3.4.16, SURVEILLANCE REQUIREMENTS.

☐ 2. Notify Chemistry to review chemistry data and Core Performance Engineering to review chemistry data and core follow trends. Chemistry will determine if a "CRUD" burst has occurred. Core Performance Engineering will determine if the source of RCS activity is failed fuel and the extent of failed fuel, if any.

☐ 3. Increase letdown flow to 120-140 gpm as follows:

a) IF PDP is in operation, THEN start up a centrifugal charging pump AND shutdown PDP per SOP-103A/B.

b) Increase letdown flow to 120-140 gpm per SOP-103A/B.

☐ 4. Notify Radiation Protection that radiation levels may increase in Auxiliary and Safeguards Buildings AND on any ARMs.

☐ 5. Make a plant announcement via Gai-Tronics of indication of an increase in RCS Activity AND a possibility of increased radiation in Auxiliary and Safeguards Buildings.

NOTE: A rapid increase of RCS fission product isotopes during steady state operation may indicate fuel cladding damage. (e.g., Xe-133, Kr-85M, Cs-137, Cs-136, Sr-84, Sr-90, Iodine).

☐ 6. IF Core Performance Engineering Review of the chemistry data indicates failed fuel, THEN proceed as follows:

a) Refer to EPP-201.

b) Refer to Technical Specifications 3.4.16.

c) Review logs for any known RCS to Secondary Leakage.

Comments / Reference: ABN-102, Step 2.3.7 NOTE, 2.3.8, & 2.3.9	Revision: 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-102
HIGH REACTOR COOLANT ACTIVITY	REVISION NO. 7	PAGE 5 OF 6

NOTE:
- An increase of RCS activated corrosion products may indicate a "CRUD" burst. (e.g., Fe-59, Co-58, Co-60, Mn-54, Mn-56, Cr-51, and Zr-95).
- The stepping or tripping of control or shutdown rods should be kept to a minimum when reactor coolant CRUD levels are high to reduce the potential for CRDM mis-stepping due to CRUD contamination of CRDM latch assemblies (CR 2009-008942).

☐ 7. IF RCS activity increase is believed to be result of RCS transient OR "CRUD" burst, THEN refer to Technical Specification 3.4.16.

☐ 8. IF RCS activity is increasing slowly during steady state operation, THEN notify Chemistry to calculate a decontamination factor (DF) for fission and activation products listed above for CVCS mixed bed ion exchanger in use AND notify Shift Manager of results.

☐ 9. IF resin depletion is indicated, THEN transfer to standby CVCS mixed bed ion exchanger per SOP-103A/B.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	1	
Change: 2	Group	2	
	K/A	W/E02 EA2.2	
Level of Difficulty: 2	Importance Rating	3.5	

Safety Injection Termination: Ability to determine and interpret the following as they apply to the SI Termination: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Proposed Question: 26

Given the following conditions:

- Following a Reactor Trip and Safety Injection, the crew has transitioned to EOS-1.1A, Safety Injection Termination.
- Centrifugal Charging Pump 1-02 and both Safety Injection Pumps have been stopped and placed in standby.
- Normal Charging flow has been established.
- Containment pressure is 1.2 psig and stable.
- Reactor Coolant System subcooling is currently 19ºF and slowly degrading.
- Pressurizer level is 18% and slowly decreasing.

Which of the following actions is to be taken in accordance with the Foldout Page of EOS-1.1A, Safety Injection Termination?

Manually…

A. …control Charging flow as necessary and continue in EOS-1.1A, Safety Injection Termination.

B. …operate Emergency Core Cooling Pumps as necessary and continue in EOS-1.1A, Safety Injection Termination.

C. …operate Emergency Core Cooling Pumps as necessary and transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

D. …actuate Safety Injection and transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

Proposed Answer: C

Explanation:

A. Plausible since this is the action required if pressurizer level is below 6%, but with subcooling below the required value ECCS Pumps must be started and a transition made to EOP-1.0.

B. Plausible since ECCS Pumps are started as necessary, but a transition to EOP-1.0 is also required.

C. Based on subcooling being less than 25°F, ECCS Pumps must be started as necessary and a transition made to EOP-1.0.

D. Plausible since a transition to EOP-1.0 is required, but starting ECCS Pumps is the action to be taken instead of manually actuating SI.

Technical Reference(s) EOS-1.1A, Attachment 1.A — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the items on EOS-1.1, Safety Injection Termination Foldout Page including any equipment, parameter, set point or condition.

Question Source: Bank ILOT5805
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: EOS-1.1A, Attachment 1.A	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.1A
SAFETY INJECTION TERMINATION	REVISION NO. 8	PAGE 18 OF 49

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-1.1A, SI TERMINATION

1. SI REINITIATION CRITERIA

Following ECCS termination, (completion of Step 12), manually start ECCS pumps as necessary and go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1, if EITHER condition listed below occurs:

- RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	1	
Change: 3	Group	2	
	K/A	W/E13 G 2.4.35	
Level of Difficulty: 3	Importance Rating	3.8	

Steam Generator Overpressure: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects

Proposed Question: 27

Given the following conditions:

- Unit 1 has experienced a Reactor Trip from a spurious Main Steam Line Isolation signal.
- The crew is attempting to control Reactor Coolant System (RCS) temperature in EOS-0.1A, Reactor Trip Response.
- Steam Generator (SG) 1-04 pressure is 1250 psig and stable.
- SGs 1-01, 1-02, and 1-03 are 1092 psig and stable.
- Field Support has reported that 1-PV-2328, SG4 ATM RLF VLV is mechanically bound and will NOT open.
- The Unit Supervisor directs that FRH-0.2A, Response to Steam Generator Overpressure be performed.

Which of the following Nuclear Equipment Operator field actions are specified in FRH-0.2A, Response to Steam Generator Overpressure to reduce SG 1-04 pressure?

Open…

A. …1-HV-2336B, MSIV 1-04 BYP VLV.

B. …1-HV-2452-1, AFWPT STM SPLY VLV MSL 4.

C. …1-HV-2412, MSL 4 BEF MSIV D\POT ISOL VLV.

D. …1MS-0712, MSL 1-04 TO AFWPT STM SPLY VLV BYP VLV.

Proposed Answer: A

Explanation:

A. Correct. Local actions to open this valve are included in FRH-0.2A, Steps 4 and 8.
B. Incorrect. Plausible because the opening of this valve is included in FRH-0.2A, Steps 4 and 8, but would be performed by the Reactor Operator in lieu of the NEO.
C. Incorrect. Plausible because the opening of this valve is included in FRH-0.2A, Steps 4 and 8, but would be performed by the Reactor Operator in lieu of the NEO.
D. Incorrect. Plausible because the opening of this valve would be effective in reducing the SG pressure, but this valve is not specified to be opened in FRH-0.2A, Steps 4 and 8. The Bases for Step 8 does allow the use of any plant specific means which could result in the utilization of this valve for pressure reduction.

Technical Reference(s): FRH-0.2A, Steps 4 & 8; FRH-0.2A, Attachment 2, Step 8 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.2, Response to Degraded Core Cooling.

Question Source:
- Bank: ____
- Modified Bank: ____ (Note changes or attach parent)
- New: X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge: X
- Comprehension or Analysis: ____

10 CFR Part 55 Content:
- 55.41: 4
- 55.43: ____

Comments / Reference: FRH-0.2A, Step 4	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.2A
RESPONSE TO STEAM GENERATOR OVERPRESSURE	REVISION NO. 8	PAGE 3 OF 13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Throughout this procedure, "affected" refers to any SG in which pressure is greater than 1235 psig.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Identify Affected SG(s):	
	a. Any SG pressure - GREATER THAN 1235 PSIG	a. Return to procedure and step in effect.
2	Verify Feedwater Isolation - COMPLETE	Manually close valve(s).
3	Check Affected SG(s) Narrow Range Level - LESS THAN 93% (86% FOR ADVERSE CONTAINMENT)	Go to FRH-0.3A, RESPONSE TO STEAM GENERATOR HIGH LEVEL, Step 1.
4	Dump Steam From The Affected SG(s): • SG atmospheric -OR- • Locally with the Main steamline isolation bypass valve -OR- • IF SG 1 or 4 affected, THEN use steam supply valve to TDAFW pump. -OR- • Before MSIV drip pot isolation valve.	Go to Step 6. OBSERVE CAUTION PRIOR TO STEP 6.

Comments / Reference: FRH-0.2A, Step 8	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.2A
RESPONSE TO STEAM GENERATOR OVERPRESSURE	REVISION NO. 8	PAGE 5 OF 13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8	Continue Attempts To Manually Or Locally Dump Steam From Affected SG(s): • SG atmospheric -OR- • Locally with the main steamline isolation bypass valve -OR- • <u>IF</u> SG 1 or 4 affected, <u>THEN</u> use steam supply valves to TDAFW pump -OR- • Before MSIV drip pot isolation valve	

Comments / Reference: FRH-0.2A, Attachment 2, Step 8 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.2A
RESPONSE TO STEAM GENERATOR OVERPRESSURE	REVISION NO. 8	PAGE 9 OF 13

ATTACHMENT 2
PAGE 2 OF 6

BASES

STEP 5: Steam release should result in affected SG(s) pressure decreasing. If this does not occur, the operator is directed to Step 6. If pressure is decreasing but is not below 1235 psig, the operator is directed to return to Step 3 and continue monitoring level and releasing steam. If steam release drops the affected SG(s) pressure to less than 1235 psig, then the steam release is controlled to maintain pressure and the operator is instructed to return to the procedure in effect.

This action contains the action verb "control" which implies continuous performance; therefore, this step has been identified as a continuous action step.

CAUTION: If AFW flow is re-established to an affected SG prior to establishing a steam release path, the AFW flow could further increase the affected SG pressure.

STEP 6: If the operator has been unsuccessful in releasing steam to lower the affected SG pressure below design pressure, the operator should proceed to isolate AFW flow to the affected SG since the AFW system is a high pressure water source. This eliminates an additional source of over pressurization of the affected SG(s) and may prevent a potential failure of secondary integrity.

STEP 7: Excessive heat transfer from the primary system may be the cause of the affected SG over pressurization. Therefore, a check on RCS hot leg temperatures is made to determine this. If RCS hot leg temperatures are greater than 535°F, a cooldown is initiated by dumping steam from the unaffected SG(s) to aid in reducing the temperature and pressure in the affected SG(s).

STEP 8: The operator should continue attempts to manually or locally release steam from the affected SG(s), utilizing the four alternative release paths plus any plant specific means identified until the challenge to the SG pressure boundary is mitigated.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/6/2014	Tier	2	
Change: 3	Group	1	
	K/A	003 K5.04	
Level of Difficulty: 3	Importance Rating	3.2	

Reactor Coolant Pump System: Knowledge of the operational implications of the following concepts as they apply to the RCPs: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow

Proposed Question: 28

Given the following conditions:

- Unit 1 is steady at 30% power when Reactor Coolant Pump 1-02 trips and causes a transient in Steam Generator 1-02.

Which of the following describes how steam flow and water level in Steam Generator 1-02 initially respond to the trip of Reactor Coolant Pump 1-02 prior to manually tripping the reactor?

Initially, Steam Generator 1-02 steam flow __________ and level __________.

A. increases increases

B. increases decreases

C. decreases decreases

D. decreases increases

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that pressure dropped in Steam Generator #2 which caused a swell and level increase.
B. Incorrect. Plausible because Steam Generator level will decrease, however, steam flow will also decrease as the Steam Generator cools and Steam Generator pressure decreases.
C. Correct. Because the Steam Generator with the tripped Reactor Coolant Pump stops steaming, steam flow will decrease and Steam Generator level will also decrease due to shrink.
D. Incorrect. Plausible because steam flow will decrease, however, Steam Generator level will also decrease due to shrink.

Technical Reference(s) ABN-101, Step 2.3.1 NOTE Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Reactor Coolant Pump System.

Question Source: Bank ILOT5789
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments / Reference: ABN-101, Step 2.3.1 NOTE	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 4 OF 48

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

[C]
CAUTION: A Reactor Coolant Pump shall NOT be started with the reactor in MODE 1 or 2.

NOTE:
- Diamond step 1 denotes Initial Operator Actions.
- With a Reactor Coolant Pump stopped, the affected loop will stop steaming.

1 Check Plant status

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a. Verify Reactor - Tripped	a. Perform the Following: 1) Trip Reactor AND GO TO EOP-0.0A/B while other qualified operators continue with this procedure. 2) GO TO Step 2.
☐ b. GO TO EOP-0.0A/B while other qualified operators continue with this procedure.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	004 K4.04	
Level of Difficulty: 3	Importance Rating	3.2	

Chemical and Volume Control System: Knowledge of the CVCS design feature(s) and/or interlock(s) that provide for the following: Manual/automatic transfers of control

Proposed Question: 29

Given the following conditions:

- 1-LT-0112, CVCS VOLUME CONTROL TANK 1-01 LEVEL TRANSMITTER 0112 has been removed from service by Instrument & Control.
- 1-LI-112A, VCT LVL indicates 50%.
- 43/1-MU RCS MU MODE SELECT is in MAN.
- ALB-6A Window 4.5 – VCT LVL LO-LO annunciates.
- 1-LI-185, VCT LVL indicates 2%.

Which of the following actions would ensure a suction source to the operating Centrifugal Charging Pump?

A. Open 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV, then close 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV.

B. Close 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV, then open 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV.

C. Open 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV, then open 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV.

D. Close 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV, then open 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV.

Proposed Answer: A

Explanation:

A. Correct. The interlocks between the suction valves are such that once LCV-112D is open, LCV-112B may be manually closed. These actions would provide a suction source from the RWST and isolate the degraded source from the VCT.

B. Incorrect. Plausible because some interlocked valves allow simultaneous movement based on the speed of valve operation (LCV-459/460 and 8149A/C). These valves do not and as such LCV-112C will NOT close until LCV-112E is opened.

C. Incorrect. Plausible if thought that opening both LCV-112D and LCV-112E would ensure a suction source, however LCV-112B or LCV-112C must be closed to ensure no gas entrainment at the suction of the CCPs from the gas in the VCT.

D. Incorrect. Plausible because some interlocked valves allow simultaneous movement based on the speed of valve operation (LCV-459/460 and 8149A/C). These valves do not and as such LCV-112B will NOT close until LCV-112D is opened.

Technical Reference(s) LO21.SYS.CS1, Pages 26 & 27 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Chemical and Volume Control system.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: LO21.SYS.CS1, Page 26	Revision: 04/28/11

u-LCV-0112B will automatically close if the following conditions are met:

The following actuation signals occur:

- Low-low level signals from both u-LT-0112 and u-LT-0185
- Train A Safety Injection signal

AND

Refueling Water Storage Tank To Charging Pump Suction Valve u-LCV-0112D is open

The automatic closure interlocks for u-LCV-0112C are the same except that this valve is a Train B component, powered from Motor Control Center uEB4-1, interlocked with Refueling Water Storage Tank To Charging Pump Suction Valve u-LCV-0112E instead of u-LCV-0112D and receiving its Safety Injection signal from Train B.

The automatic closure arrangement maintains a source of water for the charging pumps by ensuring the volume control tank source does not isolate until the refueling water storage tank is aligned to the charging pumps.

Comments / Reference: LO21.SYS.CS1, Page 27	Revision: 04/28/11

u-LCV-0112D will automatically open if the following conditions are met:

Any of the following actuation signals occur:

- Low-low level signals from both u-LT-0112 and u-LT-0185
- Train A Safety Injection signal

AND

Refueling Water Storage Tank To Charging Pump Suction Valve Handswitch, 1/u-LCV-0112D, is in the AUTO position

The automatic opening interlocks for u-LCV-0112E are the same except that this valve is a Train B component receiving its Safety Injection signal from Train B.

The automatic opening arrangement aligns a borated water source to the reactor coolant system through the charging pumps under conditions where either the normal source is lost or a safety injection signal has actuated.

When u-LCV-0112D automatically opens, conditions are established to automatically close u-LCV-0112B, isolating the VCT from the charging pumps. Similarly, when u-LCV-0112E automatically opens, conditions are established to automatically close u-LCV-0112C, again isolating the VCT from the charging pumps. The design redundancy is based on single failure criteria.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	005 K1.11	
Level of Difficulty: 3	Importance Rating	3.5	

Residual Heat Removal System: Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: RWST

Proposed Question: 30

Given the following conditions:

- Unit 1 has experienced a Large Break Loss of Coolant Accident.
- EOP-1.0A, Loss of Reactor or Secondary Coolant is in progress.
- The following signals have been RESET:
  - Safety Injection
  - Phase A Isolation
  - Phase B Isolation
- Before any other signals are reset, the Refueling Water Storage Tank (RWST) level decreases to 33%.

Which of the following describes the response of the Residual Heat Removal (RHR) Pump Suction Valves?

	RHR Containment Sump Suction Valves 1/1-8811A & 1/1-8811B	RWST to RHR Suction Valves 1/1-8812A & 1/1-8812B
A.	Automatically open	Automatically close
B.	Automatically open	Must be manually closed
C.	Must be manually opened	Must be manually closed
D.	Must be manually opened	Automatically close

Proposed Answer: B

Explanation:

A. Incorrect. Plausible since the sump suction valves receive an automatic open signal, but the RWST suction valves must be manually closed.

B. Correct. The RHR suction valves receive an auto switchover signal. Auto switchover for RHR will occur on low-low RWST level coincident with SI. Auto switchover for RHR also requires that the operator complete the alignment by closing the RWST to RHR Suction Valves. This SI signal is reset independently, so it will occur even after SI is reset.

C. Incorrect. Plausible since the RWST suction valves must be manually closed, but the sump suction valves will automatically open even with the SI reset since a separate signal is used to input this coincidence.

D. Incorrect. Plausible since one set of valves receive an automatic signal and the other set of valves must be manually operated, but the sump suction valves automatically open and the RWST suction valves must be manually closed.

Technical Reference(s): EOS-1.3A, Step 3; LO21.SYS.RH1, Pages 15 & 16 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Residual Heat Removal System.

Question Source:	Bank	ILOT6395	
	Modified Bank		(Note changes or attach parent)
	New		

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments / Reference: EOS-1.3A, Step 3	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 4 OF 54

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: Any ECCS pump taking suction from RWST should be stopped at RWST EMPTY. Any Containment Spray pump taking suction from RWST should be stopped when RWST level reaches 0%.

CAUTION: Any ECCS or Containment Spray pump that loses suction or shows indication of cavitation should be stopped. The CCP and SI pump should be stopped before stopping the RHR pump.

CAUTION: SI pumps should be stopped if RCS pressure is greater than their shutoff head pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 3	Align ECCS For Cold Leg Recirculation:	IF at least one flow path from the sump to the RCS can NOT be established or maintained, THEN go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1. Implement FRGs if required.
	a. Check open CNTMT SMP TO RHRP 1 AND RHRP 2 SUCT ISOL VLVS: • 1/1-8811A • 1/1-8811B	a. IF ONE RHR sump suction valve failed to open, THEN stop RHR pump with valve closed AND go to Step 3b to align operating RHR pump.

Comments / Reference: EOS-1.3A, Step 3	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 5 OF 54

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		3) Close 1/1-8701A(B) OR 1/1-8702A(B). 4) Open 1/1-8811A(B). 5) Start RHR pump 1(2). IF RHR pump suction valves NOT open due to SI Termination (ECCS not running for injection), THEN go to Step 4.
	b. Close RWST TO RHRP 1 AND RHRP 2 SUCT VLVS: • 1/1-8812A • 1/1-8812B	

Comments / Reference: LO21.SYS.RH1, Page 16	Revision: 10/20/11

CONTAINMENT SUMP TO RHR PUMP SUCTION ISOLATION VALVES (U-8811A&B)

The Containment Sump to RHR Pump Suction Isolation Valves allow recovery of borated water that has spilled from the Reactor Coolant System onto the Containment floor during LOCA conditions. These valves (along with the Containment Sump to Containment Spray Pump Suction Isolation Valves) also allow recovery of water sprayed from the Containment Spray System into the Containment Building. The Containment Sump to RHR Pump Suction Isolation Valves are normally closed and open automatically when the RWST reaches its LO-LO alarm at 33% level. The valves are located in independent train related lines to ensure a single failure does not cause a complete loss of Emergency Core Cooling System flow.

Bonnet Pressure Relief Valves (u-SI-0182 and u SI-0183) are provided on u-8811A and u-8811B respectively to preclude pressure locking of the Containment Sump to RHR Pump Suction Isolation Valves. Water trapped in the bonnet can thermally expand causing pressure to increase between the discs. Excessive pressure may preclude the motor from opening the valves. A relief is provided to prevent this occurrence. The relief valve lifts at 475 psig and relieves to the downstream side of the valve. The relief valve is classified as an active containment isolation valve.

The Containment Sump to RHR Pump Suction Isolation Valves are motor-operated valves which are operated from CB-04. Valve u-8811A receives power from uEB3-2 and valve u-8811B receives power from uEB4-2.

Opening the Train A valve using the Main Control Board switch requires the following interlocks be met:

- RWST to RHR Pump Suction Valve (u-8812A) must be CLOSED, and
- One of the RHR Pump Hot Leg Recirculation Isolation Valve (u-8701A or u-8702A) must be CLOSED, and
- The Main Control Board switch in the OPEN position

The interlocks for the Train B valve utilize their train B counterpart.

These interlocks aid in preventing cross-connecting the RWST and the Containment Sumps. Following a Safety Injection Actuation signal, these valves will automatically open once the RWST reaches its LO-LO alarm.

Comments / Reference: LO21.SYS.RH1, Pages 15 & 16	Revision: 10/20/11

REFUELING WATER STORAGE TANK TO RHR PUMP SUCTION VALVES (U-8812A&B)

The Refueling Water Storage Tank to RHR Pump Suction Valves allow the RHR System to transfer borated water from the RWST to the Reactor Coolant System. During normal at power operations, these valves are open as part of the RHR System standby status. These valves are closed when the RHR System is placed in the shutdown cooling mode. Once the unit had been shutdown and the reactor vessel head prepped for removal, these valves are reopened to allow filling of the Refueling Cavity.

The Refueling Water Storage Tank to RHR Pump Suction Valves are motor-operated valves which are operated from CB-04. Valve u-8812A receives power from uEB3-1 and valve u-8812B receives power from uEB4-1.

Opening the Train A valve using the Main Control Board switch requires the following interlocks be met:

- Containment Sump to RHR Pump Suction Isolation Valve (u-8811A) must be CLOSED, and
- The Main Control Board switch in the OPEN position
- The interlocks for the Train B valve utilize its Train B counterpart

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/25/14	Tier	2	
Change: 2	Group	1	
	K/A	006 A2.13	
Level of Difficulty: 2	Importance Rating	3.9	

Emergency Core Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation

Proposed Question: 31

Given the following condition:

- An inadvertent Safety Injection Actuation has occurred on Unit 1.

Which of the following is an adverse effect of allowing Safety Injection to continue that performing EOS-1.1A, Safety Injection Termination mitigates?

A. Emergency Core Cooling System Pump motor heating due to running for extended time periods at minimum flow.

B. Loss of Instrument Air to Containment will not allow the use of the normal Pressurizer Spray Valves to control Pressurizer Pressure.

C. Centrifugal Charging Pumps running in the Injection Mode will collapse the Pressurizer bubble and pressurize the RCS to the PORV setpoint.

D. Reactor Coolant Pumps will be running without adequate pump seal cooling due to seal water return containment isolation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that ECCS Pumps running at low flows would be the most significant affect.

B. Incorrect. Plausible because it could be thought that using Pressurizer Spray Valves could prevent an overpressure condition, however, the Pressurizer would continue to fill and pressurize the RCS until inventory was controlled.

C. Correct. The high head Centrifugal Charging Pumps will continue to increase inventory resulting in high pressures up to the PORV setpoint if steps to reduce flow and restore Letdown as part of SI Termination are not performed.

D. Incorrect. Plausible because the RCPs would be running with Seal Injection but Seal Return flow would be via the Seal Water Return Relief Valve to the PRT, which could lead to the conclusion that the relief to the PRT would provide inadequate seal cooling.

Technical Reference(s) OPGD 3, Attachment 6 Attached w/ Revision: See

Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from EOS-1.1, Safety Injection Termination, **STATE** the purpose/basis for the step(s).

Question Source: Bank ILOT8273
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: OPGD 3, Attachment 6	Revision: 06/13/13

1.1. Procedure Expediency

The Unit Supervisor's pace through ERGs / ABNs is always critical until the Reactor has been placed in a stable condition.
The Unit Supervisor should consider temporarily suspending expectations that may otherwise be in effect during ERG/ABN performance (e.g., Crew Briefings). any time delay in working through the procedures may contribute to the event severity. The following are examples of conditions associated with ERG/ABN performance.

- **Inadvertent SI**: To preclude overfilling the pressurizer with the increased risk of damaging the pressurizer safeties, the ERG network should be worked expeditiously to the point of re-establishing letdown (terminating the pressurizer fill.)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	007 A4.09	
Level of Difficulty: 3	Importance Rating	2.5	

Pressurizer Relief/Quench Tank System: Ability to manually operate and/or monitor in the control room: Relationship between PZR level and changing levels of the PRT and bleed holdup tank

Proposed Question: 32

Given the following conditions:

- Unit 1 is at 100% power.
- Reactor Coolant System Letdown flow is 132 gpm.
- DC power is lost to the solenoid valve for 1-8160, Letdown Containment Isolation Valve.

Which of the following describes the effect on Pressurizer, Volume Control Tank (VCT) and Pressurizer Relief Tank (PRT) level?

Pressurizer level will…

A. …decrease; VCT level will decrease; PRT level will remain the same.

B. …remain on program; VCT level will decrease; PRT level will increase.

C. …remain on program; VCT level will decrease; PRT level will remain the same.

D. …decrease; VCT level will remain the same; PRT level will increase.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the VCT level will decrease with no Letdown Return Flow. Pressurizer level will remain the same (on program), but a misconception could be that RCS fluid is being lost and Pressurizer level must be adversely affected. PRT level will increase, but could be easily missed if the operator forgets that the letdown relief valve will relieve the entire letdown flow to the PRT.
B. Correct. Closing 1-8160 will cause the letdown relief valve 1-8117 to lift to the PRT causing level to increase. The capacity of the valve is enough to pass all letdown flow; therefore, pressurizer will remain on program. With letdown isolated, VCT level will decrease.
C. Incorrect. Plausible because the Pressurizer will remain on program and VCT level will decrease. PRT level will increase, but could be easily missed if the operator forgets that the letdown relief valve will relieve the entire letdown flow to the PRT.
D. Incorrect. Plausible because the PRT level will increase. VCT level will decrease but with auto makeup it could be conceived that VCT level will remain the same, however, auto makeup cannot makeup at 132 gpm and VCT level will decrease. Pressurizer level will remain the same (on program), but a misconception could be that RCS fluid is being lost and Pressurizer level must be adversely affected.

Technical Reference(s): LO21.SYS.CS1, Pages 14 & 73; ALM-0061A, 1-ALB-6A, Window 4.3 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the components of the Chemical and Volume Control system including interrelations with other systems to include interlocks and control loops.

Question Source: Bank ILOT8080
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments / Reference: LO21.SYS.CS1 Page 14	Revision: 04/28/2011

LETDOWN RELIEF VALVE

Letdown Relief Valve u-8117 is located inside containment, downstream of the letdown orifice isolation valves. It lifts at 600 psig, directing flow to the pressurizer relief tank to limit letdown pressure. The valve is designed to protect against overpressure in the event both letdown isolation valves and all three letdown orifice isolation valves are fully open with the flow downstream of the orifices stopped. It is designed for 195 gpm at 600 psig. This capacity matches the combined design flow rate of all three orifices together.

Comments / Reference: LO21.SYS.CS1, Page 73	Revision: 04/28/2011

140 gpm. The reactor makeup system has a makeup capability of approximately 127 gpm. So, sustained operation with a leak which is just within charging capability is not feasible because of the limitations of the makeup system.

Comments / Reference: ALM-0061A, 1-ALB-6A, Window 4.3	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 65 OF 79

ANNUNCIATOR NOM./NO.: **LTDN RLF VLV OUT TEMP HI** **4.3**

PROBABLE CAUSE:

1-PK-131, LTDN HX OUT PRESS CTRL malfunction
1/1-8160, LTDN CNTMT ISOL VLV or 1/1-8152, LTDN CNTMT ISOL VLV malfunction
CVCS malfunction
High containment temperature
Safety Injection

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2014	Tier	2	
Change: 3	Group	1	
	K/A	008 K2.02	
Level of Difficulty: 2	Importance Rating	3.0	

Component Cooling Water System: Knowledge of bus power supplies to the following: CCW pump, including emergency backup

Proposed Question: 33

Given the following conditions:

- Unit 2 is in MODE 1.
- An XST1 Transformer fault has just occurred.
- All systems responded in accordance with design.

Which of the following is correct for Component Cooling Water Pump (CCWP) 2-02?

CCWP 2-02 is powered from…

A. …2EA1 which is supplied by XST2.

B. …2EA1 which is supplied by Emergency Diesel Generator 2-01.

C. …2EA2 which is supplied by XST2.

D. …2EA2 which is supplied by the Emergency Diesel Generator 2-02.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that CCWP 2-02 was powered from Train A as 2EA1 would be powered from XST2.
B. Incorrect. Plausible if thought that CCWP 2-02 was powered from Train A and that 2EA1 would be supplied by the EDG following the transformer fault.
C. Correct. The Train B CCWP would be powered from 2EA2, which would be supplied by the alternate offsite source following a fault of the preferred power supply. Following the slow transfer to XST1, the bus would automatically load the CCWP 2-02 on the bus. The EDG would not start and supply the bus if the slow transfer was successful.
D. Incorrect. Plausible as the proper power supply is listed, however, following the slow transfer to XST1 the bus would automatically load the CCWP 2-02 on the bus. The EDG would not start and supply the bus if the slow transfer was successful.

Technical Reference(s) LO21.SYS.CC1, Page 14
LO21.SYS.AC2, Page 12

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the 6.9 KV and 480 V Electrical Distribution System.

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: LO21.SYS.CC1, Page 14	Revision: 05/01/11

CCW PUMPS

The CCW pumps are located on the centerline of the Auxiliary Building, elevation 810'. They are 100% capacity, centrifugal, horizontal, double suction, single stage, motor-driven pumps with a nominal capacity of 14,700 gpm each at a head of 226 ft. The shafts have minimum leakage mechanical seals cooled by the discharge of the pump. The journal and thrust bearings are self-lubricated by oil rings.

The pumps are normally powered from uEA1 and uEA2. On a loss of power, they will be supplied from the train related emergency diesel generator. Control power for the pumps is from uED1-2 for Train A and uED2-2 for Train B.

Comments / Reference: LO21.SYS.AC2, Page 12	Revision: 04/28/11

Alternate Power Sources

XST1

The XST1 "X" winding supplies alternate power to Unit 1 Class 1E buses 1EA1 and 1EA2 during normal plant operations.

XST2

The XST2 "X" winding acts as the alternate power source to Unit 2 Class 1E buses 2EA1 and 2EA2 during normal plant operations.

Therefore, the Class 1E buses of each unit can be supplied by two independent and reliable immediate access offsite power sources. Sharing of these offsite power sources between the two units has no effect on the station electrical system reliability. Each transformer is capable of supplying the required safety-related loads of both units if it becomes necessary to safely shut down both units simultaneously.

Standby Power Sources

The Standby AC Power is provided by four Emergency Diesel Generators (EDGs) which supply Class 1E loads to ensure safe plant shutdown when preferred and alternate power sources are not available. Each EDG is capable of sequentially starting and supplying the minimum power requirements for a DBA in one unit. The four EDGs are electrically and physically independent.

In the event of a loss of the normal power source to the 6.9KV AC Safeguards bus (buses), a transfer to the alternate source will be initiated in addition to bus load shedding (slow transfer). If the transfer to the alternate source is successful, the respective DG will NOT start, and loads will be sequenced on to the bus powered by the alternate power supply. If the transfer to the alternate source is not successful, the respective DG will receive a start signal (1.0 second time delay following loss of power) and loads will be sequenced on to the bus supplied by the diesel.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	010 A3.02	
Level of Difficulty: 3	Importance Rating	3.6	

Pressurizer Pressure Control System: Ability to monitor automatic operation of the PZR PCS, including: PZR pressure

Proposed Question: 34

Given the following conditions:

- The Unit is at 100% power.
- Pressurizer Pressure is 2235 psig and stable.
- Handswitch 1/1-PS-455F, PRZR PRESS CTRL CHAN SELECT is in the 455/456 position.
- 1-PT-455A, Pressurizer Pressure Control Channel fails high.

Which of the following would be the PROGRESSION of events in the primary system (including Pressurizer Pressure Control System response), following the instrument failure?

1. Both PRZR Spray Valves (1-PCV-455B and 1-PCV-455C) CLOSE.
2. Both PRZR Spray Valves (1-PCV-455B and 1-PCV-455C) OPEN.
3. PORV 1-PCV-455A OPENS.
4. PORV 1-PCV-456 OPENS.
5. PORV 1-PCV-455A CLOSES.
6. PORV 1-PCV-456 CLOSES.
7. Reactor Trip on low Pressurizer pressure.
8. Reactor Trip on high Pressurizer pressure.

A. 1, 3, 4, 8

B. 2, 3, 5, 7

C. 3, 1, 6, 8

D. 4, 2, 5, 7

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because PORV PCV-455A will open, however, the Spray Valves will open and only PORV PCV-456 would remain closed.

B. Correct. Given Pressurizer pressure setpoint prior to the Control Channel failing high, the sequence of events is as listed.

C. Incorrect. Plausible because PORV PCV-455A will open, however, the Spray Valves will open not close, the Reactor will trip on low pressure not high pressure.

D. Incorrect. Plausible because the spray valves open, the Reactor trips on low pressure and PORV PCV-455A will close, however, not in the order listed and PORV PCV-456 remains closed.

Technical Reference(s): ABN-705, Step 2.2; LO21.SYS.PP1, Figure 4 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Pressurizer Pressure and Level Control System.

Question Source:

Bank	ILOT8526	
Modified Bank		(Note changes or attach parent)
New		

Question History: Last NRC Exam ______

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	7
55.43	

Comments / Reference: ABN-705, Step 2.2	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 12	PAGE 4 OF 26

2.2 Automatic Actions

NOTE: Control responses will only occur if failure occurs in a channel selected for control.

a. Control response for a pressurizer pressure channel failure HIGH.

1) PORV will open until pressure is reduced to 2185 psig, then the other channel will close the PORV.

- 1/u-PCV-455A, PRZR PORV
- 1/u-PCV-456, PRZR PORV

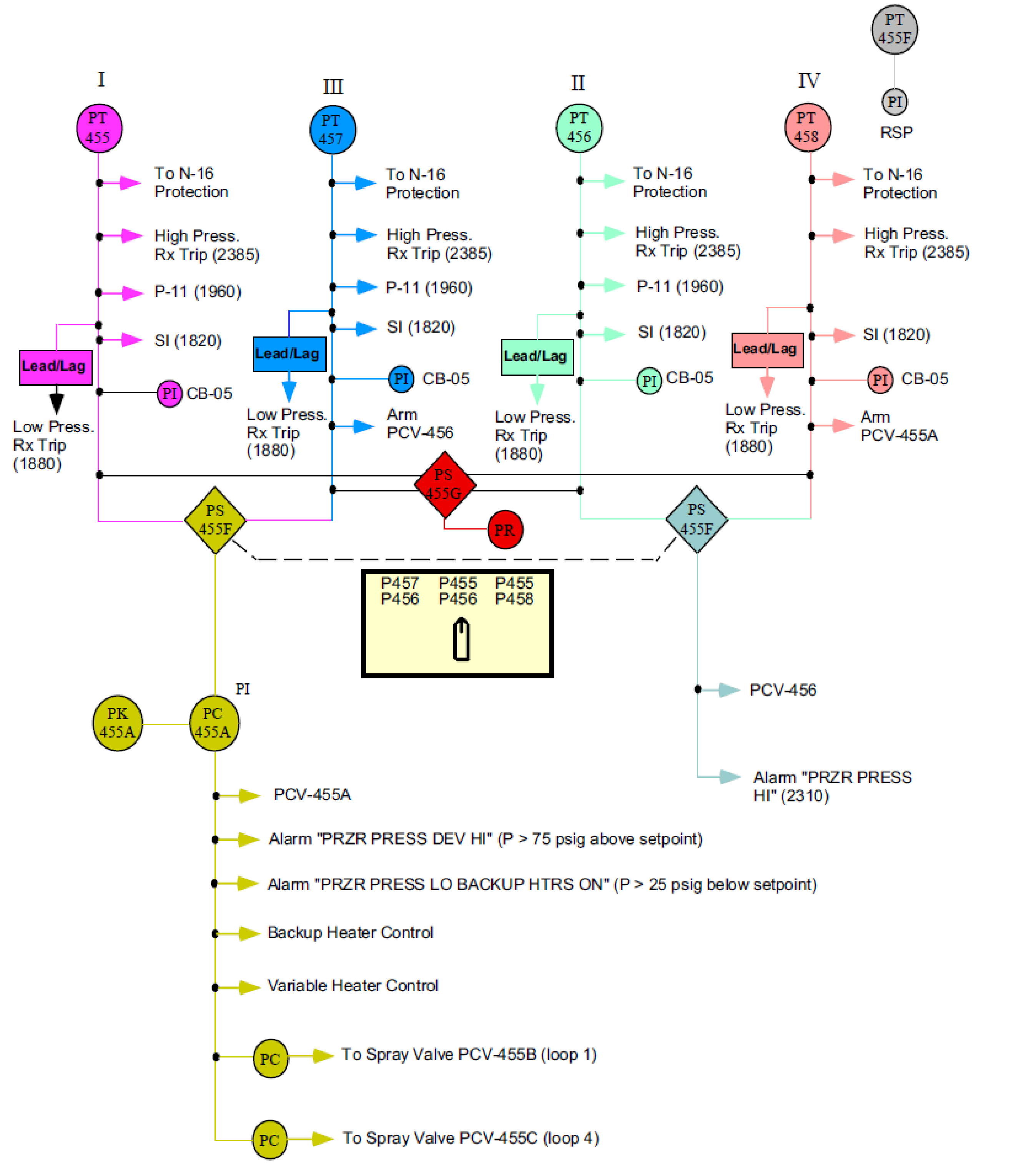
2) Variable heaters are turned off.

- 1/u-PCPR, PRZR CTRL HTR GROUP C

3) Both spray valves open.

- u-ZL-455B, RC LOOP 1 PRZR SPR VLV
- u-ZL-455C, RC LOOP 4 PRZR SPR VLV
- u-PK-455B, RC LOOP 1 PRZR SPR VLV CTRL
- u-PK-455C, RC LOOP 4 PRZR SPR VLV CTRL

Comments / Reference: LO21.SYS.PP1, Figure 4	Revision: 03/01/03



Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: 3/3/2014	Tier	2		
Change: 2	Group	1		
	K/A	012 G 2.1.32		
Level of Difficulty: 2	Importance Rating	3.8		

Reactor Protection System: Conduct of Operations: Ability to explain and apply system limits and precautions

Proposed Question: 35

With Train A of Solid State Protection System (SSPS) being tested with its Input Error Inhibit Switch in INHIBIT and its Mode Selector Switch in TEST, Train A Protection can have...

A. ...an auto Safety Injection actuation and an auto Reactor Trip.

B. ...a manual Safety Injection actuation and a manual Reactor Trip.

C. ...a manual Reactor Trip and NO Safety Injection actuation.

D. ...NO Reactor Trip and NO Safety Injection actuation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that placing the Input Error Inhibit Switch in INHIBIT would still allow an automatic Reactor Trip, however, the purpose of this switch is to prevent any external signals other than manual from being input into the Solid State Protection System. This condition includes an automatic Safety Injection (SI).
B. Incorrect. Plausible because a manual Reactor Trip is still available in this condition, however, with the Mode Selector Switch in TEST it prevents any Engineered Safety Feature (ESF) slave relay actuations such as SI.
C. Correct. As stated in the PRECAUTIONS of SOP-711A, Solid State Protection System, having the Mode Selector Switch in TEST while the Input Air Inhibit Switch is in INHIBIT blocks any ESF slave relay actuation such as Safety Injection. A manual Reactor Trip is still available in this condition.
D. Incorrect. Plausible because no SI actuation will occur because the Mode Selector Switch is in the TEST position, however, a manual Reactor Trip is still available in this condition.

Technical Reference(s): SOP-711A, Section 3.0 & Step 4.2.2; OPT-447A, Steps 5.1.5 & 5.1.6 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Solid State Protection System and **PREDICT** the system response.

Question Source:

Bank	ILOT5653	
Modified Bank		(Note changes or attach parent)
New		

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: SOP-711A, Section 3.0	Revision: 9

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-711A
SOLID STATE PROTECTION SYSTEM	REVISION NO. 9 / CONTINUOUS USE	PAGE 5 OF 96

2.2 General Warning Test Light Verification - Train B to a Slave Relay Test or Disabled Lineup

CAUTION: The following step is performed in the OPPOSITE train cabinet (TRAIN A SSPS Cabinet) to ensure Train A is in a Normal Lineup.

2.2.1 PERFORM the following step at the OPPOSITE train cabinet (Train A SSPS):

☐ A. VERIFY Train A SSPS is in a Normal Lineup per Attachment 7.1.3 OR place Train A SSPS in a Normal Lineup per the applicable section of this procedure.

B. VERIFY BOTH GENERAL WARNING TEST lights are ON:

☐ • TRAIN A (green light)

☐ • TRAIN B (amber light)

CAUTION: The following steps are performed in TRAIN B SSPS Cabinet to ensure Train B is in a Normal Lineup with NO GENERAL WARNING ALARM.

2.2.2 PERFORM the following steps at the Train B SSPS cabinet:

☐ A. VERIFY GENERAL WARNING lamp (1-ALB-6D 2.5) is OFF.

B. VERIFY BOTH GENERAL WARNING TEST lights are ON:

☐ TRAIN A (amber light)

☐ TRAIN B (green light)

3.0 PRECAUTIONS

- Except when the core is off-loaded, both trains of SSPS shall not be tested or deenergized simultaneously.
- Removal of either Train of SSPS from the Normal Lineup during MODES 1, 2, 3, and 4 or from either the MODE 5/6 Lineup or Normal Lineup during MODES 5 and 6 will place the plant in LCO 3.3.1 and 3.3.2, as applicable. (Reference ODA-308-3.3.1-S01).
- When operating with one train INPUT ERROR INHIBIT switch in INHIBIT, verify both MULTIPLEXER TEST switches are in NORMAL to prevent cycling the alarm relays.
- The INPUT ERROR INHIBIT in INHIBIT will not prevent a Reactor Trip. It will only prevent any external signals other than manual from being input into SSPS.
- The MODE SELECTOR switch in TEST will not prevent a Reactor Trip, however it will prevent Slave Relay actuations (except if S604 is in MODE 5/6).

Comments / Reference: SOP-711A, Step 4.2.2	Revision: 9

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-711A
SOLID STATE PROTECTION SYSTEM	REVISION NO. 9 CONTINUOUS USE	PAGE 7 OF 96

4.2 Notes, continued

4.2.2 With the S604 switch in NORMAL all ESF Actuation System slave relays are defeated by placing the MODE SELECTOR switch in TEST.

4.2.3 The MODE 5 & 6 position on the S604 switch allows slave relays for CVI to remain operable when the MODE SELECTOR switch is placed in TEST. Therefore, S604 shall always be placed in MODE 5 & 6 prior to placing the MODE SELECTOR switch in TEST. Additionally, S604 shall be placed in NORMAL after the MODE SELECTOR switch is returned to OPERATE except as noted in 4.2.6.

Comments / Reference: OPT-447A, Steps 5.1.5 & 5.1.6	Revision: 10

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-447A
MODE 1, 3 AND 4 TRAIN A SSPS ACTUATION LOGIC TEST	REVISION NO. 10 CONTINUOUS USE	PAGE 5 OF 55

5.1.4 If conduct of this test is interrupted while performing Section 8.0 of this procedure, proceed to Section 9.0, "Restoration/Post Work Activities".

5.1.5 The MODE SELECTOR switch in TEST will not prevent a Reactor Trip, however it will prevent Slave Relay actuations (except CVI if S604 is in MODES 5&6).

5.1.6 The INPUT ERROR INHIBIT switch in INHIBIT will not prevent a Reactor Trip. It will only prevent any external signals other than manual signals and CVI from being input into SSPS.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 3	Group	1	
	K/A	013 K3.01	
Level of Difficulty: 3	Importance Rating	4.4	

Engineered Safety Features Actuation System: Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Proposed Question: 36

Given the following conditions:

- Unit 1 has experienced a 1 inch Small Break Loss of Coolant Accident.
- A Loss of Offsite Power has occurred.
- Bus 1EA1 has experienced an 86-1 lockout.
- Station Service Water Pump 1-02 has tripped.
- Reactor Coolant System (RCS) subcooling is currently 15°F and trending toward 0°F.

If the operators are unable to depressurize the RCS to less than 650 psig, what are the expected consequences of the current plant conditions?

A. The Accumulators will inject and in combination with the Residual Heat Removal Pumps will provide adequate cooling for the reactor core.

B. Natural Circulation will continue to provide adequate cooling as long as SG levels are maintained between 50 and 60% narrow range.

C. The Reactor Coolant Pumps will continue to provide forced cooling of the RCS as long as SG levels are greater than 35% wide range.

D. Continued Reactor Coolant System inventory loss will eventually result in loss of adequate core cooling and fuel damage is expected.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because if the RCS is depressurized to the point where the accumulators inject, RHR injection is also going to occur. However, the stem states that the RCS cannot be depressurized by the operators to 650 psig which is above the accumulator injection pressure. If the RCS is not depressurized to the accumulator injection pressure adequate cooling will not be provided by the accumulators and RHR.

B. Incorrect. Plausible because if a secondary heat sink is maintained, adequate heat removal will also occur as long as the loop remains filled to transport the heat from the core to the steam generators, however, without makeup the loops will eventually drain and heat removal via the steam generators will be ineffective no matter what level remains in the steam generators.

C. Incorrect. Plausible because the RCPs can be restarted during inadequate core cooling to provide some cooling to the core. With the current plant conditions the RCPs should be stopped and would not be restarted even during inadequate core cooling unless a steam generator level of at least 43% narrow range existed. The steam generator levels provided are the minimum for delay of initiating bleed and feed for a loss of heat sink.

D. Correct. Given the degraded RCS and the inability to depressurize the RCS by the operators fuel damage is expected.

Technical Reference(s) LO21.MCO.MI2, Pages 10, 15, & 16 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** plant conditions to **DETECT** conditions which could lead to core damage from a lack of adequate core cooling and **DETERMINE** the appropriate mitigation strategies consistent with the Functional Restoration Guidelines.

Question Source: Bank ______
Modified Bank ILOT5731 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments / Reference: LO21.MCO.MI2, Page 10	Revision: 04/30/04

Suppose that a 1-inch cold leg break occurs after the plant has established a very high power history (high decay heat levels are present). Suppose also that both the SG Atmospheric relief valves (ARVs) and the steam dump valves are inoperable, and only one train of ECCS is available. The plant begins to depressurize; a reactor trip, turbine trip, and SI occur. More subcooled liquid volume leaves the break than is being added. The pressurizer continues to empty and pressure steadily drops. As the pressurizer empties, the rate of pressure decrease suddenly rises since a relatively small quantity of vapor is produced from surge-line flashing. Eventually, saturation conditions are reached in the hot legs and vessel outlet plenum and soon after in the entire primary. The break is small and is unable to remove all the decay heat. Thus, the break is insufficient to continue depressurizing the plant. Another heat sink must function to prevent the plant from reaching an over temperature condition. Using conservative assumptions, the SG safety valves are the only available heat sink. Assume that the lift pressure of the lowest safety valve is 1200 psia, which corresponds to a SG saturation temperature of about 567.2 °F.

The corresponding primary side temperature is about 567.8 °F. Since the RCS is saturated, this corresponds to a primary system pressure of about 1205 psia. The plant does not depressurize below this point as long as the safety valves are needed as a heat sink. Under the worst-case conditions, the plant may remain in this state for up to one day on a 1-inch break. Therefore, heat removal, limits plant depressurization. With primary pressure remaining high, volume flow out of the break exceeds SI flow (with plant pressure and temperature stabilized). This situation can occur when the combination of break volume removal and SG safety valve thermal energy removal is sufficient to terminate any plant pressure rise, but not sufficient to totally remove all decay heat, i.e., heat in exceeds heat out. Extended operation in this manner would obviously be undesirable, since the system mass inventory would eventually deplete. As it is, the vapor volume of the coolant system continuously rises and eventually (due to the decay heat drop-off) the reactor coolant pressure and temperature begins to decrease and SI flow increases. As long as SI flow is NOT terminated, adequate core cooling is maintained. In any case, the design of the ECCS must be such that, in the long run, it can inject more water into the RCS than is being lost out of the break.

Comments / Reference: LO21.MCO.MI2, Pages 15 & 16	Revision: 04/30/04

1-INCH BREAK WITH DEGRADED ECCS

Assumptions

Assumptions remain the same as in the 4 inch analysis. Basically, a 1 inch cold leg break occurs in a four loop plant at 100 percent power with no high head ECCS and no accumulators. The steam dumps remain available because off site power remains available. However, the RCPs trip during the accident.

Analysis (Figures 8, 9 and 10)

When the break occurs, RCS pressure inventory decrease. At 143 seconds, the low RCS pressure actuates reactor and turbine trips. The steam dumps actuate and reduce RCS temperature to no load. At 376 seconds, the RCPs trip. The decreased core flow increases core fluid enthalpy and void fraction. As the RCPs coastdown, forced flow decreases. The increased core enthalpy increases natural circulation flow. At 550 seconds, the increased void fraction starts two phase natural circulation. 1000 seconds into the event, the RCS pressure stabilizes based on the heat generation and removal relationship.

At 1500 seconds, the operators place the steam dump in pressure control mode. In addition, sufficient void fraction occurs in the RCS loops that the SGs cannot condense all of the steam. So the cold side of the U tubes increases in void fraction. Core mixture level starts to decrease from the inventory loss. As the core mixture level approaches the hot leg nozzles, the rate of level decrease slows because the drainage from the U tubes makes a significant difference.

At 4800 seconds, two phase natural circulation stops because of the inability of the SGs to condense the steam. At that time, reflux core cooling starts. In reflux, steam in the U tubes condenses. Steam from the vessel passes along the top of the hot leg, condenses in the U tubes, and returns as condensate back to the vessel along the bottom of the hot leg. Condensate in the cold side drains into the loop seal, which pulses water into the vessel downcomer. Although less efficient than natural circulation, reflux transfers heat to the SGs and cools the core.

At 7800 the core mixture level decreases below the flow holes in the guide tubes of the reactor upper internals. When the level uncovers the holes, the vessel steam enters the guide tubes, passes up the tubes, enters the head, and then passes backward through the head cooling holes in the core barrel flange. The steam then enters the cold leg and passes out the break. Similar to the clearing of the loop seal, this provides a direct vent from the reactor. The RCS pressure and core mixture levels begin to decrease. Reflux flow slows, and core mixture level starts another decrease at about 8200 seconds. Core mixture level continuously decreases below the top of the fuel and down to the bottom of the fuel.

At 9300 seconds, the upper core uncovers, decreasing decay heat transferred to the fluid. The fuel and clad retain more of the heat, increasing their temperatures. Heat removal from the coolant now exceeds heat transferred into the coolant, decreasing RCS pressure. When RCS pressure decreases, RCS temperature decreases below SG temperature. This slows reflux flow, which was adding liquid to the vessel. So core mixture level and RCS pressure decrease faster.

Comments / Reference: LO21.MCO.MI2, Pages 15 & 16	Revision: 04/30/04
The fluid temperature basically follows RCS pressure once saturation occurs early in the accident. The fluid temperature breaks away from RCS pressure at 10,250 seconds, which is after the core uncovery starts and after the upper core uncovers at 9300 seconds. For the period between core uncovery and fluid temperature increase, a stabilized fluid temperature occurs because of the reflux flow. The reflux flow, limited as it is, cools the upper core fluid. However, at 10,250 seconds, the center of the core uncovers. The steam produced in the core increases in temperature as it passes over the upper core, producing superheated steam. The superheated steam boils the reflux water, preventing core cooling and increasing the core fluid temperature. At 11,000 seconds, the upper core fluid temperature exceeds 1200 °F. With no ECCS flow, core uncovery and the core fluid temperature increase continues. At 12,000 seconds, RCS pressure still remains higher than the low head ECCS shutoff head.	

Original Question: CPNPP Exam Bank ILOT5731

For a small break LOCA (1 inch) where high head safety injection is not available, what is the expected plant condition with no operator action?

A. Low head safety injection will mitigate the LOCA and no core damage will result.

B. The core will remain covered but slight core damage will result.

C. The core will briefly uncover and slight core damage will result.

D. The core will completely uncover and severe core damage will result.

Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 3	Group	1	
	K/A	022 K4.03	
Level of Difficulty: 2	Importance Rating	3.6	

Containment Cooling System: Knowledge of the CCS design feature(s) and/or interlock(s) that provide for the following: Automatic containment isolation

Proposed Question: 37

Which of the following explains the expected condition of the Containment Ventilation Systems after a Safety Injection actuation?

A. All Containment Ventilation Fans will be stopped and Ventilation Chilled Water isolated since these systems are NOT required during an accident.

B. Only the Neutron Detector Well and Containment Fan Coolers will start or continue to run and Ventilation Chilled Water will NOT be isolated. This will ensure adequate cooling for the Intermediate and Source Range Excore Detectors.

C. Only the Neutron Detector Well fans will start or continue to run and ventilation chilled water will NOT be isolated. This will ensure adequate cooling for the Intermediate and Source Range Excore Detectors.

D. All Containment Ventilation Fans and Ventilation Chilled Water System are NOT affected since they may be required for accident conditions.

Proposed Answer: A

Explanation:

A. Correct. Following a Safety Injection signal, all Containment ventilation fans will be stopped and ventilation chilled water will be isolated by Containment Phase A Isolation.
B. Incorrect. Plausible because these fans receive start signals and are automatically loaded on the safeguards bus following a Blackout. The misconception that cooling to the excore detectors used during the Emergency Operating Procedures would be maintained is plausible.
C. Incorrect. Plausible because these fans receive start signals and are automatically loaded on the safeguards bus following a Blackout. The misconception that cooling to the excore detectors used during the Emergency Operating Procedures would be maintained is plausible.
D. Incorrect. Plausible because maintenance of as much containment cooling as possible would be practical since CPNPP does not have Safety grade containment coolers. Maintaining other coolers when possible would alleviate harsh containment conditions.

Technical Reference(s) LO21.SYS.CL1, Pages 15 & 16
EOP-0.0A, Attachment 9

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the

Containment Ventilation System.

Question Source:	Bank	ILOT4191	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	9	
	55.43		

Comments / Reference: LO21.SYS.CL1, Page 16	Revision: 05/02/11
In the event of a Safety Injection, Containment ventilation isolation is verified on both units. The Safety Injection will trip all the Primary Plant Exhaust Fans and start the four (4) ESF Exhaust Fans. This configuration does not provide enough airflow to maintain a negative pressure in the Safeguards and Auxiliary Buildings if a Containment purge is in progress on the unaffected unit. A Containment Ventilation Isolation signal will cause the valves or isolation dampers on the affected unit to close.	

Comments / Reference: LO21.SYS.CL1, Page 15	Revision: 05/02/11
Three of the Containment Air Cooling and Recirculation Fans are normally in service. If the average Containment temperature can not be maintained at approximately 100EF, the fourth unit will be started. A Safety Injection signal will trip all the running fans and the breakers can not be reset until the Safety Injection signal is reset. The Phase A Isolation signal will isolate the Ventilation Chilled Water and instrument air to Containment. Isolating the instrument air will cause the air operated dampers to fail open. The blackout sequencer will start the fans when power is restored to the bus. One of the Neutron Detector Well Cooling Systems will be in service during normal operations. The standby system may be started if detector well exhaust temperatures cannot be maintained below 150°F. When the fan is started, the associated ventilation chilled water isolation valve opens. A safety injection signal will trip the fans. The blackout sequencer will start the fans when power is restored to the bus.	

Comments / Reference: EOP-0.0A, Attachment 9	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 67 OF 117

ATTACHMENT 9
PAGE 8 OF 20

POST EVENT SYSTEM REALIGNMENT

NOTE: Unit 1 AND Unit 2 Containment cooling is normally supplied by Ventilation Chillers X-01 through X-06 common supply. The potential exists to affect Unit 2 cooling during restoration of Unit 1 Containment cooling. Only Ventilation Chillers X-01 through X-04 operate following Loss of Offsite Power.

NOTE: Prior to aligning Ventilation Chilled Water to Containment, contact Plant Staff to assess the potential for flashing to steam and water hammer of the piping inside Containment. Chilled Water to Containment should not be aligned if flashing/water hammer is expected to occur due to high containment temperatures.

4. Restore Containment cooling to service by performing the following steps:

☐ a. Notify Unit 2 Control Room of restoration of Unit 1 Containment Cooling AND to monitor Unit 2 Containment parameters.

b. Verify adequate Ventilation Chiller cooling source is aligned.

Comments / Reference: EOP-0.0A, Attachment 9	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 69 OF 117

ATTACHMENT 9
PAGE 10 OF 20

POST EVENT SYSTEM REALIGNMENT

4.d. ☐ 2) Place the remaining two recirculation pump handswitches in the AUTO AFTER STOP position.

- Chilled Water Circulating Booster Pumps X-12 through X-14 are from non-safeguards power supplies and do not receive a trip signal from either unit's SI signal.

☐ 1) Locally ensure two (2) Chilled Water Booster Pumps are running.

☐ e. Dispatch operator to start desired number of Ventilation Chillers per SOP-814, VENTILATION CHILLED WATER SYSTEM.

f. Align Containment ventilation fans by performing the following steps:

1) Start at least three (3) Containment Fan Cooler Fans.

- ☐ 1-HS-5405A, CNTMT FN CLR FN 1
- ☐ 1-HS-5409A, CNTMT FN CLR FN 2
- ☐ 1-HS-5413A, CNTMT FN CLR FN 3
- ☐ 1-HS-5417A, CNTMT FN CLR FN 4

2) Start at least one (1) Control Rod Drive Mechanism Vent Fan.

- ☐ 1-HS-5421, CRDM VENT FN 1
- ☐ 1-HS-5423, CRDM VENT FN 2

3) Start at least one (1) Neutron Detector Well Fan Cooler Unit.

- ☐ 1-HS-5435, NEUT DET WELL FN CLR FN 9 & DMPR
- ☐ 1-HS-5440, NEUT DET WELL FN CLR FN 10 & DMPR

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2014	Tier	2	
Change: 3	Group	1	
	K/A	026 A1.06	
Level of Difficulty: 4	Importance Rating	2.7	

Containment Spray System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment spray pump cooling

Proposed Question: 38

Given the following conditions:

- A Large Break Loss of Coolant Accident has occurred on Unit 2.
- Both Trains of Containment Spray have just actuated.
- Train B Component Cooling Water Pump has tripped.

Which of the following identifies the condition of Train B Containment Spray Pump (CSP) cooling?

Cooling has been lost to the CSP…

A. …bearing coolers and CSP motor cooling is degraded.

B. …bearing coolers and CSP motor cooling remains normal.

C. …seal coolers and CSP motor cooling is degraded.

D. …seal coolers and CSP motor cooling remains normal.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible since CCW supplies cooling to the pump seal coolers, but SSW supplies cooling to the pump bearing coolers, therefore cooling has not been lost to the bearing coolers. CSP motor cooling is degraded as the Safety Chiller which supplies chilled water for the room coolers will trip when the CCW Pump trips, thus the chilled water flowing to the room coolers will steadily increase in temperature resulting in degraded cooling to the motors which are cooled via the room coolers.

B. Incorrect. Plausible since CCW supplies cooling to the pump seal coolers, but SSW supplies cooling to the pump bearing coolers, therefore cooling has not been lost to the bearing coolers. CSP motor cooling is degraded as the Safety Chiller which supplies chilled water for the room coolers will trip when the CCW Pump trips, thus the chilled water flowing to the room coolers will steadily increase in temperature resulting in degraded cooling to the motors which are cooled via the room coolers. CSP motor cooling remaining normal is plausible as a chain of events must be considered to determine that room cooling is being lost from the lost of the safety chillers.

C. Correct. CCW supplies cooling to the CSP seal coolers which has been lost with the trip of the CCW pump. CSP motor cooling is degraded as the Safety Chiller which supplies chilled water for the room coolers will trip when the CCW Pump trips, thus the chilled water flowing to the room coolers will steadily increase in temperature resulting in degraded cooling to the motors which are cooled via the room coolers.

D. Incorrect. Plausible since CCW supplies cooling to the CSP seal coolers which has been lost with the trip of the CCW pump. CSP motor cooling is degraded as the Safety Chiller which supplies chilled water for the room coolers will trip when the CCW Pump trips, thus the chilled water flowing to the room coolers will steadily increase in temperature resulting in degraded cooling to the motors which are cooled via the room coolers. CSP motor cooling remaining normal is plausible as a chain of events must be considered to determine that room cooling is being lost from the lost of the safety chillers.

Technical Reference(s): SOP-204B, Step 2.2.2; LO21.SYS.CT1 page 11; LO21.SYS.CH1 page 12 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Containment Spray system including interrelations with other systems to include interlocks and control loops.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 8
55.43 ____

Comments / Reference: SOP-204B, Step 2.2.2	Revision: 6

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. SOP-204B
CONTAINMENT SPRAY SYSTEM	REVISION NO. 6 CONTINUOUS USE	PAGE 5 OF 45

2.2.2 Filling Train B

- The following valve lineups are complete:
  - ☐ SOP-204B-CT-V02, RWST Valve Lineup
  - ☐ SOP-204B-CT-V03, Chem Add Tank Valve Lineup
  - ☐ SOP-204B-CT-V04, Train B Valve Lineup
- ☐ The RWST is ≥25% AND aligned to the SI header.
- ☐ Demin Water is available in Containment for Spray Riser fill.
- ☐ The control switch lineup per SOP-204B-CT-C04, Train B Fill and Vent Control Switch Lineup is complete.

NOTE:	CCW flow to the Containment Spray Pumps and Heat Exchanger is not required provided system temperature is maintained ≤150°F and the affected train is declared inoperable per TS 3.6.6.

- ☐ CCW is available and aligned to the pump seal coolers.
- ☐ SSW is available and aligned to the pump bearing coolers.

Comments / Reference: LO21.SYS.CH1 Page 12	Revision: 3/1/2011

Chiller Trips

The chiller trips are divided into two categories: those that require a reset and those that do not require reset. The following trips require a reset after the signal clears:

- LO oil pressure
- LO evaporator pressure
- HI condenser pressure
- HI motor temperature
- HI discharge temperature
- Motor overload
- HI oil temperature

The trips that **DO NOT** require a reset and will allow an automatic restart of the compressor as soon as the signal clears include:

- LO chilled water temperature
- LO CCW flow
- LO chilled water flow

Comments / Reference: LO21.SYS.CT1 Page 11	Revision: 5/2/2011

CONTAINMENT SPRAY PUMPS (FIGURE 8)

Four Containment Spray Pumps are provided for each unit (two per train) and are of the horizontal, double-suction, centrifugal type. These pumps are designed to provide sufficient flow into Containment during accident conditions to both cool the Containment atmosphere and remove radioactive iodine. Each of the two pumps per train is 50% pumps and have a capacity of approximately 3000gpm at 260 psig discharge pressure. The pumps are designed to operate over a widely varied temperature ranging from 40□F (minimum RWST temperature) to 280□F (containment maximum design temperature).

Each of the pumps has separate bearing coolers (2) and seal coolers (2) mounted at the pump skid itself. The bearing coolers are supplied with cooling water from the related train of Station Service Water. Component Cooling Water from the same train supplies the seal coolers. ABN-501 provides alternate means of supplying cooling to the pump bearing coolers. Each set of train related pumps is located in a separate room in the safeguards building 773' elevation.

The pumps are driven by horizontal 700 hp electric motors powered from their respective 6.9 kv safeguards bus. The room/motors are kept cool by two room coolers supplied from their train related Safety Chill Water system. Both room coolers receive a start signal when either pump breaker closes. The pump breaker closing also provides a computer input and MLB (MLB-4A-1/4B-1) input for pump running indication.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	039 K5.05	
Level of Difficulty: 3	Importance Rating	2.7	

Main and Reheat Steam System: Knowledge of the operational implications of the following concepts as they apply to the MRSS: Bases for RCS cooldown limits

Proposed Question: 39

Given the following conditions:

- Unit 1 has experienced a Reactor Trip.
- A plant cooldown to MODE 5 is planned.

Which of the following describes the impact of Steam Dump Valve design on a Reactor Coolant System (RCS) cooldown?

The design of the Steam Dump Valves is such that the RCS cooldown limit of…

A. …100°F/hr is prevented during a cooldown.

B. …200°F/hr is prevented during a cooldown.

C. …100°F/hr is NOT prevented during a cooldown.

D. …200°F/hr is NOT prevented during a cooldown.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the RCS cooldown limit is 100°F/hr, however, the Steam Dump valves are NOT sized to prevent exceeding this limit and may be used in a SGTR to obtain a maximum rate cooldown. A misconception may exist that the Steam Dump valves are sized to preclude exceeding this limit.

B. Incorrect. Plausible because the Pressurizer cooldown limit is 200°F/hr and a misconception could exist that the Steam Dump valves are sized to prevent exceeding this limit.

C. Correct. The RCS cooldown limit is 100°F/hr.; however, the Steam Dump valves are NOT sized to prevent exceeding this limit and may be used in a SGTR to obtain a maximum rate cooldown.

D. Incorrect. Plausible because the Pressurizer cooldown limit is 200°F/hr and the Steam Dump valves are NOT sized to prevent exceeding the cooldown limit.

Technical Reference(s) LO21.SYS.MR1, Page 57
Technical Specification LCO 3.4.3 (PTLR)
Technical Requirements Manual TR 13.4.34

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Main Steam System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 4
55.43 ______

Comments / Reference: LO21.SYS.MR1, Page 57	Revision: 06/09/11

Plant Cooldown

Following plant shutdown, it may be desired or necessary to perform a plant cooldown for entry into a different operational mode (see Technical Specifications, Section 1.1 - Definitions). A normal plant cooldown will be conducted using IPO-005 "Plant Cooldown from Hot Standby to Cold Shutdown."

The preferred method of plant cooldown is performed using Steam Dumps in the "Steam Pressure" mode and controlled by u-PK-507 in manual. Extreme care must be exercised when opening the Steam Dumps to prevent an inadvertent steam line low pressure Safety Injection (above P-11) or Main Steam Line Isolation (if SI is blocked below P-11). Actual cooldown rate is dictated by operational need, but is procedurally limited to < 100˚F/hr at a uniform rate. Cooling down at a uniform rate is preferred, to a stepwise temperature decrease, to minimize the effects on Reactor Vessel integrity. Upon reaching the LO-LO RCS Tave (P-12) setpoint (553˚F), the steam dump system is placed in Bypass Interlock position to allow the plant to be cooled down below 553˚F using only three cooldown valves.

CPNPP Technical Specifications section 3.4.3, states "RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR." The "Pressure and Temperature Limits Report" (PTLR) specifies the maximum heatup or cooldown rate shall not exceed 100˚F in any one hour period.

Comments / Reference: Technical Specification LCO 3.4.3 (from PTLR)	Amendment: 161

2.1 RCS Temperature Rate-of-Change Limits (LCO 3.4.3)

2.1.1 Maximum Heatup Rate

The RCS heatup rate limit is 100°F in any 1-hour period.

2.1.2 Maximum Cooldown Rate

The RCS cooldown rate limit is 100°F in any 1-hour period.

2.1.3 Maximum Temperature Change During Inservice Leak and Hydrostatic Testing

During inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves, the RCS temperature change limit is 10°F in any 1-hour period.

Comments / Reference: Technical Requirements Manual TR 13.4.34	Revision: 84

Pressurizer
TR 13.4.34

13.4 REACTOR COOLANT SYSTEM

TR 13.4.34 Pressurizer

TR LCO 13.4.34 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1 hour period, and

b. A maximum cooldown of 200°F in any 1 hour period.

APPLICABILITY: At all times.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2014	Tier	2	
Change: 3	Group	1	
	K/A	059 A2.04	
Level of Difficulty: 4	Importance Rating	2.9	

<u>Main Feedwater System</u>: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry SG

Proposed Question: 40

Given the following conditions:

- Unit 1 has experienced a Loss of Heat Sink.
- Bleed and Feed cooling was established when all Steam Generators (SG) had insufficient level.
- Main Feedwater flow has been established.
- Reactor Coolant System Hot Leg temperatures are all indicating 560°F and STABLE.
- Core Exit Thermocouple temperatures are all indicating between 555°F and 565°F and STABLE.
- ALL Steam Generator Wide Range levels are 0%.

Which of the following actions is required in accordance with FRH-0.1A, Response to Loss of Secondary Heat Sink?

Establish Main Feedwater flow to...

A. ...ONE Steam Generator at a rate not to exceed 100 gpm.

B. ...ALL Steam Generators not to exceed 100 gpm per SG.

C. ...ONE Steam Generator at maximum available feed flow.

D. ...ALL Steam Generators at maximum available feed flow.

Proposed Answer: A

Explanation:

A. Correct. IAW Step 26, Table 1 of FRH-0.1A, Main Feedwater should be established to only one SG at a rate not to exceed 100 gpm.

B. Incorrect. Plausible because this is the proper feed rate and the misconception that reestablishing cooling in all loops is credible.

C. Incorrect. Plausible because IAW Step 26, Table 1 of FRH-0.1A, Main Feedwater would be established to only one SG at maximum rate if the RCS temperature was increasing, indicating a more critical situation.

D. Incorrect. Plausible because the maximum rate would be used if RCS temperature was increasing and the misconception that reestablishing cooling in all loops is credible.

Technical Reference(s): FRH-0.1A, Step 26.b, Table 1; FRH-0.1A, Attachment 4, Step 26 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1, Response to a Loss of Secondary Heat Sink.

Question Source:
- Bank: ____
- Modified Bank: ____ (Note changes or attach parent)
- New: X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge: ____
- Comprehension or Analysis: X

10 CFR Part 55 Content:
- 55.41: 10
- 55.43: ____

Comments / Reference: FRH-0.1A, Step 26.b, Table 1	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 22 OF 60

STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

b. Check for Hot, Dry SG and establish feed flow:

1) Evaluate the following parameters when feed flow capability restored AND perform appropriate actions from Table 1:

- SG wide range level
- Core exit TCs
- RCS hot leg temperatures

TABLE 1

SG WIDE RANGE LEVEL	RCS TEMPERATURE	SG FEED FLOW LIMITATIONS
ALL SGs LESS THAN 14% (19% FOR ADVERSE CONTAINMENT)	INCREASING	• Establish maximum available feed flow to ONE SG. • WHEN selected SG wide range level greater than 14% (19% FOR ADVERSE CONTAINMENT), THEN adjust feed flow as necessary to establish narrow range level.
	STABLE OR DECREASING	• Establish feed flow to ONE SG at a rate not to exceed 100 gpm. • WHEN selected SG wide range level greater than 14% (19% FOR ADVERSE CONTAINMENT), THEN adjust feed flow as necessary to establish narrow range level.

Comments / Reference: FRH-0.1A, Attachment 4, Step 26 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 50 OF 60

ATTACHMENT 4
PAGE 16 OF 26

BASES

STEP 26: Re-establishment of the secondary heat sink will permit termination of the bleed and feed heat removal method and establish stabilized plant conditions.

Attempts to establish secondary heat sink in at least one SG may have been initiated in previous steps before initiation of bleed and feed heat removal. These attempts should be continued using the actions specified in Steps 5, 7, and 9 until a secondary heat sink is restored; therefore, this step has been identified as a continuous action step.

If bleed and feed has been initiated, during restoration of secondary heat sink, feeding a dry steam generator may be necessary. If the event was initiated from high temperature and high decay heat conditions it is likely that feedwater flow will have to be established to a hot, dry steam generator. A hot, dry steam generator is defined as a steam generator in which the primary side of the steam generator is above 550°F (550°F is a temperature evaluated to be low enough that thermal stress would not lead to a failure when feedwater is established to any remaining dry steam generator.) and the secondary side has no liquid inventory. (Indicated SG level less than SG wide range level setpoints identified in this step.) Reestablishment of feedwater is the more desirable mode of recovery from a loss of secondary heat sink than remaining on bleed and feed and establishing cold leg recirculation for long term cooling because this will be more likely to avoid core uncovery. However, care must be taken when re-establishing feedwater flow to minimize the effects of thermal shock consistent with the urgency of the need to restore the secondary side heat sink.

Since the heat removal capability of one steam generator is always greater than decay heat, it is advisable to reestablish feedwater to only one steam generator. Thus, if a failure in a SG occurs due to excessive thermal stresses, the failure is isolated to one steam generator.

If bleed and feed has been initiated and RCS temperature is increasing, the re-establishment of feedwater flow should be limited to one steam generator and the flow rate used should be as high as can be made available due to the urgency of the situation. If RCS temperatures are stable or decreasing when feedwater flow is restored the flow should be directed to one steam generator and the rate should be limited to 100 gpm until wide range level is established.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	061 K6.02	
Level of Difficulty: 3	Importance Rating	2.6	

Auxiliary/Emergency Feedwater System: Knowledge of the effect that a loss or malfunction of the following will have on the AFW components: Pumps

Proposed Question: 41

Given the following conditions:

- Unit 1 is in a condition requiring Turbine Driven Auxiliary Feedwater Pump (TDAFWP) operation.
- The TDAFWP has tripped due to an overspeed condition.
- The cause of the overspeed trip has been corrected.
- The Safeguards Building NEO has been directed to reset 1-HV-2452 AFWPT 1-01 TRIP AND THROT VLV, in accordance with ABN-305, Turbine Driven Auxiliary Feedwater Pump Malfunction.
- The actions of ABN-305 have been completed prior to the NEO resetting the valve.

Which of the following describes the response of the TDAFWP as the NEO opens 1-HV-2452 in accordance with ABN-305?

The TDAFW Pump Turbine will…

A. …remain at approximately 0 rpm and speed will have to be increased by the Reactor Operator.

B. …increase speed to approximately 2000 rpm and hold speed.

C. …remain at approximately 0 rpm until both steam supply valves are re-opened.

D. …increase speed to approximately 4075 rpm and hold speed.

Proposed Answer: B

Explanation: Refer to reference ABN-305, Step 4.3.2 RNO to follow the logic for the correct answer.

A. Incorrect. Plausible if thought the TDAFWP would remain at 0 speed until the RO takes further action. However, the pump will immediately go to a minimum speed of 2000 rpm at minimum demand.

B. Correct. Following guidance of ABN-305, Step 4.3.2 RNO, the reactor operator lowers 1-SK-2452A AFWPT SPD CTRL to 0% output. This action limits the speed of the turbine to 2000 RPM when the NEO opens 1-HS-2452, AFWPT Trip and Throttle Valve.

C. Incorrect. Plausible if thought the TDAFWP would remain at 0 speed until the RO takes further action or until both steam admission valves are open. However, the pump will run at full speed on either steam admission valve being open and the pump will immediately go to a minimum speed of 2000 rpm at minimum demand.

D. Incorrect. Plausible if a loss of power to either Safeguards Bus were to occur the TDAFWP would have a full speed signal applied, however, because the actions of ABN-305 have been performed the turbine speed would only increase to ~2000 rpm.

Technical Reference(s): ABN-305, Step 4.3.2 RNO; SOP-304A, Step 5.1.2.F; ABN-602, Step 2.3.3 NOTE; LO21.SYS.AF1, Page 26 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Turbine Driven Auxiliary Feedwater Pump Malfunction in accordance with ABN-305, Auxiliary Feedwater System Malfunction.

Question Source:
- Bank: ILOT0082
- Modified Bank: ____ (Note changes or attach parent)
- New: ____

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge: ____
- Comprehension or Analysis: X

10 CFR Part 55 Content:
- 55.41: 10
- 55.43: ____

Comments / Reference: ABN-305, Step 4.3.2 RNO	Revision: 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-305
AUXILIARY FEEDWATER SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 55 OF 92

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 Verify Turbine Driven Auxiliary Feedwater Pump - NOT TRIPPED	Perform the following: a. Ensure at least one steam supply valve open: • u-HS-2452-1, AFWPT STM SPLY VLV-MSL 4 • u-HS-2452-2, AFWPT STM SPLY VLV-MSL 1 b. Lower u-SK-2452A, AFWPT SPD CTRL to 0% output. [C] c. Direct Nuclear Equipment Operator to reset the turbine trip and throttle valve per Attachment 1. d. Verify u-HS-2452 G/H, AFWPT TRIP & THROTTLE VALVE open AND valve red lights are lit. e. Adjust u-SK-2452A, AFWPT SPD CTRL to desired output. f. Return to procedure and step in effect.

Comments / Reference: ABN-602, Step 2.3.3 NOTE	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 6 OF 107

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:

- If power is greater than 10%. MDAFW should be allowed to run until the sequencer times out. The pumps will be stopped in Section 8.0, if not required. DO NOT throttle AFW above 10% power.
- The AFWP flow control and isolation valves are required to be fully open when above 10% power per TS 3.7.5.

NOTE:

- An emergency start will allow DG breaker to automatically close on a phase to ground bus fault (LOR 86-2/uEA1 or 86-2/uEA2).
- DG breaker will not automatically or manually close when a phase to phase bus fault (LOR 86-1) is present.
- An Operator Lockout signal from Blackout Sequencer (BOS) opens TDAFWP steam supply valves. The BOS also starts associated train MDAFWP. It may be necessary to limit AFW flow to prevent excessive RCS cooldown, or other adverse condition. Placing the TDAFW Pump in PULL-OUT with one safeguards bus de-energized will result in two inoperable AFW Pumps per TS 3.7.5. Throttling any train of AFW above 10% power renders the train INOPERABLE.
- Attachment 4 contains steps to deenergize the sequencer if the bus will not be needed. This would restore common equipment available to the other unit (e.g CRACs, UPS).

3 Check 6.9 KV safeguard buses - BOTH ENERGIZED	Perform the following:

Comments / Reference: SOP-304A, Step 5.1.2.F	Revision: 17

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-304A
AUXILIARY FEEDWATER SYSTEM	REVISION NO. 17 CONTINUOUS USE	PAGE 12 OF 71

5.1.2 D. CLOSE the following valves by placing the controllers at 0% output:

- ☐ • 1-FK-2459A, TD AFWP SG 1 FLO CTRL
- ☐ • 1-FK-2460A, TD AFWP SG 2 FLO CTRL
- ☐ • 1-FK-2461A, TD AFWP SG 3 FLO CTRL
- ☐ • 1-FK-2462A, TD AFWP SG 4 FLO CTRL

[C]

CAUTION: DO NOT operate the AFWPT at speeds below 1800 RPM for an extended period of time due to loss of oil flow to the bearings.

[C]

NOTE: The warm-up by-pass line around each steam supply isolation/control valve for the Turbine Driven Aux. Feedwater pump is normally LOCKED CLOSED AND will only be used in the surveillance testing of the TDAFWP operability.

[C] E. UNLOCK AND OPEN one OR both of the following valves to warm the steam lines: (approximately 2-3 minutes)

- ☐ • 1MS-0711, MSL 1-01 TO AFWPT STM SPLY VLV BYP VLV
- ☐ • 1MS-0712, MSL 1-04 TO AFWPT STM SPLY VLV BYP VLV

NOTE: In the following steps, any speed between 2000 - 2400 rpm is acceptable. |

F. Slowly OPEN 1-HV-2452, AFWPT 1-01 TRIP AND THROT VLV, by performing the following: |

1) IF performing maintenance/performance run in MODE 3 or above, THEN PERFORM the following:

☐ a. RAISE speed to 2200 rpm (2000 to 2400 rpm) |

☐ b. VERIFY governor control at approximately 2200 rpm (2000 to 2400 rpm - further opening of the Trip and Throttle valve should not increase speed). |

☐ c. FULLY OPEN 1-HV-2452, AFWPT 1-01 TRIP AND THROT VLV (both OPER & VLV red lights lit).

Comments / Reference: LO21.SYS.AF1, Page 26	Revision: 05/11/11

TDAFWP OVERSPEED TRIP

Annunciator window 4.6, "TD AFWP OVRSPD TRIP", on ALB-8B provides indication of a trip of the TDAFW Pump. When the TDAFW pump turbine trips, the AFWPT TRIP light on HS-2452F will be ON. The MDAFW pumps are started per SOP-304A(B) as necessary to maintain SG levels and the AFWPT speed controller on CB-09 is lowered to 0%, this will limit the initial speed to 2000 RPM when the PEO opens the T&TV. When directed, the AO will reset the overspeed trip linkage and slowly open the T&TV using the job aid similar to ABN-305, ATT. 1 posted on the wall in the TDAFWP room near the T&TV. When resetting the overspeed trip, it is important to note that the tappet nut and the emergency head lever have beveled mating surfaces. For the overspeed trip to reset, the tappet nut must drop into place to maintain the emergency head lever trip mechanism upright. With the trip mechanism upright, the T&TV can be repositioned which allows the emergency connecting rod to remain in the reset "standby" condition.

As the AO opens the T&TV, the governor valve stem should be monitored for movement to ensure that the valve stem is operating smoothly to maintain turbine speed. As the T&TV is opened the turbine will slowly accelerate to approximately 2000 rpm at which point the governor should maintain speed. The T&TV can then be fully opened. The AFWPT speed controller is set to zero and the T&TV is opened slowly in order to minimize the potential for turbine overspeed. After verifying that the T&TV indicates open at HS-2452G/H and the valve red lights are lit, the Control Room operator will adjust speed controller SK-2452A, AFWPT SPD CTRL on CB-09 to the desired output.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2014	Tier	2	
Change: 3	Group	1	
	K/A	062 A4.01	
Level of Difficulty: 3	Importance Rating	3.3	

AC Electrical Distribution System: Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Proposed Question: 42

The Balance of Plant Operator is transferring power from 2UT to 2ST during a normal plant shutdown in accordance with SOP-603B, 6900 V Switchgear.

CS-2A1-2, INCOMING BKR 2A1-2 is taken to the CLOSE position.

The following indications are observed WITHOUT the transfer occurring:

- Red position indicating light for breaker 2A1-1 (from 2UT) is LIT.
- Green position indicating light for breaker 2A1-2 (from 2ST) is LIT.
- 2A1 bus voltage is 6.9 KV.
- Synchroscope is at the 12 o'clock position.
- Synchroscope lights are DARK.
- Zero running volts are indicated on RUNNING VOLT V-RUN.

Which of the following prevented the bus transfer?

A. 2ST has NO power and thus the transfer is blocked.

B. The anti-parallel relay did NOT open 2UT supply breaker 2A1-1 to allow 2ST supply breaker 2A1-2 to close.

C. SS-2A1-2, BKR 2A1-2 SYNCHROSCOPE is in the OFF position.

D. The spring charging motor for 2UT supply breaker 2A1-1 is in the OFF position.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because with zero running volts indicated, a misconception that the source has no power is possible.

B. Incorrect. Plausible because the anti-parallel circuit prevents the two power sources from being paralleled for more than one second, however, the breaker from 2ST never had the opportunity to close because the synchroscope was not energized.

C. Correct. If the synchroscope is at the 12 o'clock position then the synchroscope lights should be at maximum brightness. If the lights are dark then the synchroscope has not been energized.

D. Incorrect. Plausible because the status of the charging motor can affect the breakers ability to close, however, with a green light indication on the breaker it can be assumed that the spring charging motor is in the ON position. Even if it were in the OFF position there is sufficient charge to cycle the breaker at least once.

Technical Reference(s) LO21.SYS.AC2, Pages 30 & 61
SOP-603, Step 5.1.1
SOP-603, Attachment 1

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the 6.9 KV and 480 V Electrical Distribution System and **PREDICT** the system response.

Question Source: Bank ILOT4006
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments / Reference: LO21.SYS.AC2, Page 61	Revision: 04/28/11

In the synchronization section, the synchroscope displays the phase relationship between the two sources while paralleling. The synchroscope is energized by placing the appropriate Synchroscope Switch for the breaker to be closed to ON. Running frequency and voltage meters display to the operator the on-line source's frequency and voltage (normally bus frequency and voltage). Incoming frequency and voltage meters display to the operator the oncoming source's frequency and voltage (normally DG frequency and voltage). Running and Incoming light indication below the synchroscope indicate phase differential (voltage) between the sources being paralleled. With the source's 180° out of phase, these lights would not be lit. DG control switches to raise and lower DG speed and voltage are provided to allow the operator to adjust parameters to allow the DG to pickup load when being paralleled.

Comments / Reference: LO21.SYS.AC2, Page 30	Revision: 04/28/11

Anti-Parallel Interlock

All of the 6.9KV buses have this feature. As soon as the alternate (or normal) power supply is paralleled and the breaker closed, the normal (or alternate) power supply breaker will open, preventing the two power sources from being paralleled for more than 1 second. This interlock does not affect the diesel breaker.

Comments / Reference: SOP-603, Step 5.1.1	Revision: 10

CPNPP SYSTEM OPERATION PROCEDURE MANUAL	UNIT 2	PROCEDURE NO. SOP-603B
6900 V SWITCHGEAR	REVISION NO. 10 CONTINUOUS USE	PAGE 6 OF 52

5.0 INSTRUCTIONS

5.1 Energizing 6.9KV Bus

5.1.1 Energizing a 6.9 KV Normal Bus From Station Service Transformer 2ST

This section describes the steps required to energize a 6.9 KV Normal Bus from Station Service Transformer 2ST.

A. ENSURE the prerequisites in Section 2.1 are met for the selected bus.

- ☐ • 6.9 KV SWITCHGEAR 2A1
- ☐ • 6.9 KV SWITCHGEAR 2A2
- ☐ • 6.9 KV SWITCHGEAR 2A3
- ☐ • 6.9 KV SWITCHGEAR 2A4

B. ENSURE all load breakers on the selected 6.9 KV Normal Bus being energized are OPEN.

- ☐ • 6.9 KV SWITCHGEAR 2A1
- ☐ • 6.9 KV SWITCHGEAR 2A2
- ☐ • 6.9 KV SWITCHGEAR 2A3
- ☐ • 6.9 KV SWITCHGEAR 2A4

C. TURN synchroscope ON for the selected Bus Feeder Breaker.

- ☐ • SS-2A1-2, BKR 2A1-2 SYNCHROSCOPE

Comments / Reference: SOP-603, Attachment 1	Revision: 10

CPNPP SYSTEM OPERATION PROCEDURE MANUAL	UNIT 2	PROCEDURE NO. SOP-603B
6900 V SWITCHGEAR	REVISION NO. 10 CONTINUOUS USE	PAGE 48 OF 52

[L] ATTACHMENT 1
PAGE 7 OF 9

GUIDELINES ON PROPER OPERATION OF 6.9 KV BREAKERS

2.0 E.

CAUTION: Upon reaching the point where the breaker automatically STOPS, the unlocking lever "clicks" into position (i.e., the unlocking lever seats itself with a positive mechanical action). DO NOT ATTEMPT TO RACK ANY FURTHER!

☐ 6) CONTINUE cranking until the racking mechanism automatically STOPS at the CONNECT position.

☐ 7) REMOVE Flash Protection Equipment (FPE).

☐ 8) INSTALL the CLOSE fuse.

NOTE: Performance of the following step will cause the springs to charge.

☐ 9) POSITION the spring charging motor toggle switch (Power Control Switch) up to the "ON" position.

NOTE: This following Step AND Substeps are a verification of the breaker to ensure it is installed correctly AND functional.

F. ENSURE the breaker is properly racked in by verifying the following:

☐ • Indication on the floor of the circuit breaker housing corresponds to the markings on the circuit breaker.(breaker indicates desired position, CONNECT OR TEST)

☐ • Racking release lever is fully in the CONNECT position (extreme counterclockwise position)

[C] ☐ • The spring charging motor toggle switch (Power Control Switch) is up to the ON position.

☐ • The springs are CHARGED. ("feet" are extended - labeled with "Spring Charged CAUTION" labels.)

☐ • The TRIP pushbutton (red) is flush with OR visibly extends beyond the plane of its steel housing plate.

☐ • The control power fuses are inserted with ON in the up position.

☐ • Proper indication. (Green AND blue lamps on the front of the breaker are lit. For breakers operated from the Control Room, the green lamp is lit on the CR handswitch).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	063 K2.01	
Level of Difficulty: 2	Importance Rating	2.9	

DC Electrical Distribution System: Knowledge of bus power supplies to the following: Major DC loads

Proposed Question: 43

Given the following conditions:

- Unit 1 is operating at 100%.
- A loss of control power to Reactor Trip Breaker B (RTB) occurs.

Which of the following states the DC Bus which has de-energized and the effect on plant operation?

A. 1ED1 is de-energized and the Reactor is tripped.

B. 1ED1 is de-energized and the Reactor is NOT tripped.

C. 1ED2 is de-energized and the Reactor is tripped.

D. 1ED2 is de-energized and the Reactor is NOT tripped.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a loss of the 48 VDC supply would result in an Undervoltage trip. The correct power supply for RTB B is 1ED2.
B. Incorrect. Plausible because the RTB B shunt trip would be inoperable but would not result in a reactor trip. The correct power supply for RTB B is 1ED2.
C. Incorrect. Plausible because a loss of the 48 VDC supply would result in an Undervoltage trip. 1ED2 is the correct power supply for RTB B.
D. Correct. RTB B control power is from 1ED2 and loss of control power will result in the shunt trip being inoperable but the Reactor will not trip.

Technical Reference(s) LO21.SYS.ES2, Page 32 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the DC Electrical Distribution System.

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	6	
	55.43		

Comments / Reference: LO21.SYS.ES2, Page 32	Revision: 06/09/11
The Auto Shunt Trip (the STA (STB)) relay was a post-Salem ATWS backfit to SSPS. Prior to the failure of the Undervoltage trip coil to trip the Reactor Trip breakers at Salem, the SSPS only tripped the reactor trip breakers by de-energizing the undervoltage trip coils on the breakers. The Salem ATWS made the requirement change so that the SSPS would also cause the shunt trip coils on the trip breakers to energize and trip the reactor trip breakers. The STA (STB) relay does this by closing contacts in the shunt trip coil circuit whenever the STA (STB) relay is de-energized. Thus, when the UV driver card stops sending 48 VDC to the trip breakers, the reactor trip breakers will trip due to both the undervoltage trip coil on the breaker actuating to trip the breaker and the STA (STB) relay de-energizing and its “b” contacts causing the shunt trip coil on the trip breaker to energize. Since the bypass breaker is only closed for 2 hours in a two month period (for SSPS logic testing), the bypass breakers were considered not to need the additional relay, and therefore they only automatically trip by their undervoltage relay coil de-energizing.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	064 K6.08	
Level of Difficulty: 3	Importance Rating	3.2	

Emergency Diesel Generator System: Knowledge of the effect that a loss or malfunction of the following will have on the EDG system: Fuel oil storage tanks

Proposed Question: 44

Given the following conditions:

- Unit 1 has completed a 24 hour run of emergency Diesel Generator (DG) 1-02.
- Diesel Generator 1-02 Panel, Window 3.3 – LOW LEVEL FUEL STORAGE TANK is LIT.
- Level in the DG 1-02 Fuel Oil Storage Tank (FOST) is determined to be 82,000 gallons.

Which of the following is the potential operational impact of the DG FOST level?

The FOST level is…

A. …less than the amount required by Technical Specifications and DG 1-02 may not be capable of satisfying the requirement to supply power following a Design Basis Accident for a minimum period of 6 days.

B. …less than the amount required by Technical Specifications and DG 1-02 may not be capable of satisfying the requirement to supply power following a Design Basis Accident for a minimum period of 7 days.

C. …greater than the amount required by Technical Specifications and DG 1-02 is capable of satisfying the requirement to supply power following a Design Basis Accident for a minimum period of 6 days.

D. …greater than the amount required by Technical Specifications and DG 1-02 is capable of satisfying the requirement to supply power following a Design Basis Accident for a minimum period of 7 days.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the level of 82,000 gallons is less than the Technical Specification required amount of 86,000 gallons, but is greater than the Technical Specification amount allowed for up to 48 hours that ensures a 6 day supply.
B. Correct. 82,000 gallons is less than the Technical Specification required amount of ≥ 86,000 gallons. ≥ 86,000 gallons of fuel oil in the FOST is required to ensure that the DG can supply all required loads following a DBA for a period of 7 days.
C. Incorrect. Plausible because 82,000 gallons is greater than the 74,600 gallons that ensures a 6 day supply, allowed per Technical Specifications for up to 48 hours when the 7 day supply is unavailable, but 82,000 gallons is still below the Technical Specification required amount.
D. Incorrect. Plausible because 82,000 gallons is greater than the 74,600 gallons that ensures a 6 day supply, allowed per Technical Specifications for up to 48 hours when the 7 day supply is unavailable, but 82,000 gallons is still below the Technical Specification required amount.

Technical Reference(s): Technical Specification LCO 3.8.3.A; OPT-214A, Section 5.2; Technical Specification LCO 3.8.3 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Spent Fuel Pool Cooling and Cleanup system including Technical Specifications, TRM and ODCM.

Question Source: Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: Technical Specification LCO 3.8.3.A, SR 3.8.3.1	Amendment: 161

Diesel Fuel Oil, Lube Oil, and Starting Air
3.8.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ a 7 day supply of fuel.	In accordance with the Surveillance Frequency Control Program.

Comments / Reference: OPT-214A, Section 5.2	Revision: 22

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-214A
DIESEL GENERATOR OPERABILITY TEST	REVISION NO. 22 CONTINUOUS USE	PAGE 15 OF 147

5.2 Limitations (continued)

- As a minimum, the following A.C. electrical power sources shall be OPERABLE in MODES 5 AND 6 per TS 3.8.2:
  - One circuit between the offsite transmission network AND the onsite class 1E Distribution subsystem required by LCO 3.8.10.
  - One diesel generator capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.10 with a fuel oil day tank containing a minimum volume of 1440 gallons of fuel.
- The Stored Diesel Fuel Oil, Lube Oil AND Starting Air Subsystem shall be within limits for each DG required to be OPERABLE per TS 3.8.3 as follows:
  - The Fuel Oil Storage Tank shall contain ≥86,000 gallons (MODES 1-6) of fuel.

Comments / Reference: Technical Specification LCO 3.8.3 Bases	Revision: 68

BASES

ACTIONS

C.1 (continued)

that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for approximately 7 days at full load. A small volume in the day tank in excess of the day tank requirements is credited to ensure a full 7 day supply. The fuel oil level equivalent to a 7 day supply is 86,000 gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	073 A2.02	
Level of Difficulty: 2	Importance Rating	2.7	

Process Radiation Monitoring System: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: 45

Given the following condition:

- The PC-11 is alarming due to OPERATE FAILURE - CHANNEL NO PULSES RECEIVED on X-RE-5251A, Auxiliary Building Low Volume Waste Monitor.

Which of the following describes the impact on the Auxiliary Building Drains and the required actions to mitigate the condition?

Auxiliary Building Drains will…

A. …remain in the normal alignment and would still initiate automatic actions on high radiation. Notify I&C to investigate loss of counts alarm on PC-11 per ALM-3200, Digital Radiation Monitoring System.

B. …automatically isolate and the Auxiliary Building Sump Pumps will stop. Verify the automatic actions occurred and stop any waste producing activities.

C. …remain in the normal alignment and no automatic actions would occur on high radiation. Dispatch an operator to locally align Auxiliary Building Drains to the Co-Current Waste System.

D. …automatically divert to the Co-Current Waste System. Verify the automatic action occurred and implement the requirements of the Offsite Dose Calculation Manual.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that an OPERATE FAILURE would not result in automatic initiation and the high radiation feature is still available. Also the action to have I&C investigate is somewhat correct, however, this monitor in OPERATE FAILURE will initiate the automatic actions and is a required instrument per the Offsite Dose Calculation Manual (ODCM).
B. Incorrect. Plausible because it could be thought that an OPERATE FAILURE would result in these automatic actions and the specified actions make sense to prevent sumps from backing up, however, the loss of pulses is an OPERATE FAILURE condition that results in automatic actions that must be verified. The detector is out of service and the ODCM actions apply.
C. Incorrect. Plausible because the response would be correct for some monitors, however, this monitor in OPERATE FAILURE will initiate the automatic actions and is a required instrument per the ODCM.
D. Correct. The loss of pulses is an OPERATE FAILURE condition that results in automatic actions that must be verified. The detector is out of service and the ODCM actions apply.

Technical Reference(s) ALM-3200, Pages 38, 83, & 102 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operations of the Liquid Waste Systems.

Question Source: Bank ILOT8281
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 13
55.43 ______

Comments / Reference: ALM-3200, Page 38	Revision: 4

CPSES CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 4	PAGE 38 OF 117

ALARM: **OPERATE FAILURE-CHANNEL NO PULSES RECEIVED** COLOR: BLUE

AFFECTED MONITORS:

All monitors may display this alarm.

PROBABLE CAUSES:

Loss of high voltage to the detector
Damaged signal cable
Failed detector
Failed detector pre-amplifier
Channel Item 020 LOSS OF COUNTS TIME value is too short

MONITOR RESPONSE:

Automatic actions for monitors which actuate due an OPERATE FAILURE will be initiated

OPERATOR ACTION:

1. Determine the affected monitor.

   A. IF any of the following monitors are affected, THEN notify Radwaste personnel of the alarm condition.

      - LWE076 (X-RE-5253)
      - TBDu72 (u-RE-5100)

2. Refer to Attachment 3 and ensure the automatic actions occurred as required due to the OPERATE FAILURE.

3. Refer to Attachment 1 to determine TS 0R ODCM requirements on applicable monitors.

Comments / Reference: ALM-3200, Page 83	Revision: 4

CPSES CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 4	PAGE 83 OF 117

ATTACHMENT 1
PAGE 1 OF 4

TECHNICAL SPECIFICATION/ODCM MONITORS

CHANNEL	PC11 DISPLAY	TITLE	TECHNICAL SPECIFICATION/ODCM
		Containment Ventilation Isolation Instrumentation	
u-RE-5503	CAGu97	Gaseous Radioactivity	TS 3.3.6 Table 3.3.6-1 Function 3.a
		CREFS Actuation Instrumentation	
X-RE-5895A	CRV053	Air Intake-Radiation Level	TS 3.3.7 Table 3.3.7-1 Function 3.a
X-RE-5895B	CRV054	Air Intake-Radiation Level	TS 3.3.7 Table 3.3.7-1 Function 3.a
X-RE-5896A	CRV091	Air Intake-Radiation Level	TS 3.3.7 Table 3.3.7-1 Function 3.b
X-RE-5896B	CRV092	Air Intake-Radiation Level	TS 3.3.7 Table 3.3.7-1 Function 3.b
		Post Accident Monitoring Instrumentation	
u-RE-6290A	CTEu16	Containment Area Radiation (High Range)	TS 3.3.3 Table 3.3.3-1 Function 10
u-RE-6290B	CTWu17	Containment Area Radiation (High Range)	TS 3.3.3 Table 3.3.3-1 Function 10
		RCS Leakage Detection Instrumentation	
u-RE-5502	CAPu98	Containment Atmosphere Particulate Radioactivity Monitoring System	TS 3.4.15 b
u-RE-5503	CAGu97	Containment Atmosphere Gaseous Radioactivity Monitoring System	TS 3.4.15 c
		Radioactive Liquid Effluent Monitoring Instrumentation	
X-RE-5253	LWE076	Liquid Radwaste Effluent Line	ODCM 3.3.3.4 Table 3.3-7 Item 1.a
u-RE-5100	TBDu72	Turbine Building Sumps Effluent Line	ODCM 3.3.3.4 Table 3.3-7 Item 1.b
u-RE-4269	SSWu 65	Trn A Service Water System Effluent Line	ODCM 3.3.3.4 Table 3.3-7 Item 2.a
u-RE-4270	SSWu66	Trn B Service Water System Effluent Line	ODCM 3.3.3.4 Table 3.3-7 Item 2.a
X-RE-5251A	ABP074	Aux Bld Low Vol Waste	ODCM 3.3.3.4 Table 3.3-7 Item 1.c

Comments / Reference: ALM-3200, Page 102	Revision: 4

CPSES CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 4	PAGE 102 OF 117

ATTACHMENT 3
Page 1 of 1

AUTOMATIC ACTIONS

NOTE: A loss of power to the RM-80 will result in the Automatic Actions for the associated monitor.

TITLE	CHANNEL	FUNCTION	PRINT
Plant Vent Stack Wide Range Gas Monitor	X-RE-5570A S. X-RE-5570B N.	Closes HCV-014 on High Radiation or any OPERATE FAILURE	E1-0046 Sh 62/63
Auxiliary Building Exhaust	X-RE-5701	Closes HCV-014 on High Exhaust Radiation or any OPERATE FAILURE	E1-0065 Sh 22
Liquid Waste to Circulating Water	X-RE-5253	Closes discharge to Circulating Water Circulating Water (X-RV-5253) on High Radiation or any OPERATE FAILURE	E1-0065 Sh 29
Turbine Building Drains	u-RE-5100	Closes the discharge to Low Volume Waste (u-RV-5100A) and opens discharge to Co-Current Waste	E1-0055 Sh 61/62 E2-0055 Sh 61/62
Containment Air Gaseous and Particulate	u-RE-5503 u-RE-5502	Causes Containment Ventilation Isolation on High Radiation	E1-0046 Sh 62/64 E2-0046 Sh 62/64
Control Room Air Supply (Gas)	X-RE-5895A/B X-RE-5896A/B	Initiates Control Room Emergency Recirculation on High Radiation	E1-0046 Sh 62/63 E1-0035 Sh 76/77
Secondary Sample	u-RE-4200	Isolates Steam Generator Blowdown and Sampling System on High Radiation	E1-0040 Sh 97 E2-0040 Sh 97
Common discharge AB, DG Sumps and CCW Drain Tanks	X-RE-5251A	Diverts to Cocurrent Waste System Wastewater Holdup Tanks on High Radiation or any OPERATE FAILURE	E1-0065 Sh 58

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	2	
Change: 3	Group	1	
	K/A	076 K3.07	
Level of Difficulty: 3	Importance Rating	3.7	

Service Water System: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads

Proposed Question: 46

Given the following conditions:

- Train A is the protected train on both units.
- A fault has occurred on XST2.
- All equipment responded in accordance with design, with the following noted exception:
    - The Train B Station Service Water (SSW) Pump on the affected bus failed to start on the Blackout Sequencer.

Which component is operating without cooling until the affected SSW Pump can be started?

A. Emergency Diesel Generator 1-02.

B. Centrifugal Charging Pump 1-02.

C. Emergency Diesel Generator 2-02.

D. Centrifugal Charging Pump 2-02.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the fault on XST2 will cause a loss of the preferred offsite power supply to 1EA2 but the slow transfer to the alternate offsite power supply will occur prior to the EDG getting a start signal. Therefore, the EDG 1-02 does not start and is thus not running without cooling.

B. Correct. The fault on XST2 will cause a loss of the preferred offsite power supply to 1EA2 which will result in the Blackout Sequencer operating. The Blackout Sequencer will start CCP 1-02 and should also start SSWP 1-02. Since the malfunction is a failure of the SSWP 1-02 to start, CCP 1-02 which is cooled by SSW is running without cooling water.

C. Incorrect. Plausible because the loss of XST2 will affect both units. However, the effect on Unit 2 is the loss of the alternate power supply to 2EA2 and thus the Blackout Sequencer does not operate for Unit 2 and there is no affected train of equipment.

D. Incorrect. Plausible because the loss of XST2 will affect both units. However, the effect on Unit 2 is the loss of the alternate power supply to 2EA2 and thus the Blackout Sequencer does not operate for Unit 2 and there is no affected train of equipment.

Technical Reference(s) ABN-601, Steps 2.1.b, 2.2 & 3.2 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Station Service Water Pump Trip in accordance with ABN-501, Station Service Water System Malfunction.

Question Source:

Bank		
Modified Bank		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: ABN-501, Step 2.3.5	Revision: 9

Comments / Reference: ABN-501, Step 2.3.6	Revision: 9

Comments / Reference: EOP-1.0A, Step 18	Revision: 8

Comments / Reference: EOS-1.3A, Attachment 3, Step 2 Bases	Revision: 8

Comments / Reference: EOP-0.0A, Attachment 9, Item 10	Revision: 8

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/5/2014	Tier	2	
Change: 2	Group	1	
	K/A	078 K1.02	
Level of Difficulty: 2	Importance Rating	2.7	

Instrument Air System: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service air

Proposed Question: 47

Given the following conditions:

- Unit 1 is in MODE 6 during a Refueling Outage.
- Work is in progress inside the Steam Generators.
- Core offload is in progress.
- The Wet Cask Pit is at reduced level for fuel inspection equipment repair.
- A loss of Instrument Air has occurred.

Which of the following is the impact on the Unit 1 Service Air System?

A. Service Air will be aligned for breathing air.

B. Service Air will be aligned to the Wet Cask Pit gate.

C. Service Air compressor will not start.

D. Service Air will be unavailable to Containment.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because breathing air will be lost but service air is not a backup supply for breathing air.
B. Incorrect. Plausible because instrument air supplies the gate seal but compressed air bottles are provided for the gate seal.
C. Incorrect. Plausible because the misconception could exist that instrument air is required as a support system to the service air compressor, thus rendering it incapable of starting.
D. Correct. 1-HS-3486, CNTMT SERV AIR ISOL VLV will fail close isolating service air to Containment.

Technical Reference(s) ABN-301, Steps 2.3.2 RNO & 2.3.3 RNO
ABN-301, Attachment 1
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Instrument Air System.

Question Source: Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 9
55.43 ____

Comments / Reference: ABN-301, Step 2.3.2 RNO	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 5 OF 122

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 Verify Instrument Air Header Pressure - GREATER THAN OR EQUAL TO 85 psig: • u-PI-3488, INST AIR AFTFILT OUT PRESS	Perform the following: a. Start AND align a common Instrument Air Compressor per SOP-509A. b. Attempt to restart the tripped compressor per SOP-509A/B Diagnostic Guideline. c. IF temporary air compressor available, THEN ensure it is started AND aligned per SOP-509A/B. d. Stop all unnecessary use of instrument air. [R] • Announce over Plant Page System, "ATTENTION ALL PERSONNEL, WE HAVE A LOSS OF INSTRUMENT AIR, ANYONE USING INSTRUMENT AIR AS BREATHING AIR MUST GO TO A SAFE ATMOSPHERE AND STOP BREATHING THE INSTRUMENT AIR. STOP ALL UNNECESSARY EVOLUTIONS REQUIRING INSTRUMENT AIR USAGE UNTIL FURTHER NOTICE".

Comments / Reference: ABN-301, Step 2.3.3 RNO	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 6 OF 122

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify Instrument Air Header Pressure - INCREASING OR STABLE.	Monitor Instrument Air Header Pressure continuously. a. IF header pressure continues to decrease, THEN perform the following: 1) Consult with opposite unit Control Room to crosstie Unit 1 and Unit 2 Instrument Air Headers per SOP-509A. 2) Containment Building: • IF fueling activities are in progress, THEN place ALL fuel bundles in a safe condition AND suspend fueling activities. • Ensure Fuel Transfer Cart is positioned in the Fuel Building. 3) Fuel Building: • Ensure ALL spent fuel is placed in Spent Fuel Storage Racks.

Comments / Reference: ABN-301, Attachment 1	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 38 OF 122

ATTACHMENT 1
PAGE 1 OF 15

CONTROL BOARD AIR OPERATED VALVE FAILURE POSITIONS

LOCATION	COMPONENT	NOMENCLATURE	FAILURE POSITION
CB-01	u-ZL-3464	INSTR AIR COMM COMPR 2 UNIT u SPLY VLV	F.O.
CB-01	u-ZL-3476	INSTR AIR COMM COMPR 1 UNIT u SPLY VLV	F.C.
CB-01	u-HS-3486	CNTMT SERV AIR ISOL VLV	F.C.
CB-01	u-HS-3487	CNTMT INSTR AIR ISOL VLV	F.C.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	1	
	K/A	103 A3.01	
Level of Difficulty: 3	Importance Rating	3.9	

Containment System: Ability to monitor automatic operation of the Containment system, including: Containment isolation

Proposed Question: 48

Given the following conditions:

- Unit 1 is responding to a Large Break Loss of Coolant Accident in accordance with EOP-0.0A, Reactor Trip or Safety Injection.
- While verifying Containment Isolation Phase A, the Balance of Plant Operator determines that automatic and manual actuation of Containment Isolation Phase A will NOT function.

Which of the following valves must be closed as part of Containment Isolation Phase A?

A. 1/1-8153, XS LTDN ISOL VLV.
   1/1-8154, XS LTDN ISOL VLV.

B. 1/1-LCV-0459, U1 LTDN ISOL VLV 0459.
   1/1-LCV-0460, U1 LTDN ISOL VLV 0460.

C. 1-HV-6082, CH WTR RET ISOL VLV.
   1-HV-6084, CH WTR SPLY ISOL VLV.

D. 1-HV-4650, VENT CHLR CCW SPLY & RET VLV.
   1-HV-4631, PSC CCW SPLY & RET VLV.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 1/1-8152 & 1/1-8160, LTDN ISOL VLVs are part of a Phase A Containment Isolation Signal, however, the Excess Letdown Isolation Valves are located inside Containment and are isolated via the Seal Water Return lines.
B. Incorrect. Plausible because these valves will close, however, the actuation signal is a Safety Injection slave relay.
C. Correct. The Chill Water Return and Supply Isolation Valves are isolated on a Phase A Containment Isolation Signal.
D. Incorrect. Plausible because these valves do receive a close signal, however, it is generated from a Safety Injection Actuation Signal.

Technical Reference(s) EOP-0.0A, Attachments 2, 4, 7, & 10 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Containment System and **PREDICT** the system response.

Question Source: Bank ILOT7344
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9
55.43

Comments / Reference: EOP-0.0A, Attachment 2	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 27 OF 117

ATTACHMENT 2
PAGE 6 OF 9

SAFETY INJECTION ACTUATION ALIGNMENT

TABLE 1
PAGE 1 OF 4

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	CONDITION
UNIT 1 MAIN CONTROL BOARD			
☐ CB-03	X-HS-5534	H2 PRG SPLY FN 4	STOPPED
☐ CB-03	X-HS-5532	H2 PRG SPLY FN 3	STOPPED
☐ CB-04	1/1-8716A	RHRP 1 XTIE VLV	OPEN
☐ CB-04	1/1-8716B	RHRP 2 XTIE VLV	OPEN
☐ CB-06	1/1-8153	XS LTDN ISOL VLV	CLOSED/H.S. IN CLOSED
☐ CB-06	1/1-8154	XS LTDN ISOL VLV	CLOSED/H.S. IN CLOSED

Comments / Reference: EOP-0.0A, Attachment 2	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 24 OF 117

ATTACHMENT 2
PAGE 3 OF 9

SAFETY INJECTION ACTUATION ALIGNMENT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5	Verify CCW Pumps - RUNNING	☐ Manually start pump(s).
☐ 6	Verify RHR Pumps - RUNNING	☐ Manually start pump(s).
7	Verify Proper CVCS Alignment:	
	☐ a. Verify CCPs - RUNNING	☐ a. Manually start pump(s).
	b. Verify Letdown Relief Valve isolation:	
	☐ 1) Letdown orifice isolation valves - CLOSED	☐ 1) Manually close valve(s).
	2) Letdown isolation valves - CLOSED	☐ 2) Manually close valve(s).
	☐ • 1/1-LCV-459	
	☐ • 1/1-LCV-460	

Comments / Reference: EOP-0.0A, Attachment 4	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 33 OF 117

ATTACHMENT 4
PAGE 1 OF 6

PHASE A ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	ESFAS TRAIN	MLB LOCATION
☐ CB-01	1-HS-5366(1)	CNTMT DEMIN WTR ISOL VLV	A	1-MLB-1A1/3.4
☐ CB-01	1-HS-5365(1)	CNTMT DEMIN WTR ISOL VLV	B	1-MLB-1B1/3.4
☐ CB-01	1-HS-3487(1)	CNTMT INSTR AIR ISOL VLV	B	1-MLB-4B2/3.6
☐ CB-01	1-HS-3486(1)	CNTMT SERV AIR ISOL VLV	B	1-MLB-4B2/3.6
☐ CB-01	1-HS-5158(1)	CNTMT SMP DRN ISOL VLV	A	1-MLB-4A1/4.9
☐ CB-01	1-HS-5157(1)	CNTMT SMP DRN ISOL VLV	B	1-MLB-4B1/4.9
☐ CB-02	1/1-8823(3)	SI TO CL 1 • 4 TEST ISOL VLV	A	1-MLB-1A2/3.8
☐ CB-02	1/1-8888(1)	ACCUM FILL ISOL VLV	B	1-MLB-1B2/3.7
☐ CB-02	1/1-8881(3)	SI TO HL 2 & 3 TEST ISOL VLV	A	1-MLB-1A2/3.7
☐ CB-02	1/1-8824(3)	SI TO HL 1 & 4 TEST ISOL VLV	A	1-MLB-1A2/4.8
☐ CB-02	1/1-8871(1)	SI TEST HDR RET ISOL VLV	A	1-MLB-1A2/2.7
☐ CB-02	1/1-8964(1)	SI TEST HDR RET ISOL VLV	B	1-MLB-1B2/2.7
☐ CB-02	1-HS-4165A	PSS ISOL VLV	A	(NOTE 1)
☐ CB-02	1-HS-4167A	PSS ISOL VLV	B	(NOTE 1)
☐ CB-03	1-HS-4075C(1)	CNTMT FIRE PROT ISOL VLV	A	1-MLB-1A2/2.5
☐ CB-03	1-HS-4075B(1)	CNTMT FIRE PROT ISOL VLV	B	1-MLB-1B2/2.5
☐ CB-03	1-HS-6082(1)	CH WTR RET ISOL VLV	B	1-MLB-45B/1.5
☐ CB-03	1-HS-6083(1)	CH WTR RET ISOL VLV	A	1-MLB-45A/1.5
☐ CB-03	1-HS-6084(1)	CH WTR SPLY ISOL VLV	B	1-MLB-45B/2.5

Comments / Reference: EOP-0.0A, Attachment 7	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 43 OF 117

ATTACHMENT 7
PAGE 2 OF 8

SAFETY INJECTION ACTUATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	CONDITION	MLB LOCATION
UNIT 1 MAIN CONTROL BOARD (Con't)				
☐ CB-02	1-HS-4766	CSP 2	STARTED	1-MLB-4B1/1.2
☐ CB-02	1-HS-4767	CSP 4	STARTED	1-MLB-4B1/2.2
☐ CB-03	1-HS-4536	CCWP 1 RECIRC VLV	CLOSED	1-MLB-4A1/2.4
☐ CB-03	1-HS-4572	RHR HX 1 CCW RET VLV	AUTO, 40% FLO LIGHT LIT	1-MLB-4A1/2.3
☐ CB-03	1-HS-4537	CCWP 2 RECIRC VLV	CLOSED	1-MLB-4B1/2.4
☐ CB-03	1-HS-4573	RHR HX 2 CCW RET VLV	AUTO, 40% FLO LIGHT LIT	1-MLB-4B1/2.3
☐ CB-03	1-HS-4518A	CCWP 1	STARTED	1-MLB-4A1/1.4
☐ CB-03	1-HS-4519A	CCWP 2	STARTED	1-MLB-4B1/1.4
☐ CB-03	1-HS-4650	VENT CHLR CCW SPLY & RET VLV	CLOSED	1-MLB-4B1/2.7
☐ CB-03	1-HS-4631	PSC CCW SPLY & RET VLV	CLOSED	1-MLB-4A2/4.7

Comments / Reference: EOP-0.0A, Attachment 10	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 108 OF 117

ATTACHMENT 10
PAGE 29 OF 38

BASES

involves a check valve or closed system, the check valve or closed system barrier serves as a barrier for the containment isolation function. (The check valve (CI-030) can be credited for the inside containment isolation of the Containment Instrument Air penetration if the outside containment isolation valve (HS-3487) fails to close or does not have MLB indication available. The closed system function CCW supply and return can be credited for the inside containment isolation if outside containment isolation valves (HS-4710 and/or HS-4711) are not closed or cannot be verified provided.) The check valve or the closed system can be credited as a functional barrier since the penetration would be placed in an isolated condition as required by Technical Specifications if leakage exceeding the allowed limits. Additionally, if the penetration involves a Phase A isolation valve within an essential flow path and the Phase A valve has failed to close or cannot be verified , the essential flow path portion of the penetration can be credited as being pressurized or isolated

When checking the MLBs DARK, the Monitor Light Boxes should be back lit by pressing the test button to ensure that burned out bulbs do not give an indication that the penetration valve is closed.

4. Non-essential ventilation penetrations are isolated to prevent potential release of radioactive materials from containment. The appropriate MLB light indications to verify CVI are 1-MLB-45A and 45B green windows lit. One side of the containment penetration being isolated is sufficient to ensure adequate containment isolation at this time in the response and recovery action sequence. Subsequent steps may be performed. However, actions to close the redundant isolation valve should be pursued as time allows.

5. CCW pumps provide cooling to certain safeguards components. Component Cooling Water is assumed to be available for subsequent actions. ABN-502, COMPONENT COOLING WATER SYSTEM MALFUNCTION provides additional guidance for the contingency of a loss of Component Cooling Water System and can be referenced as required.

6. RHR pumps provide makeup inventory to the RCS for core cooling during accident conditions. Since an SI has actuated, both RHR pumps have a start signal; therefore, the operator should verify that they are running.

7. CCPs provide makeup inventory to the RCS for core cooling during accident conditions. Since an SI has actuated, both CCPs have a start signal; therefore, the operator should verify that they are running. Train A SI actuation generates a close signal to the letdown isolation valves (LCV-0459 & LCV-0460). The operator is instructed to verify isolation of the letdown isolation valves to ensure an inventory loss pathway is not set up through the letdown relief valve to the PRT.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	2	
Change: 3	Group	1	
	K/A	003 A1.02	
Level of Difficulty: 2	Importance Rating	2.9	

Reactor Coolant Pump System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP pump and motor bearing temperatures

Proposed Question: 49

Given the following conditions on Reactor Coolant Pump (RCP) 1-02:

- Lower seal water bearing (pump radial) temperature is 221ºF.
- Shaft vibration is 14 mils and steady.
- Motor bearing temperature is 207ºF.
- Number one (#1) seal water leakoff temperature is 215ºF.

RCP 1-02 must be stopped in accordance with ABN-101, Reactor Coolant Pump Trip/Malfunction due to high...

A. ...temperature on the lower seal water bearing.

B. ...pump shaft vibration.

C. ...motor bearing temperature.

D. ...seal water leakoff temperature.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because there is a temperature limit on the lower seal water bearing, however, that value is 225°F.
B. Incorrect. Plausible because there is a limit on pump shaft vibration, however, that value is less than 20 mils and between 15 mils and 20 mils with increasing amplitude of 1 mil per hour.
C. Correct. With a motor bearing temperature greater than 195°F RCP 1-02 must be stopped.
D. Incorrect. Plausible because seal leak off temperature is a monitored parameter for a loss of seal injection and/or thermal barrier cooling water, however, only abnormal seal leak off flow requires an RCP to be stopped.

Technical Reference(s): ABN-101, Steps 3.3.2, 6.3.1, & Section 9.1; ABN-101, Attachment 1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Reactor Coolant System.

Question Source: Bank ILOT2044
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3
55.43

Comments / Reference: ABN-101, Step 3.3.2	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 7 OF 48

3.3 Operator Actions

		ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	1	Verify affected RCP IN OPERATION	Determine if any maintenance being performed on affected RCP which would cause alarm.
[C] ☐	2	Check all motor bearing temperatures on affected pump - LESS THAN 195°F.	Perform the following: a. Manually trip Reactor AND GO TO EOP-0.0A/B while other qualified operators continue with this procedure. b. Stop affected RCP. c. GO TO Section 2.0 of this procedure.

Comments / Reference: ABN-101, Step 6.3.1	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 24 OF 48

6.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Amplitude trip rate of 1 mil/hr is based on operation at 100% power. IF not in Mode 1, THEN the amplitude rate of change may be ignored. IF DAS connected to the vibration monitoring panel, THEN filtered data should be used to determine trip criteria. (EVAL-2000-002454-01)

[C]

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Check RCP vibration - WITHIN LIMITS:	Perform the following:
☐ a. RCP shaft vibration: • LESS THAN 20 mils AND • IF between 15 mils and 20 mils, THEN amplitude increasing LESS THAN ONE mil/hr.	1) Trip Reactor AND GO TO EOP-0.0A/B while other operators continue this procedure. 2) Stop affected RCP. 3) GO TO Section 2.0 of this procedure.
☐ b. RCP frame vibration: • LESS THAN 5 mils AND • IF between 3 mils and 5 mils, THEN amplitude increasing LESS THAN 0.2 mils/hr.	

Comments / Reference: ABN-101, Section 9.1	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 40 OF 48

9.0 LOSS OF SEAL INJECTION AND THERMAL BARRIER COOLING WATER

9.1 Symptoms

a. Annunciator Alarms

- ANY RCP THBR CLR CCW RET TEMP HI (3B-2.11)
- ANY RCP MOTOR CLR CCW RET FLO LO (3B-2.12)
- ANY RCP THBR CLR CCW RET FLO LO (3B-3.11)
- ANY RCP UP BRG L\O CLR CCW RET FLO LO (3B-3.12)
- ANY RCP LOW BRG L\O CLR CCW RET FLO LO (3B-4.12)
- ANY RCP 1 SEAL LKOFF FLO HI (5A-1.2)
- ANY RCP SEAL WTR INJ FLO LO (5A-1.6)

b. Plant Indications

- Computer alarms on RCP bearing temperatures.
- Computer alarm on RCP motor winding temperatures.
- Possible increase in RCP vibration.
- Increased Number 1 seal leakoff flow or temperature.
- Increased radial bearing temperature.

Comments / Reference: ABN-101, Attachment 1	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 46 OF 48

ATTACHMENT 1
PAGE 1 OF 1
RCP PARAMETERS

NOTE: The following list may aid determination of the validity of a temperature alarm or indication change:

- Local RTD (stator) monitoring (System Engineering/I&C) outside bioshield
  U1- RTD terminals: TBX-RCDARK-01[RCP 1, 2]; TBX-RCDARK-02 [RCP 3, 4]
  U2- RTD terminals: TCX-RCDARK-01[RCP 1, 2]; TCX-RCDARK-02 [RCP 3, 4]
- Thermographic performance comparison between pumps (System Engineering/Predictive Maintenance)
- Local evidence of restricted air flow
- Vibration change
- RCP motor amps high or changing
- Affected RCP loop flow or temperature change
- Bus voltage high or low, phase imbalance
- RCP motor air cooler air outlet temperature change
- Affected cooler CCW inlet/outlet temperature change
- Loose Parts Monitoring System alarm
- RCP seal leakoff or injection, flow or temperature change

Monitor the parameters below, as determined by Unit Supervisor:

IF motor bearing temperature is greater than or equal to 190°F, THEN perform Section 3.0 for RCP High or Low Lube Oil Level, while continuing.

IF motor bearing temperature increases by approximately 2°F from previous reading AND NO significant change in L\O Cooler CCW temperatures is observed, THEN notify System Engineering and Duty Manager.

IF any RCP bearing oil reservoir alarm LIT, THEN perform Section 3.0 while continuing section in effect.

RCP OPERATING LIMITS					
PARAMETER	LIMIT	RCP 1	RCP 2	RCP 3	RCP 4
MOT STAT WNDG TEMP	300°F	T0412A	T0432A	T0452A	T0472A
MOT UP RDL BRG TEMP	195°F	T0413A	T0433A	T0453A	T0473A
MOT UP THR BRG TEMP	195°F	T0414A	T0434A	T0454A	T0474A
MOT LOW RDL BRG TEMP	195°F	T0415A	T0435A	T0455A	T0475A
MOT LOW THR BRG TEMP	195°F	T0416A	T0436A	T0456A	T0476A
LOW SEAL WTR BEARING TEMP (Pump Bearing)	225°F	T0417A	T0437A	T0457A	T0477A
SEAL WTR IN TEMP	235°F	T0181A	T0182A	T0183A	T0184A

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: 3/3/2014	Tier	2		
Change: 2	Group	1		
	K/A	004 A3.15		
Level of Difficulty: 3	Importance Rating	3.5		

Chemical and Volume Control System: Ability to monitor automatic operation of the CVCS, including: PZR pressure and temperature

Proposed Question: 50

Given the following conditions:

- The Unit 1 Reactor Coolant System is currently at 125ºF with the Pressurizer solid.
- Reactor Coolant System (RCS) pressure is 340 psig.
- Train A Residual Heat Removal (RHR) is in service, with Letdown flow via HCV-0128, RHR LTDN FLO CTRL, (fully open) and PCV-131, LTDN HX OUT PRESS CTRL (in AUTO).
- RHR discharge pressure is 510 psig.

If RHR Pump 1-01 were to trip, what would be the expected RCS pressure response?

Reactor Coolant System pressure would…

A. …decrease to PCV-131 AUTO setpoint.

B. …decrease to RWST static head pressure.

C. …increase to RHR pump discharge relief setpoint.

D. …increase to PORV LTOP setpoint.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that the automatic setpoint on PCV-131 is lower than RCS pressure, however, the setpoint is RCS pressure plus RHR pump differential pressure.
B. Incorrect. Plausible if thought that the RWST was aligned in this condition and that RCS pressure would eventually lower, however, suction of the pump is aligned to the RCS.
C. Incorrect. Plausible because there are relief valves on both the suction and discharge side of the RHR pump, however, the suction side relief lifts at 450 psig while the discharge side relief setpoint is 600 psig. With anticipated pressure increase between 100 and 150 psig and current RCS pressure at 340 psig one could anticipate lifting of the suction side but not the discharge side relief.
D. Correct. With RCS temperature at 125°F and RCS pressure at 340 psig, tripping of the RHR Pump would cause an expected rise in RCS pressure between 100 and 150 psig. This would result in lifting of the PORV since the LTOP setpoint when less than 150°F is 375 psig per the TDM.

Technical Reference(s): IPO-005A, Section 3.3
TDM-301A, PORV LTOP Setpoints
LO21.SYS.RH1, Pages 21 & 22
LO21.SYS.RH1, Figure

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the actions for placing the plant in a solid condition in accordance with IPO-005, Plant Cooldown from Hot Standby to Cold Shutdown.

Question Source:
Bank ILOT5914
Modified Bank ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 5
55.43 ____

Comments / Reference: IPO-005A, Section 3.3	Revision: 25

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-005A
PLANT COOLDOWN FROM HOT STANDBY TO COLD SHUTDOWN	REVISION NO. 25 CONTINUOUS USE	PAGE 12 OF 171

3.3 Solid Plant Operation Precautions

3.3.1 The RHR System shall not be isolated from the RCS, unless there is a steam bubble in the pressurizer, or the charging pumps are stopped. At least one RHR suction path is required to be aligned to the RCS to ensure at least one suction relief is aligned. Maintaining a RHR suction relief available during water solid operation minimizes a challenge to the LTOP System.

3.3.2 When the plant is water solid and RCS pressure is being maintained by 1-PK-131, LTDN HX OUT PRESS CTRL, letdown flow will bypass the normal letdown orifices and 1-HC-128, RHR LTDN FLO CTRL should be in the full open position. During this mode of operation, all three letdown orifices should also remain open to align the letdown relief to the RCS.

3.3.3 When the RCS is in a solid condition, with stable pressure, a charging and letdown flow mismatch will exist that is approximately equal to seal return flow and any other RCS leakage path.

3.3.4 When RCS pressure is being maintained by 1-PK-131, LTDN HX OUT PRESS CTRL, changes to the flow rate throughout RHR loop by throttling of valves or starting and stopping RHR pumps will result in changes to RCS pressure. Stopping the RHR pumps may cause an increase in RCS pressure of between 100 and 150 psig.

3.3.5 When the last RCP is stopped, RCS pressure can quickly drop below 100 psig therefore,Hatch Closure DIDCP is verified to be in effect prior to securing the last RCP.

3.3.6 If all RCPs are stopped and the RCS is being cooled down, a non-uniform temperature distribution may occur in the Reactor Coolant loops. RCP restart should not be attempted unless a steam bubble is formed in the Pressurizer AND secondary water temperature of each SG <50°F above each of the RCS cold legs.

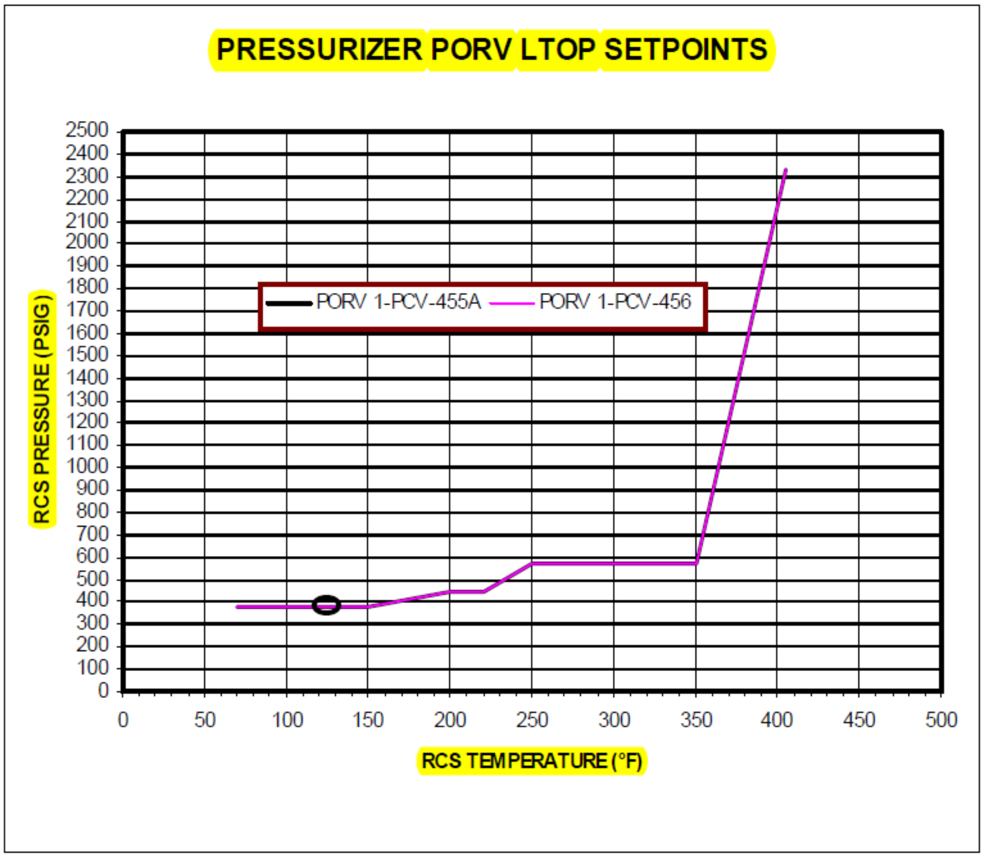
3.3.7 Extreme care should be taken when adjusting charging and letdown flow during solid plant operation. A flow mismatch between charging and letdown could cause a pressure transient that may result in an LTOP event.

3.3.8 1-PK-131, LTDN HX OUT PRESS CTRL should be placed in manual whenever an RHR pump or an RCP is started or stopped during solid plant operations.

Comments / Reference: TDM-301A, PORV LTOP Setpoints	Revision: 10

CPNPP TECHNICAL DATA MANUAL	UNIT 1	PROCEDURE NO. TDM-301A
RCS TEMPERATURE & PRESSURE LIMITS	REVISION NO. 10 INFORMATION USE	PAGE 6 OF 7

[L] PRESSURIZER PORV LTOP SETPOINTS



Pressurizer PORV Setpoints for LTOP Function	
Adjusted RCS Temperature (°F)	PORV No. 1 and No. 2 Final Setpoint (psig)
70	375
150	375

Comments / Reference: LO21.SYS.RH1, Page 21	Revision: 10/20/11

RHR PUMP SUCTION RELIEF VALVES (U-8708A&B)

The RHR System is designed for 600psig, thus when the RCS Hot Leg Recirculation Isolation Valves are open, the RHR System requires overpressure protection. The RHR Pump Suction Relief Valves have a design capacity of 900 gpm and discharge to the Pressurizer Relief Tank.

The relief setpoint is 450 psig and relief capacity requirements are calculated based on 2 analyzed situations:

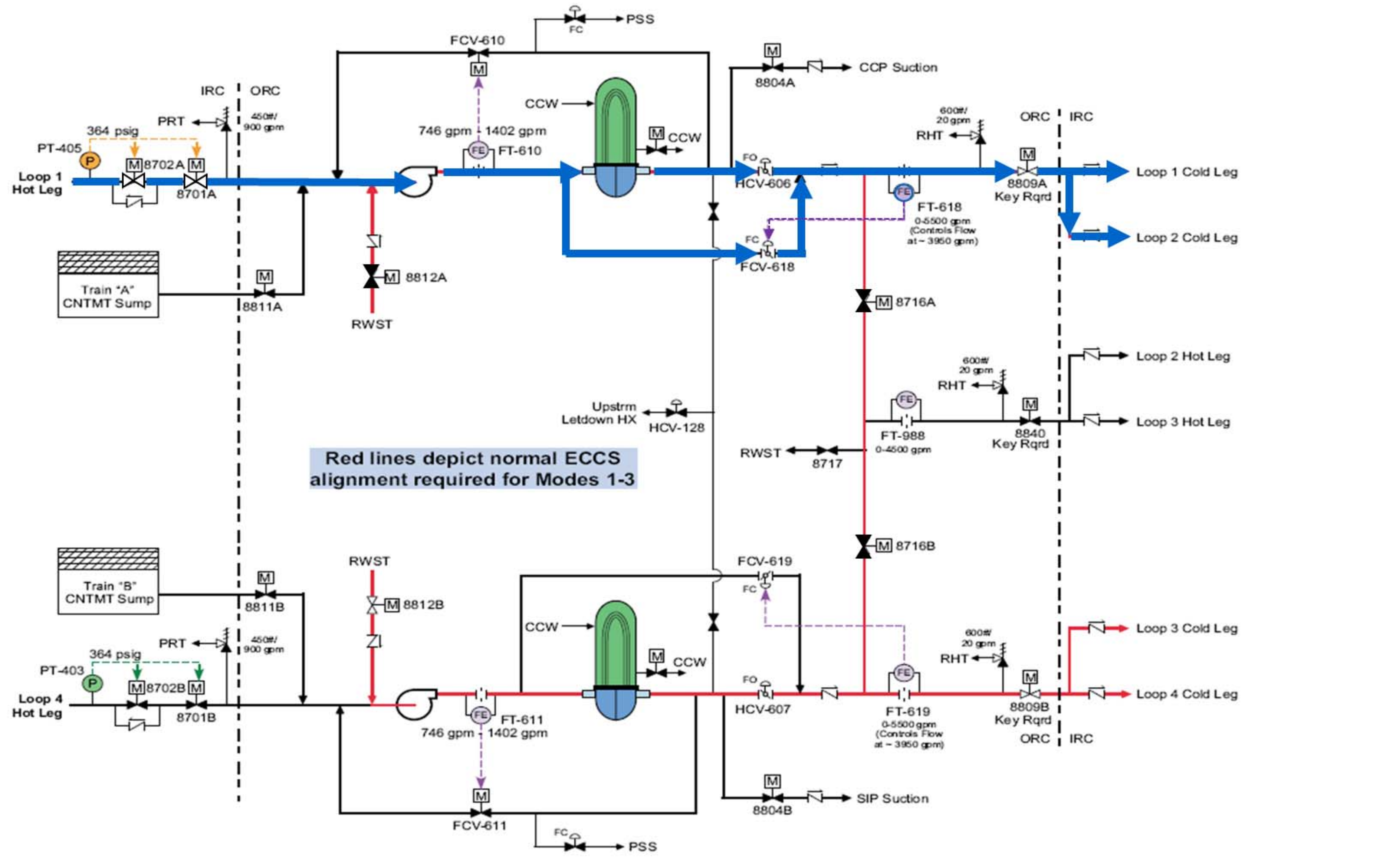
Comments / Reference: LO21.SYS.RH1, Page 22	Revision: 10/20/11

RHR COLD LEG INJECTION RELIEF VALVE (U-8856A&B)

The RHR Cold Leg Injection Relief Valves provide overpressure protection from the effects of slow thermal expansion of fluid trapped within the lines they protect. This condition could result from stopping flow during LOCA conditions when the RHR system would contain radioactive materials that would generate heat as they decay. The leg of piping bounded by the discharge check valve (downstream of the Flow Control Valve), u-8716A/B and u-8809A/B would be trapped, and as the temperature increases, the pressure would increase. This pressure increase could rupture piping if not relieved.

The relief valves also protect against the leakage of the Reactor Coolant System water past the check valves located upstream of the RHR System injection point. These valves are set to open when pressure reaches 600psig and have the capacity to relieve 20 gpm to the Recycle Holdup Tank.

Comments / Reference: LO21.SYS.RH1, Figure	Revision: 10/20/11



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	012 A2.02	
Level of Difficulty: 3	Importance Rating	3.6	

Reactor Protection System: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power

Proposed Question: 51

Given the following conditions on Unit 1:

- Solid State Protection System (SSPS) Train A is in the Slave Relay Testing alignment.
- One of the 48 VDC power supplies to the Train B SSPS just failed.

Which of the following identifies the impact on the Reactor Protection System and what action should be taken to mitigate the situation?

A. A Reactor trip should occur based on a General Warning Condition in both SSPS trains.
Ensure a Reactor Trip occurs and enter EOP-0.0A, Reactor Trip or Safety Injection.

B. Train B SSPS continues to have 48 VDC power from its other 48 VDC power supply.
Complete testing in the Train A SSPS while tracking repairs to the Train B power supply.

C. A General Warning Alarm for Train B SSPS will annunciate and Train B will be inoperable.
Stop all testing in the Train A SSPS until repairs are made to the failed power supply.

D. A single trip will be initiated in each trip function on Train B SSPS due to the power loss.
Stop all testing and BYPASS the individual channels on Train B SSPS. Place the Channels in the TRIP condition if not corrected within 72 hours.

Proposed Answer: A

Explanation:

A. Correct. The Train A alignment causes the Train A Solid State Protection System to be in a degraded condition with a General Warning Alarm. If both trains have a General Warning Condition a Reactor trip is generated and the actions would be to ensure the trip occurs and to perform EOP-0.0A Reactor Trip or Safety Injection actions.

B. Incorrect. Plausible because it could be thought that this was the reason for having two 48 VDC power supplies and would have no effect which would make the action appropriate, however, the loss of the power supply creates a second General Warning Alarm condition resulting in a Reactor Trip.

C. Incorrect. Plausible because it could be thought that Train A was not in a degraded condition and though this could cause a General Warning there would be no trip signal generated.

D. Incorrect. Plausible because it could be thought that this would just cause individual trips on each trip parameter on that channel and as long as another channel was not in trip then there would be no other affect, however, a General Warning Alarm for Train B will occur which will complete the logic to trip the Reactor.

Technical Reference(s): ALM-0064A, 1-ALB-6D, Window 2.5; SOP-711A, Step 5.4.2 NOTE & CAUTION — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Solid State Protection System and **PREDICT** the system response.

Question Source:
- Bank: ILOT8216
- Modified Bank: ______ (Note changes or attach parent)
- New: ______

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge: ______
- Comprehension or Analysis: X

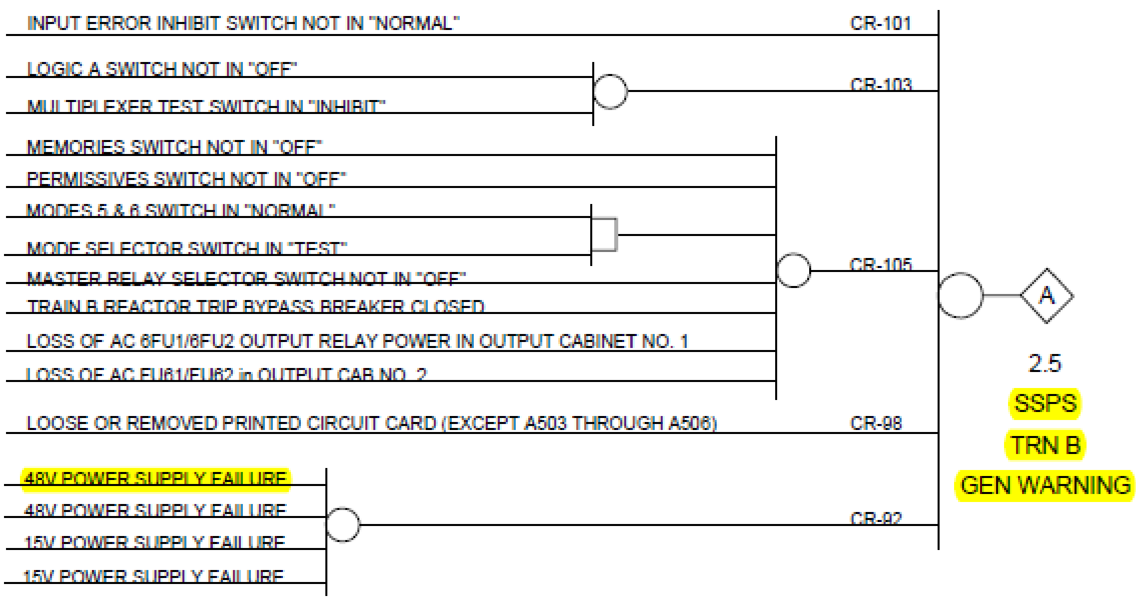
10 CFR Part 55 Content:
- 55.41: 7
- 55.43: ______

Comments / Reference: ALM-0064A, 1-ALB-6D, Window 2.5	Revision: 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 58 OF 147

ANNUNCIATOR NO.: 2.5

LOGIC:



ANNUNCIATOR NOM./NO.: SSPS TRN B GEN WARNING 2.5

PROBABLE CAUSE:

Surveillance testing
Loss of power
Internal power supply failure

NOTE: Controlled evolutions for authorized testing should not require an alarm response.

AUTOMATIC ACTIONS: None

NOTE:
- The SSPS trouble alarm generates a GENERAL WARNING condition on the associated train. If a GENERAL WARNING condition exists on both trains, a Reactor trip is actuated.
- If a GENERAL WARNING condition exists on both trains and power $<$P-9, no first out annunciator will be in alarm.
- If a GENERAL WARNING condition exists on both trains and power $\geq$ P-9, a RX $>$50% PWR TURB TRIP first out alarm will be illuminated.

Comments / Reference: SOP-711A, Step 5.4.2 NOTE & CAUTION	Revision: 9

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-711A
SOLID STATE PROTECTION SYSTEM	REVISION NO. 9 CONTINUOUS USE	PAGE 44 OF 96

5.4.2 Train A Normal Lineup to Slave Relay Testing Lineup

This section describes the steps to take Train A SSPS from a Normal Lineup to a Slave Relay Testing Lineup. The steps are to be performed at TBX-ESELSP-01A, SOLID STATE PROTECTION SYSTEM TRAIN A INPUT/LOGIC CABINET 1-SP-01A, unless otherwise indicated.

CAUTION: Except when the core is off-loaded, both trains of SSPS SHALL not be tested or disabled simultaneously.

NOTE: When SSPS is placed in the Slave Relay Testing Lineup, all input signals are defeated BUT output relays can be energized to cause actuations. The LCOAR must consider all required SSPS signals against the present plant mode.

A. PERFORM the following:

☐ 1) Pre-evolution briefing using Attachment 7.1.7

☐ 2) Prerequisite 2.1

☐ B. INITIATE LCOAR. (ALL SSPS input signals are defeated).

NOTE: DO NOT change any switch position in the following step without Shift Manager approval.

☐ C. VERIFY the control switch lineup on Attachment 7.1.3 is complete.

CAUTION: IF a GENERAL WARNING alarm is in on the opposite train, THEN the momentary GENERAL WARNING alarm will result in a Reactor Trip.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	1	
	K/A	061 K1.04	
Level of Difficulty: 3	Importance Rating	3.9	

Auxiliary/Emergency Feedwater System: Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: RCS

Proposed Question: 52

Given the following conditions:

- On Unit 1, ECA-2.1A, Uncontrolled Depressurization of All Steam Generators is in progress.
- Auxiliary Feedwater (AFW) flow is 100 gpm to each Steam Generator (SG).
- Reactor Coolant Pumps have been secured.

Which of the following describes the expected plant response to the AFW flow reduction and what actions are to be taken as SG pressures decrease?

A. Reactor Coolant System Hot Leg temperatures will eventually increase causing a loss of Natural Circulation and the crew will then raise AFW flow while continuing in ECA-2.1A, Uncontrolled Depressurization of All Steam Generators.

B. Reactor Coolant System Hot Leg temperatures will eventually increase causing a loss of Natural Circulation and the crew will then transition to FRH-0.1A, Response to Loss of Secondary Heat Sink.

C. The SGs will eventually become completely depressurized due to inadequate secondary heat sink and the crew will then transition to EOP-2.0A, Faulted Steam Generator Isolation.

D. The SGs will eventually become completely depressurized due to inadequate secondary heat sink and the crew will then transition to FRH-0.1A, Response to Loss of Secondary Heat Sink.

Proposed Answer: A

Explanation:

A. Correct. When AFW flow is reduced to 100 gpm, eventually Hot Leg temperatures will rise when SG inventory is depleted. AFW flow is raised to restore Natural Circulation.

B. Incorrect. Plausible because Hot Leg temperatures will begin to increase as SG inventory lowers, however, FRH-0.1A conditions are not met because the AFW flow reduction was controlled by the crew.

C. Incorrect. Plausible because the SGs depressurize as long as they are faulted, however, transition to EOP-2.0A is only performed when one Steam Generator re-pressurizes.

D. Incorrect. Plausible because the SGs depressurize as long as they are faulted, however, ECA-2.1A must be performed to completion unless a SG is isolated or tubes rupture.

Technical Reference(s): ECA-2.1A, Step 2; ECA-2.1A, Attachment 4, Step 2 Bases; ECA-2.1A, Attachment 1.A; FRH-0.1A, CSFST

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from ECA-2.1, Uncontrolled Depressurization of All Steam Generators, **STATE** the purpose/basis for the step(s).

Question Source:

Bank	ILOT8236	
Modified Bank		(Note changes or attach parent)
New		

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	5
55.43	

Comments / Reference: ECA-2.1A, Step 2	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-2.1A
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 4 OF 72

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: A minimum AFW flow of 100 gpm must be maintained to each SG with a narrow range level less than 43% (50% FOR ADVERSE CONTAINMENT).

NOTE: Shutdown margin should be monitored during RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Control AFW Flow To Minimize RCS Cooldown:	
	a. Check cooldown rate in RCS cold legs - LESS THAN 100°F/HR	a. Decrease AFW flow to 100 gpm to each SG. Go to Step 2c.
	b. Check narrow range level in all SGs - LESS THAN 60%	b. Control AFW flow to maintain narrow range level less than 60% in all SGs.
	c. Check RCS hot leg temperatures - STABLE OR DECREASING	c. Control AFW flow or dump steam to stabilize RCS hot leg temperatures.

Comments / Reference: ECA-2.1A, Attachment 4, Step 2 Bases	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-2.1A
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 44 OF 72

ATTACHMENT 4
PAGE 3 OF 31

BASES

In addition, as SG pressure and steam flow rate drop, RCS hot leg temperatures will stabilize and start increasing. The operator controls AFW flow or dumps steam to stabilize the RCS hot leg temperatures. This allows the safety injection flow to establish conditions for SI termination and minimizes thermal stresses that may be generated.

The operator controls AFW flow during subsequent procedure steps; therefore, this step has been identified as a Continuous Action Step.

Comments / Reference: ECA-2.1A, Attachment 1.A	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-2.1A
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 29 OF 72

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT PAGE FOR ECA-2.1A - UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS

1. SI REINITIATION CRITERIA

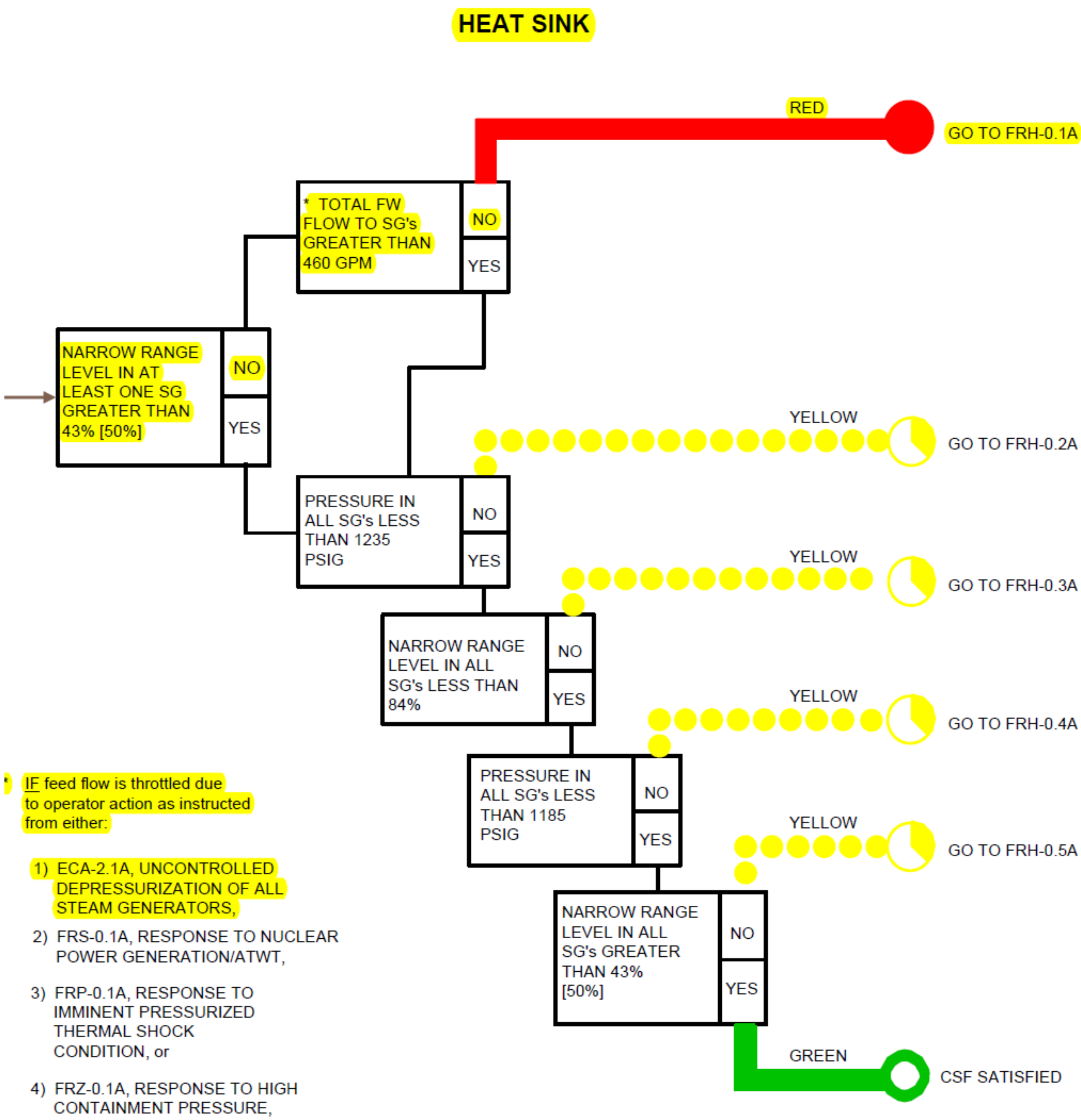
   Manually start ECCS pumps as necessary if EITHER condition listed below occurs:

   - RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)

2. EOP-2.0A TRANSITION CRITERIA

   IF any SG pressure increases at any time, except while performing ECCS Termination in Step 10 to 24, THEN go to EOP-2.0A, FAULTED STEAM GENERATOR ISOLATION, Step 1.

Comments / Reference: FRH-0.1A, CSFST	Revision: 8



* IF feed flow is throttled due to operator action as instructed from either:

1) ECA-2.1A, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS,
2) FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT,
3) FRP-0.1A, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, or
4) FRZ-0.1A, RESPONSE TO HIGH CONTAINMENT PRESSURE,

THEN FRH-0.1A does not need to be implemented.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/14	Tier	2	
Change: 2	Group	1	
	K/A	064 G 2.4.34	
Level of Difficulty: 2	Importance Rating	4.2	

Emergency Diesel Generator System: Emergency Procedures/Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects

Proposed Question: 53

Given the following conditions on Unit 1:

- The Unit 1 Control Room was evacuated due to smoke entering the area.
- ABN-905A, Loss of Control Room Habitability, Attachment 11, Local Start of Diesel Generators is being performed.
- A procedure CAUTION in ABN-905A warns against running the Auxiliary Oil Pump for longer than one minute in the HAND Mode.

Which of the following describes the potential impact on operations that the CAUTION is trying to prevent?

A. Extended operation in hand could result in the Discharge Oil Pressure Regulators sticking in the diverted position and insufficient oil may be supplied to the Diesel Generator on start.

B. Extended operation in hand could result in the Engine Lube Oil Pump Suction Relief sticking open and result in insufficient oil being supplied to the Diesel Generator on start.

C. The Lube Oil Strainer is not rated for extended periods of operation with the higher differential pressure when the oil is being diverted to the sump and damage to the strainer internals could occur.

D. Extended operation in hand could result in flooding the Turbocharger with oil and resultant ignition of the oil when the Diesel Generator is started which will damage the Turbocharger Bearings.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because problems have occurred with the oil pressure regulators sticking, however, not due to this operation. The concern is flooding the Turbocharger with oil and then igniting the oil when the Diesel starts causing exhaust pressure surges that damage the Turbocharger Bearings.
B. Incorrect. Plausible because this is a possible failure mode when the engine is operated in reverse, however, this is not the concern in this case.
C. Incorrect. Plausible because this is a possible failure mode, however, this is not the concern in this case.
D. Correct. The concern is flooding the Turbocharger with oil and then igniting the oil when the Diesel starts causing exhaust pressure surges that damage the Turbocharger Bearings.

Technical Reference(s) ABN-905A, Attachment 11 Attached w/ Revision: See Comments / Reference
LO21.SYS.ED1, Pages 41 & 44

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Loss Of Control Room Habitability in accordance with ABN-905, Loss Of Control Room Habitability.

Question Source: Bank ILOT8247
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ABN-905A, Attachment 11	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 52 OF 74

ATTACHMENT 11
PAGE 1 OF 3

LOCAL START OF DIESEL GENERATORS

1. Start Train A Diesel Generator as follows:

NOTE: With 1-HS-3413-3B, MASTER SWITCH in LOCAL or 43/1EG1, DG 1 BKR 1EG1 CTRL XFER switch in HSP, the Diesel Generator Auto Starts are defeated.

☐ a. Place 1-HS-3413-3B, MASTER SWITCH (DG local generator control panel) in LOCAL.

CAUTION: Do NOT run the Aux Lube oil pump in HAND for more than one (1) minute. This is to prevent flooding the turbo chargers with oil.

☐ b. Place 1-HS-3411-1, AUXILIARY LUBE OIL PUMP handswitch in HAND (DG local engine control panel) AND allow Lube Oil pressure to stabilize (40 - 65 psig). |

☐ c. Stop the Auxiliary Lube Oil Pump and place the handswitch in AUTO.

Comments / Reference: LO21.SYS.ED1, Page 41	Revision: 05/02/11

Diesel Generator Engine Lube Oil Pump Suction Relief Valve uDO-0155 (uDO-0255) is located by the lube oil sump tank and lifts at > 70 psig to relieve excess pressure back to the lube oil sump. The relief is provided to protect against a sudden pressure buildup in the suction in the event of a reverse rotation of the diesel engine. During engine operation, the lube oil pressure will be approximately 55 to 60 psig.

Comments / Reference: LO21.SYS.ED1, Page 44	Revision: 05/02/11

Diesel Generator Lube Oil Duplex Filter Differential Pressure Indicator u-PI-3411-2B (u-PI-3412-2B) on the engine control panel (0 to 60 psid) is used to evaluate the condition of the in-service filter. The maximum operating limit is 20 psid across a vessel.

Diesel Generator Lube Oil Duplex Filter Inlet And Outlet Pressure Indicators u-PI-3411-2H and u-PI-3411-2I (u-PI-3412-2H and u-PI-3412-2I), respectively, provide local indication (0 to 160 psig) at a gageboard on the engine skid.

Each filter has two pressure sensing taps, coming directly off of the vessels, for indication of individual vessel inlet and outlet pressure (0 to 160 psig). These pressure instruments, u-PI-3411-2D, -2E, -2F and -2G (u-PI-3412-2D, -2E, -2F and -2G) are mounted on a gageboard on the engine skid.

Diesel Generator Lube Oil Filter Differential Pressure Switch u-PS-3411-2A (u-PS-3412-2A) senses the differential pressure across the duplex filters and causes the HIGH ΔP LUBE FILTER alarm on the engine control panel if it exceeds 20 psid (PS-17C).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/14	Tier	2	
Change: 2	Group	1	
	K/A	078 A3.01	
Level of Difficulty: 3	Importance Rating	3.1	

Instrument Air System: Ability to monitor automatic operation of the IAS, including: Air pressure

Proposed Question: 54

Given the following conditions:

- Instrument Air (IA) Compressor 1-01 is operating as the LEAD compressor.
- IA Compressor 1-02 is in an AUTO-START condition as the BACKUP compressor.
- IA Compressor X-01 is in STANDBY and aligned to Unit 1 through Air Dryer X-01.
- The following sequence of events occur:
  - At 1415, 1-ALB-01, Window 2.4 – CNTMT INSTR AIR HDR PRESS LO, alarms as pressure drops to 84 psig.
  - At 1416, 1-ALB-01, Window 3.3 – INSTR AIR HDR PRESS LO, alarms as pressure drops to 85 psig.
  - All other Unit 1 Control Room alarms related to the IA System remain clear.
  - At 1420, a stuck-open relief valve on Air Dryer 1-01 reseats.
  - At 1422, both Instrument Air alarms (1-ALB-01, Windows – 2.4 and 3.3) clear.
  - At 1423, Instrument Air header pressure is 93 psig and slowly rising.

At 1423, assuming NO additional operator actions and with IA Compressor 1-01 running and loaded, which of the following is the status of IA Compressors 1-02 and X-01 in accordance with SOP-509A, Instrument Air System?

IA Compressor 1-02 is __________ and IA Compressor X-01 is __________.

A. running and loaded; running and loaded

B. running and loaded; shutdown

C. running and unloaded; running and unloaded

D. running and unloaded; shutdown

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, the BACKUP and STANDBY IA Compressors will both be running and loaded. Air pressure must reach 115 psig to place the compressors in an unloaded condition and then they will run for 20 minutes then shutdown to an Auto-Start condition.

B. Incorrect. Plausible because the LEAD compressor is running and loaded and given that the low pressure alarms are clear it could be thought that the STANDBY compressor would shutdown, however air pressure must reach 115 psig for both compressors to unload and then they must run unloaded for 20 minutes before shutting down.

C. Incorrect. Plausible because the BACKUP and STANDBY IA Compressors will be running, however, they will also be loaded until air pressure reaches 115 psig at which point they would unload.

D. Incorrect. Plausible because it could be thought that the BACKUP compressor will unload when the low pressure alarms clear and that the STANDBY compressor will shut down when the low pressure alarms clear.

Technical Reference(s): SOP-509A, Step 5.2.1.J NOTE; ALM-0011A, 1-ALB-01, Windows 2.4 & 3.3; SOP-509A, Step 5.4.1.H NOTE — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Instrument Air System.

Question Source:
- Bank: ILOT8178
- Modified Bank: ______ (Note changes or attach parent)
- New: ______

Question History: Last NRC Exam 2012

Question Cognitive Level:
- Memory or Fundamental Knowledge: ______
- Comprehension or Analysis: X

10 CFR Part 55 Content:
- 55.41: 10
- 55.43: ______

Comments / Reference: SOP-509A, Step 5.2.1.J NOTE	Revision: 22

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-509A
INSTRUMENT AIR SYSTEM	REVISION NO. 22 CONTINUOUS USE	PAGE 14 OF 271

NOTE: Upon restoration of power to the air compressor, local alarm lights will be illuminated and will reset automatically upon start.

5.2.1 F. PERFORM the following to ensure power available to Instrument Air Compressor 1-01:

- ☐ ENSURE 1EB3/11D/BKR (1CICO1), INSTRUMENT AIR COMPRESSOR 1-01 FEEDER BREAKER is racked into CONNECT AND Closed.
- ☐ ENSURE CP1-CIDSNB-03, INSTR AIR COMPRESSOR 1-01 CONTROL PNL DISCONNECT SWITCH (LOCAL) is ON.

☐ G. At Instrument Air Compressor 1-01, ENSURE 1-HS-3457A, LEAD/BACKUP SELECTOR SWITCH FOR INST AIR COMPRESSOR 1-01 is in the BACKUP position.

☐ H. ENSURE the UNLOAD/NORMAL Switch on the Instrument Air Compressor 1-01 Panel to UNLOAD.

☐ I. ENSURE OPEN 1CI-0006, INST AIR COMP 1-01 OUT ISOL VLV .

NOTE:

- If an air compressor operates unloaded for approximately 20 minutes, it will automatically shutdown to an Auto-Start condition. The air compressor is in an Auto-Start condition when the Automatic Operation light is ON.
- If an air compressor is in an Auto-Start condition, it will not start until low pressure is sensed (105 psig if in LEAD, 100 psig if in BACKUP). Once low pressure is sensed, the Compressor will start and load.

Comments / Reference: SOP-509A, Step 5.4.1.H NOTE	Revision: 22

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-509A
INSTRUMENT AIR SYSTEM	REVISION NO. 22 CONTINUOUS USE	PAGE 27 OF 271

NOTE: IF the function keys or arrow keys are not used for approximately 4 minutes, THEN the display will automatically return to the Main Screen.

5.4.1 H. At the Elektronikon Control Panel, using function keys AND arrow keys, SCROLL to set X-01 Instrument Air Compressor to either LEAD (Press. Band 1) OR STANDBY (Press. Band 2) as follows:

1. IF desired to return to the Mainscreen,
   THEN
   PERFORM the following:
   - ☐ a. DEPRESS the F1 function key (beneath << Menu >>).
   - ☐ b. Again, DEPRESS the F1 function key (beneath << Menu >>) to return to Menu.
   - ☐ c. DEPRESS the F1 function key (beneath << Mainscreen >>) to return to Mainscreen.
2. ☐ From the Mainscreen, DEPRESS the F1 function key (beneath << Menu >>).

NOTE: A hi-lited " ➜" next to each menu item shows what will be selected when depressing the tabulator key

3. ☐ Using the arrow keys located above AND below the tabulator key, SCROLL to << Modify Parameters >>.
4. ☐ DEPRESS the tabulator key to select << Modify Parameters >>.
5. ☐ Using the arrow keys located above AND below the tabulator key, SCROLL to << Configuration >>.
6. ☐ DEPRESS the tabulator key to select << Configuration >>.

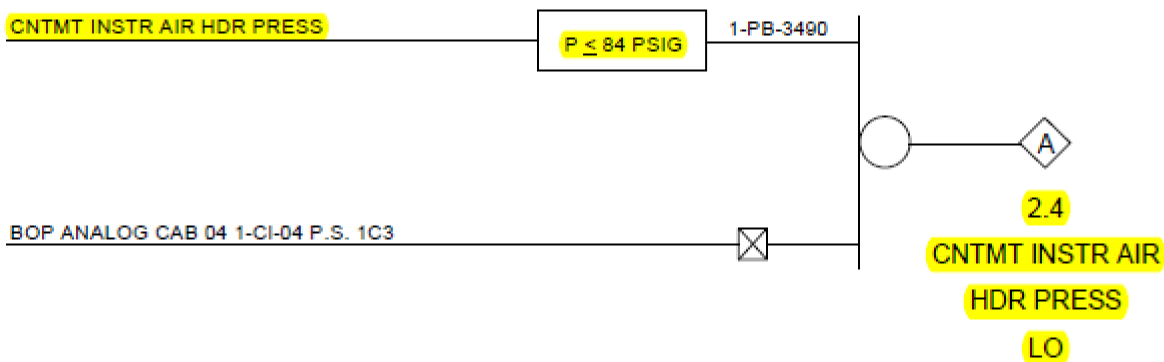
NOTE:
- IF << Press. Band 1 >> is indicated, THEN the Compressor is in LEAD, AND will control pressure between 105 psig and 115 psig.
- IF << Press. Band 2 >> is indicated, THEN the Compressor is in STANDBY, and will control between 95 psig and 115 psig.

Comments / Reference: ALM-0011A, 1-ALB-01, Windows 2.4 & 3.3	Revision: 10

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0011A
ALARM PROCEDURE 1-ALB-1	REVISION NO. 10	PAGE 45 OF 142

ANNUNCIATOR NO.: 2.4

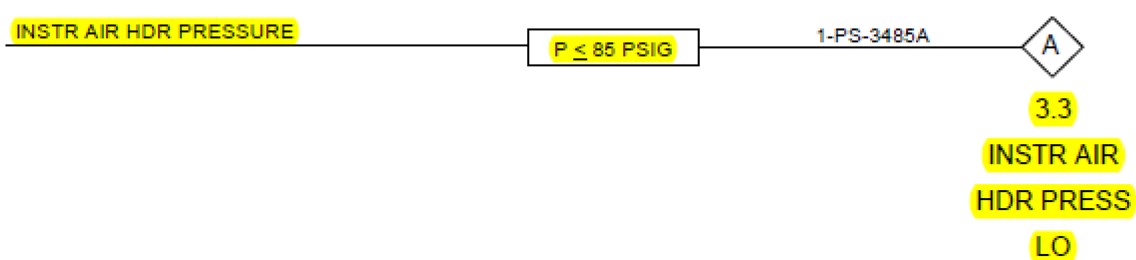
LOGIC:



CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0011A
ALARM PROCEDURE 1-ALB-1	REVISION NO. 10	PAGE 79 OF 142

ANNUNCIATOR NO.: 3.3

LOGIC:



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 2	Group	1	
	K/A	006 K4.16	
Level of Difficulty: 2	Importance Rating	3.2	

Emergency Core Cooling System: Knowledge of the ECCS design feature(s) and/or interlock(s) that provide for the following: Interlocks between RHR valves and RCS

Proposed Question: 55

Which of the following would PREVENT the Reactor Operator from manually opening 1/1-8701A, RHRP1 HL RECIRC ISOL VLV?

A. 1/1-8804A, RHRP 1 TO CCP SUCT VLV, is open.

B. 1/1-8809A, RHR TO CL 1&2 INJ ISOL VLV, is open.

C. 1/1-8811A, CNTMT SMP TO RHRP 1 SUCT ISOL VLV, is closed.

D. 1/1-8812A, RWST TO RHRP 1 SUCT VLV, is closed.

Proposed Answer: A

Explanation:

A. Correct. 1/1-8701A is interlocked with 1/1-8804A and prevents 1/1-8701A from opening when 1/1-8804A is in the OPEN position. These interlocks prevent a suction source from the RCS hot leg to the RHR pump from interfering with other RHR pump alignments.

B. Incorrect. Plausible if thought that 1/1-8701A is interlocked with 1/1-8809A, however, there are no interlocks associated with these valves and they remain open in all modes of plant operation with the exception of ECCS hot leg injection.

C. Incorrect. Plausible because 1/1-8701A is interlocked with 1/1-8811A, however, 1/1-8811A must be in the OPEN vice CLOSED position to prevent 1/1-8701A from opening.

D. Incorrect. Plausible because 1/1-8701A is interlocked with 1/1-8812A, however, 1/1-8812A must be in the OPEN vice CLOSED position to prevent 1/1-8701A from opening.

Technical Reference(s) LO21.SYS.RH1, Pages 15 & 18 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Residual Heat Removal system including interrelations with other systems to include interlocks and control loops.

Question Source: Bank ILOT7140

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __7__
55.43 ____

Comments / Reference: LO21.SYS.RH1, Page 15	Revision: 10/20/11

In order for the Reactor Operator to open the Train A RHR Pump Hot Leg Recirculation Isolation Valves (u-8701A and u-8702A), the following interlocks must be met:

- the Containment Sump to RHR Pump Suction Isolation Valve (u-8811A) must be CLOSED, and
- the RWST to RHR Pump Suction Valve (u-8812A) must be CLOSED, and
- the RHR Pump to CCP/SIP Suction Valve (u-8804A) must be CLOSED, and
- detected Reactor Coolant System pressure from pressure transmitter PT-405 (Train A) must be less than 364 psig, and
- the valve handswitch on CB-04 placed in its OPEN position.

The interlocks for the Train B valves utilize their train B counterpart. PT-403 would be used instead of PT-405. The above interlocks are bypassed when control is transferred to the Remote Shutdown Panel for u-8701A and u-8701B.

Comments / Reference: LO21.SYS.RH1, Page 18	Revision: 10/20/11

RHR TO COLD LEG INJECTION ISOLATION VALVE (U-8809A&B)

The RHR to Cold Leg Injection Isolation Valves provide a means to isolate the discharge of the RHR System from the Reactor Coolant System. These valves remain open in all modes of plant operation with the exception of ECCS Hot Leg Injection Mode.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/14	Tier	2	
Change: 2	Group	2	
	K/A	001 K2.01	
Level of Difficulty: 2	Importance Rating	3.5	

Control Rod Drive System: Knowledge of bus power supplies to the following: One-line diagram of power supply to MG sets

Proposed Question: 56

One method to remove power from the control rods during an Anticipated Transient without Trip is to remove power from the Reactor Trip Breakers (RTBs).

The power flow path to the Unit 1 RTBs is from 480V buses...

A. ...1B1/1B2 - Motor Breakers – Motor Generators – Generator Breakers – RTBs.

B. ...1B1/1B2 - Generator Breakers – Motor Generators – Motor Breakers – RTBs.

C. ...1B3/1B4 - Motor Breakers – Motor Generators – Generator Breakers – RTBs.

D. ...1B3/1B4 - Generator Breakers – Motor Generators – Motor Breakers – RTBs.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if a misconception of which non safety 480 VAC Buses are the power supplies as the remaining portion of the one-line diagram is correct.
B. Incorrect. Plausible if a misconception of which non safety 480 VAC Buses are the power supplies and a misconception to the order of the Generator Breakers and Motor Breakers existed.
C. Correct. The electrical one-line diagram to the Reactor Trip Breakers is 1B3/1B4 – Motor Breaker – Motor Generators – Generator Breaker – RTB.
D. Incorrect. Plausible if a misconception to the order of the Generator Breakers and Motor Breakers existed.

Technical Reference(s) LO21.SYS.CR1, Page 14
LO21.SYS.CR1, Figure 5
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Rod Control System.

Question Source: Bank ILOT6013
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

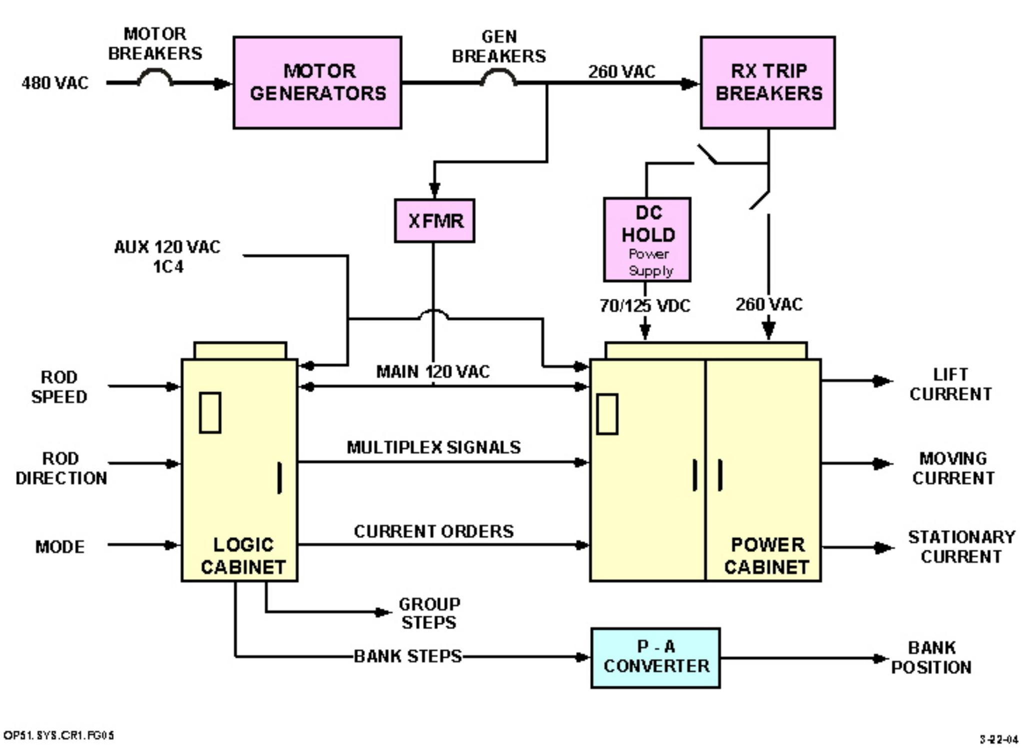
Question Cognitive Level: Memory or Fundamental Knowledge ___X___
Comprehension or Analysis _______

10 CFR Part 55 Content: 55.41 ______6______
55.43 __________

Comments / Reference: LO21.SYS.CR1, Page 14	Revision: 05/02/11
ROD CONTROL GENERAL DESCRIPTION The **Rod Control System** is housed in seven cabinets (located 832 safeguard building) with main power supplied through two paralleled, full capacity, rod drive **motor generators** (M-Gs). Power is fed from **<u>u</u>B3** and **<u>u</u>B4** (480 vac) to the M-Gs.	

Comments / Reference: LO21.SYS.CR1, Figure 5	Revision: 05/02/11

ROD CONTROL SYSTEM BLOCK DIAGRAM



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/14	Tier	2	
Change: 2	Group	2	
	K/A	002 K4.05	
Level of Difficulty: 3	Importance Rating	3.8	

Reactor Coolant System: Knowledge of RCS design feature(s) and/or interlock(s) that provide for the following: Detection of RCS leakage

Proposed Question: 57

Which of the following design features is used to assist in identifying the volume of Reactor Coolant System leakage from the Reactor Vessel Head when the Reactor Vessel Flange Leakoff Temperature High alarms in the Control Room?

Flow into the…

A. …Reactor Coolant Drain Tank.

B. …Containment Sump.

C. …Pressurizer Relief Tank.

D. …Volume Control Tank.

Proposed Answer: A

Explanation:

A. Correct. When conditions permit, Containment entry is made and Reactor Vessel Flange Leakoff is directed to the Reactor Coolant Drain Tank (RCDT).
B. Incorrect. Plausible because the Reactor Coolant Pump number 3 seal is directed to the Containment Sump while the number 2 seal is directed to the RCDT.
C. Incorrect. Plausible because numerous components drain to the PRT such as RHR suction reliefs and CVCS Letdown and seal return reliefs, however, Reactor Vessel Flange Leakoff is directed to the RCDT.
D. Incorrect. Plausible because the Reactor Coolant Pump Number 1 Seal is directed to the Volume Control Tank, however, the Number 2 Seal is directed to the RCDT and the Number 3 Seal is directed to the Containment Sump.

Technical Reference(s): ALM-0053A, 1-ALB-5C, Window 1.1; LO21.SYS.RC1, Pages 11 & 14; LO21.SYS.RC1, Figure 17; LO21.SYS.CS1, Pages 9 & 10

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	**EXPLAIN** the instrumentation and controls of the Reactor Vessel, Internals and Core Components System and **PREDICT** the system response. **ANALYZE** the response to Excessive Reactor Coolant Leakage in accordance with ABN-103, Excessive Reactor Coolant Leakage.
Question Source:	Bank ______ Modified Bank ______ (Note changes or attach parent) New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 3 55.43 ______

Comments / Reference: ALM-0053A, 1-ALB-5C, Window 1.1	Revision: 7

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0053A
ALARM PROCEDURE 1-ALB-5C	REVISION NO. 7	PAGE 7 OF 71

ANNUNCIATOR NOM./NO.: **RV FLANGE LKOFF TEMP HI** **1.1**

PROBABLE CAUSE:

High Containment temperature
Reactor vessel O-ring failure

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

1. VERIFY 1-TI-5400A, CNTMT AVE TEMP is <110°F.
   A. IF temperature is $\geq$110°F,
      THEN
      START an additional containment fan cooler per SOP-801A.
2. MONITOR 1-TI-401, RV FLANGE LKOFF TEMP.
3. CLOSE 1/1-8032, RV SEAL LKOFF VLV.
4. NOTIFY Chemistry to increase monitoring of containment atmosphere to detect possible outer O-ring failure.
5. PERFORM OPT-303 to determine leakage rate, as applicable.
6. WHEN conditions permit,
   THEN
   PERFORM a containment entry per STA-620 to align outer O-ring seal leakoff to RCDT.
   A. CLOSE 1RC-8069B, RV 1-01 HEAD INNER SL LKOFF ISOL VLV.
   B. OPEN 1RC-8069A, RV 1-01 HEAD OUTER SL LKOFF ISOL VLV.
   C. OPEN 1/1-8032, RV SEAL LKOFF VLV.

Comments / Reference: LO21.SYS.RC1, Page 11	Revision: 04/28/11

Reactor Vessel Flange Seal

For both Unit's, two self-energizing O-ring gaskets, constructed of silver plated Ni-Cr-Fe alloy, form the pressure boundary seal between the closure head and reactor vessel flanges. Each closure head flange has two machined-in ½ inch wide by ¼-inch deep grooves and 32 screw taps in the closure head flange. Sixteen (16) retainer clips and Allen head screws hold each O-ring in place. For O-ring leak detection, a tapped space outside the inner O-ring and a space outside the outer O-ring drain to one-inch piping connections on the lower reactor vessel flange. Piping from these drain connections extends through the missile barrier walls to allow operator interaction during plant operation.

Outside the missile barrier, each pipe from these connections reduces to ¾-inch and contains a manual isolation valve (uRC-8069A, B). Manual isolation valves provide local isolation in case of leakage from the respective O-ring. A common line joins the inner and outer O-ring leakoff connections. This common line contains a normally closed, manual ¾-inch tell-tale valve and a ¾-inch to ⅜-inch reducer. Valve u-8032 is downstream of the reducer, draining to the Reactor Coolant Drain Tank (RCDT). It is air-operated and manually controlled from Main Control Board CB05 to provide remote isolation capability for a leaking O-ring. This valve will fail open upon a loss of power or instrument air. A bottom-mounted, strap-on RTD (u-TE-0401) provides reactor vessel flange leak-off temperature indication at CB05.

During normal operation, inner O-ring isolation valve uRC-8069B is open. Should the inner O-ring leak, a high temperature alarm actuates at 140˚F on CB05, informing the operator of the leak. The operator monitors reactor vessel flange leak-off temperature and closes u-8032, isolating the leak. Procedure directs shutting manual isolation valve uRC-8069B and opening uRC-8069A, transferring RCS pressure boundary maintenance to the outer O-ring. Opening valve u-8032 then transfers leak detection to the outer O-ring.

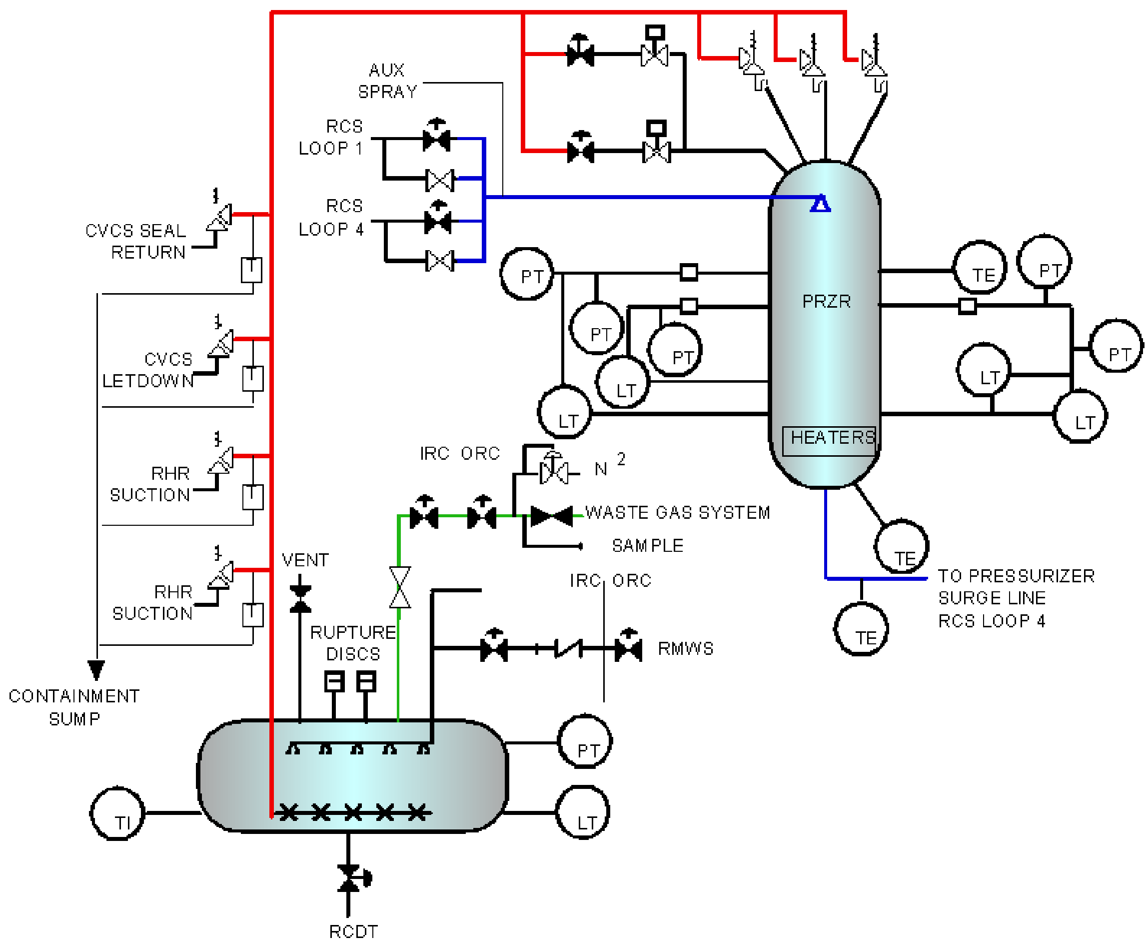
Comments / Reference: LO21.SYS.RC1, Page 14	Revision: 04/28/11

Number two seal is a face rubbing seal consisting of a carbon insert shrunk into a stainless steel ring. The carbon insert rubs against a rotating, chrome carbide coated, surface on a stainless steel runner. Leakage from the number two seal joins with the outer dam leakage of the number three seal and drains to the reactor coolant drain tank.

Number three seal is a face rubbing seal consisting of a carbon insert, with two concentric sealing faces or "dams", shrunk into a stainless steel seal ring. These dams rub against a rotating, chrome carbide coated, surface on a stainless steel runner. Clean water is injected between the dams on the seal ring at a pressure greater than the number two seal leak-off cavity. Part of the injected water flows past the outer dam where it joins the leakage from the number two seal and passes out of the pump, through the number two seal leak-off connection. The remainder of the injected water flows past the inner dam and drains through the number three seal leak-off piping to a floor drain sump inside containment.

Comments / Reference: LO21.SYS.RC1, Figure 17	Revision: 04/28/11

OP51.SYS.RC1



PRESSURIZER AND RELIEF TANK

FIGURE 17

Rev. 0

Comments / Reference: LO21.SYS.CS1, Pages 9 & 10	Revision: 04/28/11

through the #1 seal. The majority of the #1 seal leakoff flow (approximately 3 gpm per pump) is routed to a header with the seal leakoff from the other three reactor coolant pumps. The seal return flow, along with water from the excess letdown heat exchanger, is directed from the containment to the safeguards building, through the seal water return filter, the seal water heat exchanger, and then to the suction of the charging pumps. An alternate path can be aligned such that seal return flow out of the seal water heat exchanger is directed to the volume control tank through a spray nozzle.

A very small portion of the #1 seal leakoff on each reactor coolant pump (approximately 3 gallons per hour) leaks through the #2 seal and is routed to the reactor coolant drain tank in the containment building. A #3 seal provides a final barrier to prevent leakage of reactor coolant to the containment atmosphere. The #3 seal consists of two sealing faces called dams. Reactor makeup water from a standpipe is injected between the dams at a pressure greater than that of the #2 seal leakoff. Part of this clean injection water flows past the inner dam and is directed to the containment sump. The remainder of the injected water flows past the outer dam where it joins the leakage from the #2 seal and goes to the reactor coolant drain tank.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	2	
Change: 3	Group	2	
	K/A	011 K5.12	
Level of Difficulty: 3	Importance Rating	2.7	

Pressurizer Level Control System: Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: Criteria and purpose of PZR level program

Proposed Question: 58

Given the following conditions:

- Unit 1 is being operated at 40% power with Pressurizer level at program.
- RCS temperature is within 0.1ºF of T_{REF}.

Assuming all control systems are maintained on program, which of the following describes how Pressurizer level will change as Reactor Power is increased from 40% to 60% and the purpose of the level change?

Level increases from approximately…

A. …34% to approximately 43% to maintain a relatively constant mass in the RCS.

B. …34% to approximately 43% to maintain a relatively constant volume in the RCS.

C. …39% to approximately 46% to maintain a relatively constant mass in the RCS.

D. …39% to approximately 46% to maintain a relatively constant volume in the RCS.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible since programmed level is designed to maintain a constant mass as temperature changes and these values would be valid if the Pressurizer level program were similar to the feed pump speed program which starts at 20% power instead of no-load Tave at 0% power, but the actual values are 39% and 46%.

B. Incorrect. Plausible since these values would be valid if the Pressurizer level program were similar to the feed pump speed program which starts at 20% power instead of no-load Tave at 0% power, but the actual values are 39% and 46%.

C. Correct. Programmed level is between 25% and 60% for a Tave of no-load to full temperature, which is equivalent to 0% to 100% power. At 40% power, level should be 39% and at 60% power level should be 46%. The program is designed to allow a constant RCS mass as the RCS heats up and cools down.

D. Incorrect. Plausible since these are the correct values of Pressurizer level, but the program is designed to maintain a constant mass, not a constant volume.

Technical Reference(s) LO21.SYS.PP1, Page 5 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Pressurizer Pressure and Level Control System and **PREDICT** the system response.

Question Source: Bank ILOT6269
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments / Reference: LO21.SYS.PP1, Page 5	Revision: 05/05/11
Average Reactor Coolant System temperature (T_{AVG}) increases from 557°F at 0% reactor power to 585.4 °F (589.2°F) at 100% reactor power. Pressurizer level is programmed to change as a function of the T_{AVG} change. This allows the water in the RCS to expand as temperature increases from 0 - 100% power, raising pressurizer level from 25% to 60% without having to drain water from the RCS. In the same manner, pressurizer level is allowed to decrease during power reduction as the RCS water cools without the need to add water to make up for the contraction. The RCS **volume** is allowed to change as a result of temperature changes, while the **mass** of the RCS water remains constant. This reduces transient response time and the amount of water required to be processed during normal operations.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	2	
Change: 3	Group	2	
	K/A	015 A3.02	
Level of Difficulty: 4	Importance Rating	3.7	

Nuclear Instrumentation System: Ability to monitor automatic operation of the NIS, including: Maximum disagreement allowed between channels

Proposed Question: 59

Which of the following conditions would cause ALB-6D, Window 3.4 – PR CHAN DEV to annunciate?

Power Range channels are…

A. …> 2% between the highest and lowest channels.

B. …> 2% between the highest channel and the average of the four channels.

C. …> 5% between the highest and lowest channels.

D. …> 5% between the highest channel and the average of the four channels.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Technical Specification Surveillance Requirement (TS SR) 3.3.1.2 requires an NIS adjustment if calorimetric power exceeds NIS power by 2%, therefore, a misconception could exist that the alarm should be generated if any channels deviate more than 2% between the highest and lowest channels.

B. Incorrect. Plausible because Technical Specification Surveillance Requirement (TS SR) 3.3.1.2 requires an NIS adjustment if calorimetric power exceeds NIS power by 2%, therefore, a misconception could exist that the alarm should be generated if any channel deviates more than 2% from the average of the channels.

C. Correct. In accordance with ALM-0064A, Window 3.4 annunciates when the deviation between the highest and lowest channels is greater than 5%.

D. Incorrect. Plausible because in accordance with ALM-0064A, Window 1.4 and 2.4 annunciate when the deviation between the highest upper/lower detector and the average of the four upper/lower detectors is greater than 5%.

Technical Reference(s) ALM-0064A, 1.4, 2.4 & 3.4 Attached w/ Revision: See Comments / Reference

Technical Specification SR 3.3.1.2

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Excore Instrumentation System and **PREDICT** the system response.

Question Source: Bank

Modified Bank ILOT7133 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

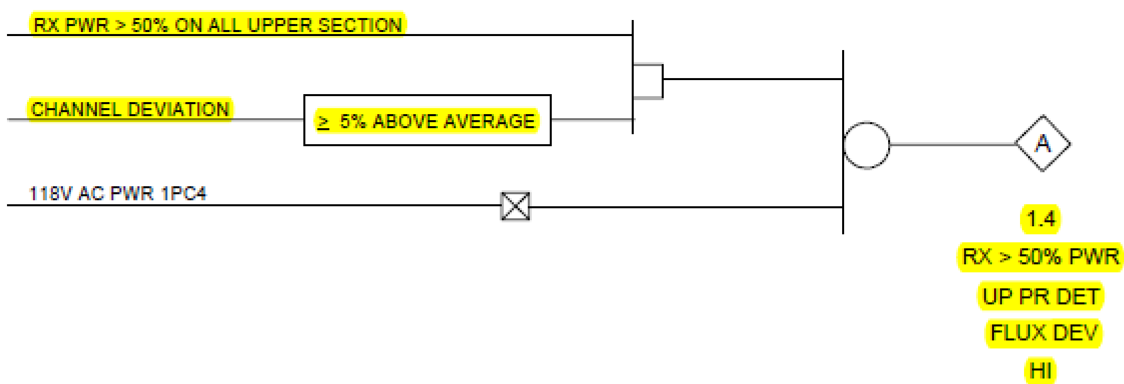
55.43

Comments / Reference: ALM-0064A, Window 1.4	Revision: 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 16 OF 147

ANNUNCIATOR NO.: 1.4

LOGIC:

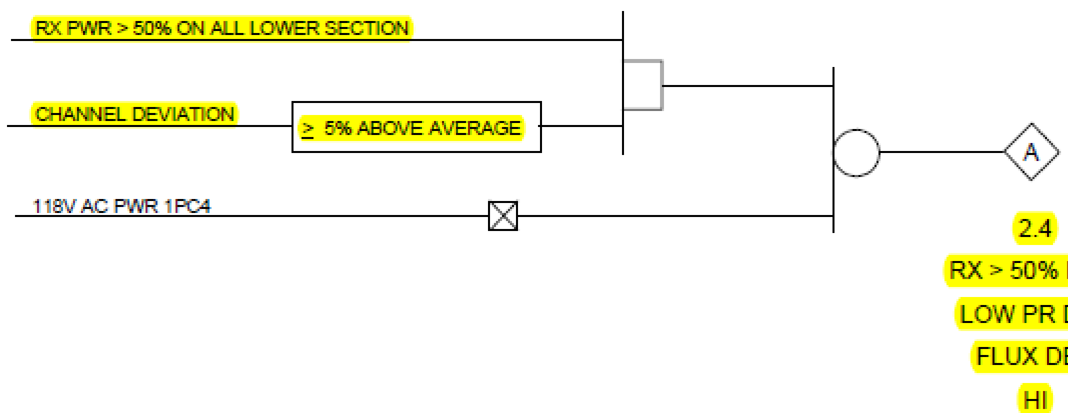


Comments / Reference: ALM-0064A, Window 2.4	Revision: 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 56 OF 147

ANNUNCIATOR NO.: 2.4

LOGIC:

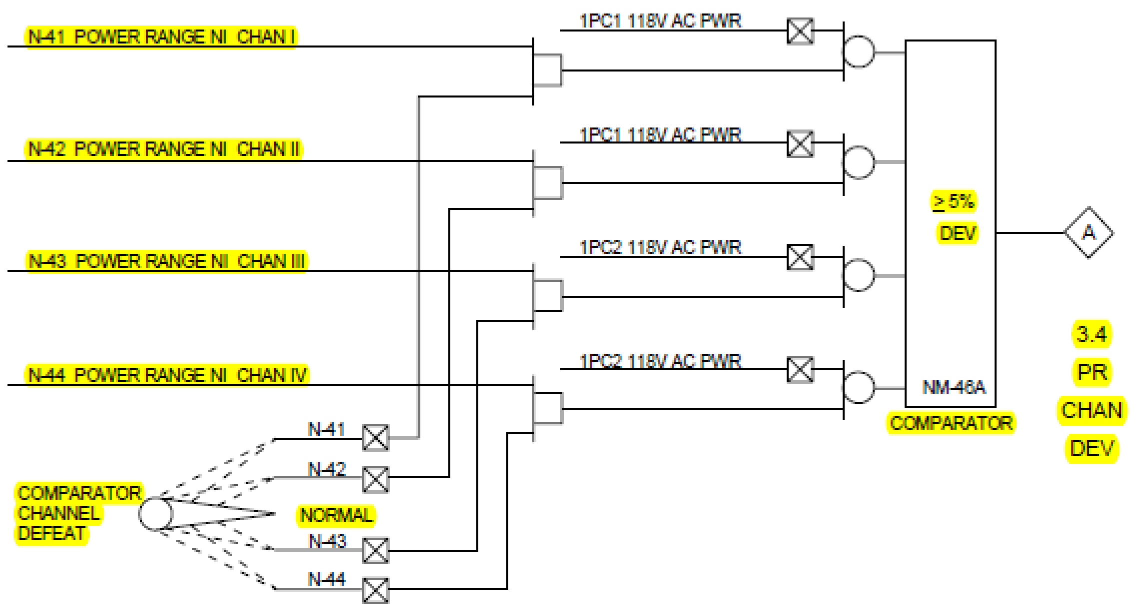


Comments / Reference: ALM-0064A, Window 3.4	Revision: 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 90 OF 147

ANNUNCIATOR NO.: 3.4

LOGIC:



Comments / Reference: ALM-0064A, Window 3.4	Revision: 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 91 OF 147

ANNUNCIATOR NOM./NO.: PR CHAN DEV 3.4

PROBABLE CAUSE:

Dropped rod
Instrument malfunction
Blown instrument or control power fuse
Xenon transient

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

1. Monitor the CTRL ROD POSN bezel for indication of rod misalignment.
   A. If one or more rods are misaligned ≥ 12 steps from their associated bank position, refer to ABN-712.
   B. If one or more rods are dropped, refer to ABN-712.

2. Verify the plant computer is available to monitor QPTR.
   A. From the NSSS MENU, select TILT REVIEW and verify the following TS points are < 1.02:

Excore Detector Power Tilts:	NIS N-43 (Q1)	NIS N-42 (Q2)	NIS N-44 (Q3)	NIS N-41 (Q4)
• Radial Upper Flux Tilt	U1159	U1160	U1161	U1162
• Radial Lower Flux Tilt	U1151	U1152	U1153	U1154

3. Perform OPT-302 to determine if a flux tilt is causing the alarm condition.

4. Monitor reactor power.
   - 1-NI-41B, PR POWER CHAN I
   - 1-NI-42B, PR POWER CHAN II
   - 1-NI-43B, PR POWER CHAN III
   - 1-NI-44B, PR POWER CHAN IV

   A. If one channel is indicating > 5% difference from the remaining operable channels, refer to ABN-703.

Comments / Reference: Technical Specification SR 3.3.1.2	Amendment: 161

RTS Instrumentation
3.3.1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2	----------NOTE---------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. ------------------------ Compare results of calorimetric heat balance calculation to NIS Power Range channel and N-16 Power Monitor channel outputs. Adjust NIS Power Range channel outputs if calorimetric heat balance calculation exceeds NIS Power Range channel outputs by more than +2% RTP. Adjust N-16 Power Monitor channel outputs if calorimetric heat balance calculation exceeds N-16 Power Monitor channel outputs by more than +2% RTP.	In accordance with the Surveillance Frequency Control Program.

Original Question: ILOT7133

Which of the following conditions would cause ALB-6D, Window 1.4, RX > 50% PWR UP PR DET FLUX DEV HI, to annunciate?

A. Power range upper detector deviation is > 2% between the highest and lowest upper detectors.

B. Power range upper detector deviation is > 2% between the highest upper detector and the average of the four upper detectors.

C. Power range upper detector deviation is > 5% between the highest and lowest upper detectors.

D. Power range upper detector deviation is > 5% between the highest upper detector and the average of the four upper detectors.

Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	2	
	K/A	016 K3.06	
Level of Difficulty: 3	Importance Rating	3.5	

Non-Nuclear Instrumentation System: Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: AFW system

Proposed Question: 60

Given the following conditions:

- Unit 1 is at 40% power.
- Instrument and Control (I&C) is performing a calibration on Steam Generator Narrow Range Level Channel, Loop 1, Protection Set II, CH 0519.
- During restoration the I&C Technician places the channel in TRIP.
- 1-TSLB-3, Window 2.2 – SG 1 LVL HI-HI LB-519A is LIT.
- 1-TSLB-5, Window 2.4 – SG 1 LVL LO-LO LB-519B is LIT.

Which of the following describes the automatic actuation signals generated if a loss of 1PC1, 118 VAC INSTRUMENT DISTRIBUTION PANEL (CHAN I) 1PC1 occurs?

A. Reactor Trip; only Motor Driven Auxiliary Feedwater Pumps 1-01 and 1-02 receive auto start signals.

B. Reactor Trip; Motor Driven Auxiliary Feedwater Pumps 1-01 and 1-02 receive auto start; Turbine Driven Auxiliary Feedwater Pump receives an auto start.

C. Turbine Trip and Feedwater Isolation.

D. Turbine Trip, Reactor Trip and Feedwater Isolation.

Proposed Answer: A

Explanation:

A. Correct. With TSLB-5 Window – 2.4 LIT, a loss of Panel 1PC1 will result in a 2 of 4 LVL LO-LO coincidence on one Steam Generator which will cause a Reactor Trip and only starts both MDAFW Pumps.

B. Incorrect. Plausible because with TSLB-5 Window – 2.4 LIT a loss of Panel 1PC1 will result in a 2 of 4 LVL LO-LO coincidence on one Steam Generator which will cause a Reactor Trip and only start both MDAFW Pumps. A LVL LO-LO coincidence on two Steam Generators would also auto start the TDAFW Pump.

C. Incorrect. Plausible because one Steam Generator LVL HI-HI TSLB is LIT. A failure of Panel PC1 could be thought to meet the 2 of 3 LVL HI-HI coincidence on one Steam Generator which will cause a Turbine Trip and Feedwater Isolation. However, the LVL HI-HI is 2 of 3 and the channel fed by Panel PC1 is not one of the 3 channels as it is used for normal control function.

D. Incorrect. Plausible because one Steam Generator LVL HI-HI TSLB is LIT. A failure of Panel PC1 could be thought to meet the 2 of 3 LVL HI-HI coincidence on one Steam Generator which will cause a Turbine Trip and Feedwater Isolation. However, the LVL HI-HI is 2 of 3 and the channel fed by Panel PC1 is not one of the 3 channels as it is used for normal control function. Additionally, it is plausible that a Reactor Trip signal would be generated as a Turbine Trip above 50% power would initiate an immediate Reactor Trip.

Technical Reference(s): INC-7296A, Section 9; ABN-603, Attachment 1; Technical Specification Table 3.3.1-1; Technical Specification Table 3.3.2-1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Main Feedwater System.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments / Reference: INC-7296A, Section 9	Revision: 7

CPNPP INSTRUMENT AND CONTROL MANUAL	UNIT 1	PROCEDURE NO. INC-7296A
COT & CHANNEL CALIBRATION STEAM GENERATOR NR LEVEL, LOOP 1, PROT. SET II, CH 0519	REVISION NO. 7	PAGE 28 OF 58

9.0 RESTORATION/POST WORK ACTIVITIES

9.1 Ensure ramp generator is setup per Step 8.1.10 and THEN perform Steps 8.2.1 through 8.2.5 recording the Channel Operational Test and Dynamic Functional Test "AS LEFT" values.

[V] 9.2 IF applicable, THEN verify appropriate SmartForm evaluation has been completed (See Step 8.2.6).

9.3 Disconnect the test equipment.

NOTE: Steps 9.4 through 9.9 place the loop in **TRIP** condition for a short duration to verify ALBs/TSLBs (input relay testing).

9.4 Notify the Reactor Operator that the loop will be placed in **TRIP**.

[V] 9.5 Verify that the following alarm AND trip status lights are **OFF**. (This step is applicable in modes 1, 2 or 3.)

CAUTION: IF ANY OF THE FOLLOWING ALARM OR TRIP STATUS LIGHTS ARE ON, PLACING THIS CHANNEL IN "TRIP" MAY CAUSE AN ENGINEERED SAFETY FEATURES ACTUATION OR REACTOR TRIP.

Comments / Reference: ABN-603, Attachment 1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 24 OF 34

ATTACHMENT 1
PAGE 1 OF 2

PROTECTION SET CABINETS AND MISCELLANEOUS LOADS

	DESCRIPTION	PROTECTION BUS			
		u PC1	u PC2	u PC3	u PC4
PROTECTION SET	RCL 1 FLOW	FT-414	FT-415	FT-416	-----------
	RCL 2 FLOW	FT-424	FT-425	FT-426	-----------
	RCL 3 FLOW	FT-434	FT-435	FT-436	-----------
	RCL 4 FLOW	FT-444	FT-445	FT-446	-----------
	BAT 1 LEVEL	XLT-102	-----------	-----------	XLT-104
	BAT 2 LEVEL	-----------	-----------	XLT-105	XLT-106
	PRZR LEVEL	LT-459(S)	LT-460(S)	LT-461(S)	-----------
	S/G LEVEL-WIDE RANGE	LT-501	LT-502	LT-503	LT-504
	S/G 1 LEVEL	LT-551(S)	LT-519(S)	LT-518	LT-517
	S/G 2 LEVEL	LT-529(S)	LT-552(S)	LT-528	LT-527
	S/G 3 LEVEL	LT-539(S)	LT-553(S)	LT-538	LT-537
	S/G 4 LEVEL	LT-554(S)	LT-549(S)	LT-548	LT-547
	RWST LEVEL	LT-930	LT-931	LT-932	LT-933
	SOURCE RANGE	N31	N32	-----------	-----------
	INTERMEDIATE RANGE	N35	N36	-----------	-----------
	POWER RANGE	N41	N42	N43	N44

Comments / Reference: Technical Specification Table 3.3.1-1, Item 14	Amendment: 161

RTS Instrumentation
3.3.1

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span
10. Reactor Coolant Flow - Low	1(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.6% of indicated loop flow (Unit 1) ≥ 88.8% of indicated loop flow (Unit 2)
11. Not Used					
12. Undervoltage RCPs	1(g)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 4753 V
13. Underfrequency RCPs	1(g)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 57.06 Hz
14. Steam Generator (SG) Water Level Low-Low (l)	1, 2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 37.5% of narrow range instrument span (Unit 1) (q)(r) ≥ 34.9% of narrow range instrument span (Unit 2)(q)(r)
15. Not Used.					

Comments / Reference: Technical Specification Table 3.3.1-1, Item 16	Amendment: 161

RTS Instrumentation
3.3.1

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
16. Turbine Trip					
a. Low Fluid Oil Pressure	1[(j)]	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 46.6 psig
b. Turbine Stop Valve Closure	1[(j)]	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open

Comments / Reference: Technical Specification Table 3.3.2-1, Item 5.b	Amendment: 161

ESFAS Instrumentation
3.3.2

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	1, 2[(j)]	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. SG Water Level -- High High (P-14)	1, 2[(j)]	3 per SG[(p)]	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤84.5% of narrow range span (Unit 1) [(q)(r)] ≤82.0% of narrow range span (Unit 2) [(q)(r)]
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

Comments / Reference: Technical Specification Table 3.3.2-1, Item 6.c	Amendment: 161

ESFAS Instrumentation
3.3.2

Table 3.3.2-1 (page 5 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1, 2, 3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Not Used.					
c. SG Water Level Low-Low	1, 2, 3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥37.5% of narrow range span (Unit 1) [q][r] ≥34.9% of narrow range span (Unit 2) [q][r]

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	2	
	K/A	033 G 2.4.31	
Level of Difficulty: 2	Importance Rating	4.2	

Spent Fuel Pool Cooling System: Emergency Procedures/Plan: Knowledge of annunciator alarms, indications, or response procedures

Proposed Question: 61

Given the following conditions:

- X-LS-4849A-1 and X-LS-4849A-2, Spent Fuel Pool level switches have both failed low.
- Annunciator 1-ALB-6B, Window 4.4 – SFPCS TRBL is in alarm.
- The Nuclear Equipment Operator reports that Window 1.1 – SFP CS PUMP 1 TRIP is in alarm on the Spent Fuel Pool Panel.

Which of the following identifies the effect on Spent Fuel Pool Cooling Water Pump X-01?

Spent Fuel Pool Cooling Water Pump X-01…

A. ...can only be started at the Spent Fuel Pool Panel.
Once started, all pump interlocks are restored.

B. ...can only be started at the breaker.
Once started, all pump interlocks are restored.

C. ...can only be started at the Spent Fuel Pool Panel.
In this condition, the pump will NOT be load shed on a Safety Injection or Blackout Sequencer Signal.

D. ...can only be started at the breaker.
In this condition, the pump will NOT be load shed on a Safety Injection or Blackout Sequencer Signal.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought the SFP Pump could be started with a low level condition.
B. Incorrect. Plausible because the SFP Pump can be started locally, however, in this condition all pump interlocks are bypassed.
C. Incorrect. Plausible because the SFP Pump will not load shed on a safety injection Signal or Blackout Sequencer Signal; however, with a low level condition the pump can only be started locally.
D. Correct. Given the conditions listed, operation of the SFP Pump is as stated.

Technical Reference(s): ABN-909, Step 4.3.1; ALM-0701, Window 1.1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Spent Fuel Pool Cooling and Cleanup System.

Question Source:	Bank	ILOT8409	
	Modified Bank		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments / Reference: ABN-909, Step 4.3.1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-909
SPENT FUEL POOL/REFUELING CAVITY MALFUNCTION	REVISION NO. 8	PAGE 25 OF 39

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If the both level channels are failed low, the affected Spent Fuel Cooling Pump cannot be started remotely. A local start removes all interlocks from the circuit. The pump will not be load shed on an SIS or BOS. The local start switches should not be used as a normal means of controlling SF system operation

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Locally or by Plant Computer verify affected Spent Fuel Pool level - NORMAL • L4800A • L4801A • L4802A • L4803A	GO TO Section 2.0

Comments / Reference: ALM-0701, Window 1.1	Revision: 5

CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-0701
ALARM PROCEDURE SPENT FUEL POOL PANEL	REVISION NO. 5	PAGE 7 OF 85

ANNUNCIATOR NOM./NO.: **SFP CW PUMP 1 TRIP** **1.1**

PROBABLE CAUSE:

Motor Overload
Blown Control Power Fuse
Breaker trip
SFP 1 low-low level

AUTOMATIC ACTIONS: None

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	2	
	K/A	035 K1.12	
Level of Difficulty: 2	Importance Rating	3.7	

Steam Generator System: Knowledge of the physical connections and/or cause-effect relationships between the SGS and the following systems: RPS

Proposed Question: 62

Given the following conditions:

- Unit 2 is operating at 100% power.
- 2-PT-0544, MAIN STEAM LINE 2-04 PRESSURE TRANSMITTER 0544 PROT CHAN 1 develops a leak causing 2-PI-544A, MSL 4 PRESS CHAN I to indicate 0 psig coincident with the following annunciators:
  - 2-ALB-8A, Window 4.7 – MSL 4 1 OF 3 PRESS LO.
  - 2-ALB-8A, Window 4.8 – SG 4 STM & FW FLO MISMATCH.
  - 2-ALB-8A, Window 4.12 – SG 4 LVL DEV.

Assuming NO operation action, which of the following identifies the expected plant response?

A. The Turbine will trip at 84% level in Steam Generator 2-04 causing a Reactor Trip.

B. The Turbine will trip at 81.5% level in Steam Generator 2-04 causing a Reactor Trip.

C. The Reactor will trip at 38% level in Steam Generator 2-04 causing a Turbine Trip.

D. The Reactor will trip at 35.4% level in Steam Generator 2-04 causing a Turbine Trip.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because if the channel failed high the SG level would rise and the turbine would trip at 84% SG level on Unit 1 causing a reactor trip.
B. Incorrect. Plausible because if the channel failed high on Unit 2 the SG level would rise and the turbine would trip at 81.5% SG level causing a reactor trip.
C. Incorrect. Plausible because channel 544 is the controlling channel which when it fails low causes feed pump speed to lower and at 38% level in Unit 1 SG 1-04 a reactor trip would be generated which would then cause a turbine trip.
D. Correct. Channel 544 is the controlling channel which when it fails low causes feed pump speed to lower and at 35.4% level on a Unit 2 SG a reactor trip would be generated which would then cause a turbine trip.

Technical Reference(s) ABN-707, Sections 2.1 & 2.2 Attached w/ Revision: See

Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Steam Flow Instrument Malfunction in accordance with ABN-707 Steam Flow Instrument Malfunction.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-707, Section 2.1	Revision: 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-707
STEAM FLOW INSTRUMENT MALFUNCTION	REVISION NO. 6	PAGE 3 OF 9

2.0 Steam Flow Instrument Malfunction

2.1 Symptoms

a. Annunciator Alarms

- MSL 1 1 OF 3 PRESS LO (8A-1.7)
- MSL 2 1 OF 3 PRESS LO (8A-2.7)
- MSL 3 1 OF 3 PRESS LO (8A-3.7)
- MSL 4 1 OF 3 PRESS LO (8A-4.7)

- SG 1 STM & FW FLO MISMATCH (8A-1.8)
- SG 2 STM & FW FLO MISMATCH (8A-2.8)
- SG 3 STM & FW FLO MISMATCH (8A-3.8)
- SG 4 STM & FW FLO MISMATCH (8A-4.8)

- SG 1 LVL DEV (8A-1.12)
- SG 2 LVL DEV (8A-2.12)
- SG 3 LVL DEV (8A-3.12)
- SG 4 LVL DEV (8A-4.12)

Comments / Reference: ABN-707, Sections 2.1 & 2.2	Revision: 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-707
STEAM FLOW INSTRUMENT MALFUNCTION	REVISION NO. 6	PAGE 4 OF 9

2.1 b. • One steam line pressure higher or lower than the others.

- u-PI-514A, MSL 1 PRESS CHAN I
- u-PI-515A, MSL 1 PRESS CHAN II
- u-PI-516A, MSL 1 PRESS CHAN IV
- u-PI-2325, MSL 1 PRESS

- u-PI-524A, MSL 2 PRESS CHAN I
- u-PI-525A, MSL 2 PRESS CHAN II
- u-PI-526A, MSL 2 PRESS CHAN III
- u-PI-2326, MSL 2 PRESS

- u-PI-534A, MSL 3 PRESS CHAN I
- u-PI-535A, MSL 3 PRESS CHAN II
- u-PI-536A, MSL 3 PRESS CHAN III
- u-PI-2327, MSL 3 PRESS

- u-PI-544A, MSL 4 PRESS CHAN I
- u-PI-545A, MSL 4 PRESS CHAN II
- u-PI-546A, MSL 4 PRESS CHAN IV
- u-PI-2328, MSL 4 PRESS

• Annunciator alarms on steam and feedwater flow mismatch and narrow range steam generator level at the same time could indicate a common instrument line failure. (See Attachment 2).

2.2 Automatic Actions

a. Failed channel selected for control

- Steam flow channel failing HIGH (steam flow failed high OR pressure compensation failed high) will cause feedwater flow to increase and feedwater pump speed to increase due to larger programmed Δp between feedwater pressure and steam pressure. Without operator action, turbine will trip at 84%(81.5%) SG level.

- Steam flow channel failing LOW (steam flow failed low OR pressure compensation failed low) will cause feedwater flow to decrease and feedwater pump speed to decrease to achieve a smaller Δp between feedwater pressure and steam pressure. Without operator action, reactor will trip at 38%(35.4%) steam generator level.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	2	
Change: 3	Group	2	
	K/A	041 A1.02	
Level of Difficulty: 3	Importance Rating	3.1	

<u>Steam Dump/Turbine Bypass Control System</u>: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure

Proposed Question: 63

Given the following conditions:

- Unit 1 was operating at 20% power when a manual Reactor Trip was initiated for a planned shutdown.
- 1-PK-507, STM DMP PRESS CTRL setpoint is zero and in AUTO.

Which of the following statements describes the Steam Dump System response if handswitch 43/1-SD, STM DMP MODE SELECT is placed in the Steam Pressure Mode?

A. The HI-1 and HI-2 bistables will actuate opening all Steam Dump Valves.
When Steam Header pressure reaches 1057 psig, all Steam Dump Valves will be closed by the P-12 interlock.

B. The HI-1 and HI-2 bistables will actuate opening all Steam Dump Valves.
When Steam Header pressure reaches 1057 psig, all but three Steam Dump Valves will close and temperature will be controlled by the remaining three Steam Dump Valves in automatic.

C. A proportional error signal will open all Steam Dump Valves.
When Steam Header pressure reaches 1057 psig, all but three Steam Dump Valves will close and temperature will be controlled by the remaining three Steam Dump Valves in automatic.

D. A proportional error signal will open all Steam Dump Valves.
When Steam Header pressure reaches 1057 psig, all Steam Dump Valves will be closed by the P-12 interlock.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because on a plant trip the HI-1 and HI-2 bistables input to the steam dump controller based on differential temperature between actual Tave and no-load Tave (557°F) at 1092 psig steam pressure, however once placed in the Steam Pressure Mode the HI-1 and HI-2 bistables are no longer in the dump control circuit. With setpoint at zero, the controller will be calling for the steam dumps to open to maintain 200 psig so all the Steam Dump Valves will open until RCS Tave reaches 553°F (1057 psig steam pressure) at which time all Steam Dump Valves will close.

B. Incorrect. Plausible because on a plant trip the HI-1 and HI-2 bistables input to the steam dump controller based on differential temperature between actual Tave and no-load Tave (557°F) at 1092 psig steam pressure, however once placed in the Steam Pressure Mode the HI-1 and HI-2 bistables are no longer in the steam dump control circuit. With setpoint at zero, the controller will be calling for the steam dumps to open to maintain 200 psig so all the Steam Dump Valves will open until RCS Tave reaches 553°F (1057 psig steam pressure) at which time all Steam Dump Valves will close, however, three dump valves (cooldown valves) may be bypassed to allow the cooldown valves to be open below 553°F.

C. Incorrect. Plausible because 1-PK-507 compares controller setpoint to steam header pressure and creates an error signal to position the Steam Dump Valves. With setpoint at zero the controller will be calling for the steam dumps to open to maintain 200 psig so all the dump valves will open until RCS Tave reaches 553°F (1057 psig steam pressure) at which time all dump valves will close, however three dump valves (cooldown valves) may be bypassed to allow the cooldown valves to be open below 553°F.

D. Correct. 1-PK-507 compares controller setpoint to steam header pressure and creates an error signal to position the Steam Dump Valves. With setpoint at zero, the controller will be calling for the steam dumps to open to maintain 200 psig so all the Steam Dump Valves will open until RCS Tave reaches 553°F (1057 psig steam pressure) at which time all Steam Dump Valves will close.

Technical Reference(s): LO21.SYS.SD1, Figure 3; Steam Tables — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: **DEMONSTRATE** an understanding of the components of the Steam Dump System including interrelations with other systems to include interlocks and control loops.

Question Source:
Bank ______
Modified Bank ILOT5907 (Note changes or attach parent)
New ______

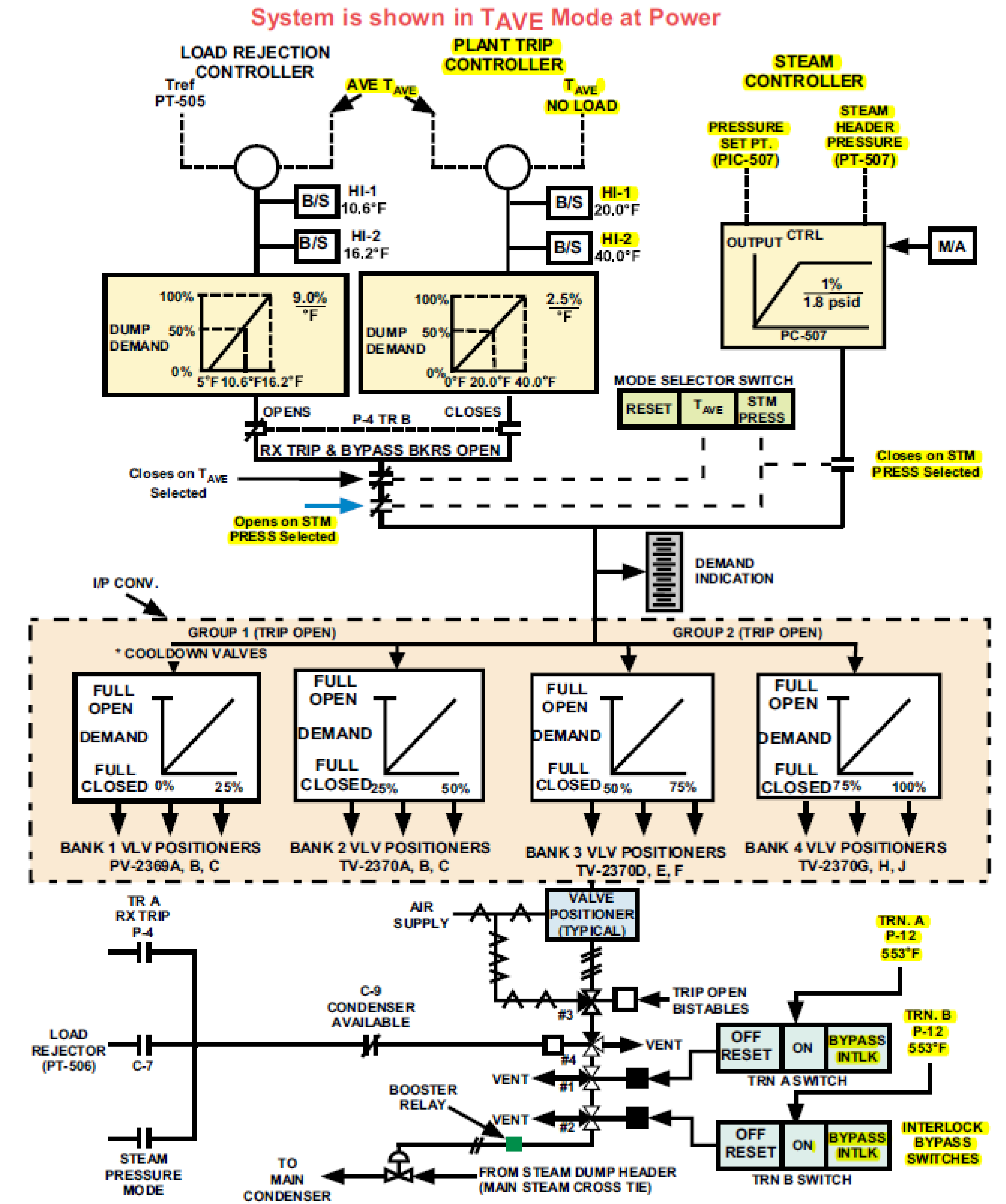
Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: LO21.SYS.SD1, Figure 3	Revision: 11/30/10

STEAM DUMP CONTROL SYSTEM



Original Question: ILOT5907

The plant was tripped from 20% Reactor Power. The steam dump system pressure controller potentiometer is set to zero and in AUTO. Which ONE of the following statements describes steam dump system response if the select switch is placed in the steam pressure mode?

A. The HI-1 and HI-2 bistables will actuate causing all steam dump valves to open. When RCS Tave reaches 553 degrees, all steam dump valves will be closed by the P-12 interlock.

B. A proportional error signal will open all steam dumps. When RCS Tave reaches 553 degrees, all but three valves close. PV-2369A, B, C will be controlled by the proportional controller.

C. A proportional error signal will open all steam dumps. RCS Tave reaches 553 degrees, all steam dump valves will be closed by the P-12 interlock.

D. The HI-1 and HI-2 bistables will actuate causing all steam dump valves to open. When Tave reaches NO-LOAD condition, valves will modulate to maintain NO-LOAD Tave.

Answer: C

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	2	
	K/A	045 A4.06	
Level of Difficulty: 4	Importance Rating	2.8	

Main Turbine Generator System: Ability to manually operate and/or monitor in the control room: Turbine stop valves

Proposed Question: 64

Given the following conditions:

- Unit 1 is responding to an inadvertent closure of Turbine High Pressure (HP) Stop Valve 1 in accordance with ABN-401, Main Turbine Malfunction.
- Engineering has recommended that HP Stop Valve 1 be opened from the Control Room using the HP Valve Test Display in accordance with OPT-217A, Turbine Overspeed Protection System Test.

Which of the following describes the expected sequence of component operation when using the HP Valve Test Display to open HP Stop Valve 1?

A. CLOSE HP Control Valve 1.
TRIP HP Stop Valve 1.
OPEN HP Stop Valve 1.
OPEN HP Control Valve 1.

B. TRIP HP Stop Valve 1.
CLOSE HP Control Valve 1.
OPEN HP Stop Valve 1.
OPEN HP Control Valve 1.

C. CLOSE HP Control Valve 1.
TRIP HP Stop Valve 1.
OPEN HP Control Valve 1.
OPEN HP Stop Valve 1.

D. TRIP HP Stop Valve 1.
CLOSE HP Control Valve 1.
OPEN HP Control Valve 1.
OPEN HP Stop Valve 1.

Proposed Answer: A

Explanation:

A. Correct. IAW OPT-217A, the correct sequence of valve operation is to close the HP CTRL VLV, trip the HP STOP VLV, then open the HP STOP VLV and open the HP CTRL VLV.
B. Incorrect. Plausible because with HP Stop Valve 1 closed it could be thought that tripping the stop valve first would prepare the stop valve for re-opening prior to closing the control valve.
C. Incorrect. Plausible because it could be thought that with the stop valve failing closed that the sequence of the control valve opening prior to opening the stop valve would ensure operation of the stop valve with a higher differential pressure.
D. Incorrect. Plausible because with HP Stop Valve 1 closed it could be thought that tripping the stop valve first would prepare the stop valve for re-opening prior to closing the control valve and that with the stop valve failing closed that the sequence of the control valve opening prior to opening the stop valve would ensure operation of the stop valve with a higher differential pressure.

Technical Reference(s) ABN-401, Step 9.3.14 / OPT-217A, Step 8.1.I NOTE — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of Main Turbine and its support systems.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments / Reference: ABN-401, Step 9.3.14	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-401
MAIN TURBINE MALFUNCTION	REVISION NO. 12	PAGE 55 OF 66

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- With a stop valve closed and its associated control valve open, the MSR for reheat supply can bypass the MSR tubes. This condition may result in unequal MSR load and subsequent turbine vibration or blade damage.
- The following steps, which attempt to reopen the affected valve, may result in rapid changes to turbine load. Reactor and turbine parameters should be closely monitored while these actions are in progress.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 14 With System Engineering assistance, restore the affected valve by performing the appropriate section (Section 8.1, 8.2 or 8.3) of OPT-217A/B in its entirety for the affected valve.	IF the affected valve still can NOT be opened AND the valve is one of the following, THEN isolate EHC fluid to the affected valve per Attachment 6: • HP Control Valve • LP Control Valve • LP Stop Valve

Comments / Reference: OPT-217A, Step 8.1.I NOTE	Revision: 17

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-217A
TURBINE OVERSPEED PROTECTION SYSTEM TEST	REVISION NO. 17	PAGE 9 OF 44
	CONTINUOUS USE	

NOTE:

- After the test is complete the TEST COMPLETE Bar will turn red.
- The Test Bar associated with the valve being tested will turn yellow while it is being tested.
- Program Step Description

  Main Program
  Step 1: "Close Control Valve"
  Step 2: "Trip Stop Valve"
  Step 3: "Verify Stop Vlv Closed"
  Reset Program
  Step 51: "Reset Stop Valve"
  Step 52: "Close MSR Htg Stm Vlv"
  Step 53: "Open Stop Valve"
  Step 54: "Open CV & Htg Stm Vlv"
- A runtime fault will clear out any other selected HP valve test selections. Any remaining valve tests will have to re-selected and run per section 8.1 or 8.3.

8.1 I. On the HP Valve Test Display in the HP Valve Test Section, VERIFY the Test Bars for the valve being tested turns yellow AND valves cycle as follows:

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	2	
Change: 3	Group	2	
	K/A	086 K4.01	
Level of Difficulty: 4	Importance Rating	3.1	

Fire Protection System: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Adequate supply of water for FPS

Proposed Question: 65

Given the following condition:

- Both Units are in MODE 1.

In accordance with SOP-904, Fire Protection Main Water Supply and Fire Pumps System, which of the following is the minimum required level in each Fire Water Storage Tank?

A. 53%.

B. 82%.

C. 85%.

D. 95%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this is the minimum required level for the Condensate Storage Tank. This is a plausible misconception as the CST is one of the very large outside storage tanks.
B. Correct. In accordance with SOP-904 and OWI-104-18. With level below 82% the tank is considered impaired.
C. Incorrect. Plausible because this is the minimum level for the Fire Water Storage Tanks per the OWI-104-18 NEO perimeter logs, to alert the NEO that the tank should be filled prior to decreasing to the minimum level.
D. Incorrect. Plausible because this is the minimum level in the Refueling Water Storage Tank. This is a plausible misconception as the RWST is one of the very large outside storage tanks.

Technical Reference(s): SOP-904, Section 4.1; OWI-104-18; Technical Specification LCO 3.5.4 Bases; Technical Specification LCO 3.7.6

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operations of the Fire

Protection System.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: SOP-904, Section 4.1	Revision: 15

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT COMMON	PROCEDURE NO. SOP-904
FIRE PROTECTION MAIN WATER SUPPLY AND FIRE PUMPS SYSTEM	REVISION NO. 15 CONTINUOUS USE	PAGE 7 OF 208

4.0 LIMITATIONS/NOTES

4.1 Limitations

The Fire Suppression Water System must be operable at all times with the following equipment:

- At least 3 Fire Suppression Pumps, each with a capacity of 2000 GPM, with their discharge aligned to the Fire Suppression Water Supply Header,
- Separate water supplies, each with a minimum level of 464,400 gal, (approximately 82% by pump house level gauge; approximately 39 ft at tank gauge).

Comments / Reference: OWI-104-18	Revision: 60

PERIMETER			Units	Low	High
ALTERNATE GENERATOR TO 6.9 KV SWGR TRANSFER SWITCH [15]	CP1-EPDSNA-01		SAT	SAT	SAT
ALTERNATE GENERATOR TO 6.9 KV SWGR TRANSFER SWITCH [16]	CP1-EPDSNA-01		SAT	SAT	SAT
U1 APG1 GENERAL AREA INSPECTION [17]			SAT	SAT	SAT
U1 APG1 TEMP BATTERY CHARGER [18]			SAT	SAT	SAT
U1 APG1 FUEL OIL LEVEL [19]			SAT	SAT	SAT
U1 APG1 ENGINE HEATERS [20]			SAT	SAT	SAT
U1 APG2 GENERAL AREA INSPECTION [17]			SAT	SAT	SAT
U1 APG2 TEMP BATTERY CHARGER [18]			SAT	SAT	SAT
U1 APG2 FUEL OIL LEVEL [19]			SAT	SAT	SAT
U1 APG2 ENGINE HEATERS [20]			SAT	SAT	SAT
U1 APG3 GENERAL AREA INSPECTION [17]			SAT	SAT	SAT
U1 APG3 TEMP BATTERY CHARGER [18]			SAT	SAT	SAT
U1 APG3 FUEL OIL LEVEL [19]			SAT	SAT	SAT
U1 APG3 ENGINE HEATERS [20]			SAT	SAT	SAT
HEAT TRACE PANEL X-HT-2 [21]	CPX-EPDPNB-21	ABN-912	SAT	SAT	SAT
SWIS TRASH RACK CLEAR OF DEBRIS [22]			SAT	SAT	SAT
SWIS GENERAL AREA/ EQUIP INSPECTION [23]			SAT	SAT	SAT
XFMR TXB 38 LIQUID LEVEL [24]	X-LIS-4091		SAT	SAT	SAT
XFMR TXB38 PRESSURE [25]	X-PI-4091		PSIG	-10	10
XFMR TXB38 LIQUID (HIGH) TEMPERATURE [11]	X-TIS-4091		DEG C		70
XFMR TXB38 LIQUID (CURRENT) TEMP	X-TIS-4091		DEG C		70
FPH LOCAL CTRL PNL X-01 [26]	CPX-FPCPLV-01		SAT/UNSAT	SAT	SAT
FPH FWSTK X-01 LVL [27]	X-LI-4077B	ABN-901	%	85	100
FPH FWSTK X-02 LVL [27]	X-LI-4078B	ABN-901	%	85	100
FPH ELECTRIC FIRE PUMP AVAILABLE [28]		ABN-901	SAT	SAT	SAT

Comments / Reference: Technical Specification LCO 3.5.4 Bases	Revision: 68

RWST
B 3.5.4

BASES

ACTIONS (continued)

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Control Board indication may be used in the surveillances of the required indicated RWST water volume. The indicated level of 95%, which includes

Comments / Reference: Technical Specification LCO 3.7.6	Amendment: 161

CST
3.7.6

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ 53%.	In accordance with the Surveillance Frequency Control Program.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2013	Tier	3	
Change: 2	Category	1	
	K/A	G 2.1.26	
Level of Difficulty: 2	Importance Rating	3.4	

<u>Conduct of Operations</u>: Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen)

Proposed Question: 66

Which of the following describes the requirements necessary to meet the definition of a Confined Space in accordance with STA-628, Confined Space Entry?

Large enough to be entered with the…

A. …entire body, limited entry and/or exit, and not designed for continuous occupancy.

B. …head and shoulders, limited entry and/or exit, and not designed for continuous occupancy.

C. …entire body, limited exit conditions, and designed for sporadic occupancy.

D. …head and shoulders, limited exit conditions, and designed for sporadic occupancy.

Proposed Answer: A

Explanation:

A. Correct. As defined in STA-628.
B. Incorrect. Plausible because the last 2 conditions are correct, however, it must be large enough to be entered with the entire body.
C. Incorrect. Plausible because the 1st condition is correct, however, the last 2 conditions are not.
D. Incorrect. Plausible if thought that accessibility via the head and shoulders were sufficient to define as a confined space, however, none of the conditions listed is correct.

Technical Reference(s) STA-628, Step 4.3 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source: Bank ILOT8431
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __10__
55.43 ______

Comments / Reference: ABN-628, Step 4.3	Revision: 11

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-628
CONFINED SPACE ENTRY	REVISION NO. 11 / INFORMATION USE	Page 3 of 40

4.0 DEFINITIONS/ACRONYMS

4.1 Attendant – A person normally assigned to remain at the entrance point providing constant communication with those in the confined space and notification to the Control Room in the event of an emergency. In special circumstances, remote monitoring may be used in lieu of the presence of the attendant at the confined space entrance.

4.2 Class C Atmosphere – An atmosphere in which all of the following conditions exist:

4.2.1 Oxygen percentage no less than 19.5%, nor greater than 23.5%.

4.2.2 Flammability percentage less than 10% of the Lower Explosive Limit (LEL) for those flammable hazard(s) which are present.

4.2.3 Toxicity concentration less than the Threshold Limit Value (TLV) and/or Permissible Exposure Limit (PEL) for those toxic substances which are present.

4.3 Confined Space – Any space in which all of the following conditions exist:

4.3.1 Large enough to be entered with the entire body.

4.3.2 Limited entry and/or exit.

4.3.3 Not designed for continuous occupancy.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	3	
Change: 1	Category	1	
	K/A	G 2.1.3	
Level of Difficulty: 2	Importance Rating	3.7	

Conduct of Operations: Knowledge of shift or short-term relief turnover practices

Proposed Question: 67

Given the following conditions:

- You are on watch in the Control Room as the BOP with both Units at 100% power.
- Shifts are 12 hours long and all shifts are manned to the minimum composition in accordance with ODA-102, Conduct of Operations.
- Your relief is not on site for Shift Turnover.

Which of the following describes the ODA-102, Conduct of Operations guidance in this situation?

Shift composition may...

A. ...NOT drop below the minimum unless an operator exceeds 12 hours on watch. Turnover your watchstation to the on-coming RO and depart.

B. ...NOT drop below the minimum as a result of an on-coming watchstander being absent. Remain on watch.

C. ...be one less than the minimum for two hours while attempting to find a replacement. Turnover your watchstation to the on-coming RO and attempt to contact a replacement.

D. ...be one less than the minimum for two hours. Turnover your watchstation to the on-coming RO but remain on site in standby until a replacement is found.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because 12 hours is the maximum shift time excluding turnover, however, because the oncoming shift member is late or absent the position should not be unmanned.
B. Correct. Per the guidance in ODA-102, Item 13.
C. Incorrect. Plausible because shift composition can be one less than minimum for two hours, however, this does not apply when an oncoming shift member is late or absent.
D. Incorrect. Plausible because shift composition can be one less than minimum for two hours, however, this does not apply when an oncoming shift member is not yet present.

Technical Reference(s) ODA-102, Attachment 8A, Item 13 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **CONDUCT** shift relief and turnover in accordance with station procedures; **VERIFYING** that an adequate number of qualified personnel are available for turnover and **ENSURING** that all personnel are properly relieved.

RECOGNIZE the conditions under which the Operations crew may be less than the minimum requirement.

Question Source: Bank ILOT5824
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: ODA-102, Attachment 8A, Item 13	Revision: 26

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 26 INFORMATION USE	PAGE 37 OF 37

ATTACHMENT 8.A
PAGE 3 OF 3

[C] MINIMUM SHIFT CREW COMPOSITION
[00021, 00044, 00136, 01078, 05131, 05699, 06025, 06029, 06771, 07194, 08649, 22609, 22739, 23344]

[C] (12) A site Fire Brigade of at least five members shall be maintained onsite at all times. The Fire Brigade shall not include the SM and the four other members of the minimum Operations shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. The Fire Brigade may be less than the minimum requirements for a period of time not to exceed 2 hours to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

[C] (13) The Operations shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

Examination Outline Cross-reference:		Level	RO	SRO
Rev. Date: 3/4/2014		Tier	3	
Change: 1		Category	1	
		K/A	G 2.1.4	
Level of Difficulty: 3		Importance Rating	3.3	

Conduct of Operations: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: 68

Given the following conditions:

- A Nuclear Regulatory Commission (NRC) Licensed Reactor Operator (RO) has just been convicted of a felony.
- An appeal has been filed on their behalf.

Which of the following is required regarding notification of the NRC?

A. If the conviction is upheld following appeal, the RO must notify the NRC in writing within 30 days.

B. The Shift Manager is responsible for notifying the NRC in writing within 30 days.

C. The RO is responsible for notifying the NRC in writing within 30 days.

D. If the conviction is upheld following appeal, the Plant Manager must notify the NRC in writing within 30 days.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the RO must notify the NRC in writing within 30 days, however, there is no allowance for any appeal time.
B. Incorrect. Plausible because the NRC must be notified in writing within 30 days, however, it is the individual license holder's responsibility to provide that information.
C. Correct. As required per STA-501, Attachment 8.B and 10CFR55.53(g).
D. Incorrect. Plausible because the NRC must be notified in writing within 30 days, however, it is the individual license holder's responsibility to provide that information.

Technical Reference(s) STA-501, Attachment 8.B Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given an event related to system operation/status, **CLASSIFY** which events must be reported to external agencies, with respect to written reports.

Question Source:	Bank	ILOT4548	
	Modified Bank		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments / Reference: STA-501, Attachment 8.B — Revision: 17

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-501
NONROUTINE REPORTING	REVISION NO. 17 INFORMATION USE	PAGE 25 OF 196

ATTACHMENT 8.B
PAGE 12 OF 25

REPORT	SOURCE OF REQUIREMENT	TIMING	REFER TO NONROUTINE REPORT DESCRIPTION
• Events covered in paragraph (ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, it is not required to report an event pursuant to paragraph (ix)(A) of this section if the event results from: *(1)* A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or *(2)* Normal and expected wear or degradation.	10CFR50.73(a)(2)(ix)(B)	60 day written report	NR-13
Submit supplemental information as a supplement to a previously submitted LER	10CFR50.73(c)	As specified in the LER	NR-13
Notification of change in licensed operator status (reassignment, termination or permanent disability)	10CFR50.74(a) 10CFR50.74(b) 10CFR50.74(c) 10CFR55.21 10CFR55.25	Written report within 30 days of learning of or change of diagnosis	NR-14
Felony conviction of a licensed operator	10CFR55.53(g) 10CFR55.5(b)(2)(iv)	Written report within 30 days	NR-15

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/4/2014	Tier	3	
Change: 2	Category	2	
	K/A	G 2.2.39	
Level of Difficulty: 3	Importance Rating	3.9	

Equipment Control: Knowledge of less than or equal to one hour Technical Specification action statements for systems

Proposed Question: 69

Unit 1 is in MODE 6 and the Shift Chemist reports that the boron concentration from the last sample of the refueling canal is less than required by Technical Specification LCO 3.9.1, Boron Concentration.

Which of the following describes the action requirements of Technical Specification LCO 3.9.1, Boron Concentration?

A. Within one hour verify all dilution paths are isolated. (1CS-8455 or valves 1CS-8560, 1-FCV-111B, 1CS-8441, and 1CS-8453.)

B. Immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate boration of the Reactor Coolant System.

C. Within one hour initiate boration until the boron concentration is within limits specified in the COLR.

D. Immediately suspend all movement of fuel assemblies in the Refueling Canal and restore the boron concentration to within its limit within 4 hours.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because positive reactivity additions must be immediately suspended and isolating dilution flow paths would assist in performing this action, however, this action is required to be performed immediately.
B. Correct. As outlined in Technical Specification LCO 3.9.1.
C. Incorrect. Plausible because Boration must be performed to restore boron concentration to greater than the limit specified in the COLR, however, it must be immediately restored to within limits.
D. Incorrect. Plausible because movement of fuel assemblies must be suspended, however, boron concentration must be immediately restored to within limits.

Technical Reference(s) Technical Specification LCO 3.9.1 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Spent Fuel Pool Cooling and Cleanup system including Technical Specifications, TRM and ODCM.

Question Source: Bank ILOT5517
Modified Bank ____________ (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____________

Comments / Reference: Technical Specification LCO 3.9.1	Amendment 161

Boron Concentration
3.9.1

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of all filled portions of the Reactor Coolant System, the refueling canal, and the refueling cavity, that have direct access to the reactor vessel, shall be maintained within the limit specified in the COLR.

-----NOTE-----
While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier	3	
Change: 3	Category	2	
	K/A	G 2.2.37	
Level of Difficulty: 3	Importance Rating	3.6	

Equipment Control: Ability to determine operability and/or availability of safety related equipment

Proposed Question: 70

Which of the following describes equipment that is available but NOT operable?

A. Safety Injection Accumulator parameters have been verified in accordance with IPO-001A, Plant Heatup from Cold Shutdown to Hot Standby.

B. Main Steam Isolation Valve closure times have been verified in accordance with IPO-002A, Startup from Hot Standby.

C. Auxiliary Feedwater Flow Control and Isolation Valve positions have been verified in accordance with IPO-003A, Power Operations.

D. Safety Injection train alignment has been verified in accordance with IPO-010A, RCS Reduced Inventory Operations.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a misconception could exist that the actions taken in IPO-001A for the SI accumulators are to place the accumulators in an Available status, but the actions are actually to ensure operability of the accumulators.

B. Incorrect. Plausible because a misconception could exist that the actions taken in IPO-002A for the MSIVs are to place the accumulators in an Available status, but the actions are actually to ensure operability of the MSIVs.

C. Incorrect. Plausible because a misconception could exist that the actions taken in IPO-001A for the AFW flow control and isolation valves are to place the AFW system in an Available status, but the actions are actually to ensure operability of the AFW system.

D. Correct. The SI train alignment actions in IPO-010A are to ensure that the SI train is Available, but the pump breaker is racked out ensuring that the SI train is not operable.

Technical Reference(s): IPO-010A, Attachment 1; IPO-001A, Step 5.3.15, IPO-002A, Step 5.1.3, IPO-003A, IPO-003A, Step 5.1.7 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Reactor Coolant System including Technical Specifications, TRM and ODCM.

Question Source:
Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments / Reference: IPO-010A, Attachment 1	Revision: 18

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-010A
REACTOR COOLANT SYSTEM REDUCED INVENTORY OPERATIONS	REVISION NO. 18 CONTINUOUS USE	PAGE 128 OF 195

ATTACHMENT 1
PAGE 5 OF 12

[C] SHIFTLY CHECKLIST

3.0 Equipment Operability / Availability

[C] A. VERIFY the RHR heat exchanger inlet temperature <72°F above the outlet temperature for the inservice heat exchanger.

- 1-TR-612, RHR HX 1 IN TEMP/RHR HX 1 OUT TEMP ______/______ Initials Date
- 1-TR-613, RHR HX 2 IN TEMP/RHR HX 2 OUT TEMP ______/______ Initials Date

CAUTION:
- The 6.9 KV feeder breakers for the SIPs shall remain racked out with the Reactor Vessel Head on.
- Aligning a flowpath from the RWST to the RCS may result in a level increase in the RCS due to gravity drain.

NOTE: The safety injection train is considered available even though the 6.9 KV supply breaker for the pump will remain racked out and will only be racked in if needed for emergency make-up.

[C] B. ENSURE that one train of Safety Injection is available by performing the following steps:

1) VERIFY no work in progress or active clearances associated with work activities affect the selected train availability. ______/______ Initials Date

2) VERIFY that the 6.9 KV bus supplying the selected pump is energized. ______/______ Initials Date

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 12/17/2013	Tier	3	
Change: 0	Category	3	
	K/A	G 2.3.5	
Level of Difficulty: 2	Importance Rating	2.9	

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: 71

Given the following conditions:

- A Portable Frisker is being used to perform a whole body frisk.
- Background radiation is at 150 counts per minute.

Which of the following is the minimum count rate at which an individual is considered to be contaminated in accordance with STA-653, Contamination Control Program?

A. 175 counts per minute

B. 200 counts per minute

C. 225 counts per minute

D. 250 counts per minute

Proposed Answer: D

Explanation:
A. Incorrect. Plausible if actual level of 100 cpm above background cannot be recalled.
B. Incorrect. Plausible if actual level of 100 cpm above background cannot be recalled.
C. Incorrect. Plausible if actual level of 100 cpm above background cannot be recalled.
D. Correct. With a background radiation of 150 cpm + a detected radiation level of 100 cpm above background = 250 cpm.

Technical Reference(s) STA-653, Step 6.6.2 & Attachment 3 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** how to monitor personnel and personal items for contamination, including the use of friskers and personnel contamination monitors.

Question Source: Bank ILOT8316
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11
55.43

Comments / Reference: STA-653, Step 6.6.2	Revision: 16

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-653
CONTAMINATION CONTROL PROGRAM	REVISION NO. 16 / INFORMATION USE	Page 12 of 19

6.6.7 Protective Clothing worn inside a Contaminated Area should be removed at the step off pad.
[CR-2011-005658]

6.6.8 Attachment 2 provides guidance for donning and removing PCs.

[C] 6.7 Personnel Monitoring
[00816]

6.7.1 Contamination monitoring requirements should be posted at the exit of Satellite/Alternate RCA's.
[CR-2011-005658]

6.7.2 Unless otherwise posted or authorized, all personnel shall monitor themselves after handling contaminated materials or exiting a contaminated area, at the nearest available frisker or PCM, and when exiting at the access control point.

6.7.3 The frisker is most commonly used for monitoring after exiting a contaminated area or after handling contaminated material. Frisking should be done with a background count rate of less than 300 counts per minute (cpm).

Comments / Reference: STA-653, Attachment 3	Revision: 16

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-653
CONTAMINATION CONTROL PROGRAM	REVISION NO. 16 INFORMATION USE	Page 18 of 19

ATTACHMENT 3
PAGE 1 OF 2

GUIDELINES FOR PERSONAL MONITORING

Monitoring With a Frisker

NOTE: Due to background radiation levels some friskers may indicate a background count rate greater than 300 cpm. These friskers may be used to perform a gross contamination check. In-plant low background frisker stations are provided as necessary.

1. Ensure meter is turned on and the scale switch is set at X1. Observe background level momentarily.

2. Without picking up the probe, frisk both sides of one hand. The probe should be about ½ inch away from the surface area being frisked.

3. Pick up probe and frisk remainder of body, scanning at a slow rate. Special attention shall be given to the face, soles of feet, hands, knees, posterior, and any surface left exposed while wearing protective clothing and dosimetry.

4. If an increase in the count rate is noted (visual or audible), return the probe to the spot and verify count rate. A significant and abrupt rise/drop in the count rate may indicate the presence of a DRP. Notify Radiation Protection.

5. If the frisker alarms or a continuous count rate of 100 cpm above background or greater is noted, remain at that point and notify, or have a co-worker notify, Radiation Protection for assistance. If contamination is not detected, proceed as usual.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	3	
Change: 2	Category	3	
	K/A	G 2.3.14	
Level of Difficulty: 2	Importance Rating	3.4	

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities

Proposed Question: 72

Given the following condition:

- Unit 2 is in MODE 6 and Containment Fan Coolers are being alternated in accordance with SOP-801B, Containment Ventilation System.

In accordance with SOP-801B, Containment Ventilation System, which of the following describes the CAUTION applicable to alternating the running Containment Fan Coolers?

A. Alternating Fan Coolers can cause changes in indicated radiation levels due to noble gases in stagnant pockets of air.

B. Always stop the running fan prior to starting the standby fan to protect the discharge ductwork.

C. Always stop the running fan prior to starting the standby fan to protect the suction ductwork.

D. Monitor to ensure air flows into Containment from the Equipment Hatch and not from Containment out of the Equipment Hatch.

Proposed Answer: A

Explanation:

A. Correct. The CAUTION is concerned with stagnant pockets of air with noble gases that will be introduced into the Containment atmosphere and the potential to cause automatic Containment Isolation.

B. Incorrect. Plausible because it could be thought that restrictions on the number of fans discharging into the ductwork existed, however, there is no CAUTION pertaining to this issue in this procedure and the only CAUTION in this section refers to stagnant pockets of air with noble gases that will be introduced into the Containment atmosphere.

C. Incorrect. Plausible because it could be thought that restrictions on the number of fans taking suction from a suction ductwork existed, however, there is no common suction ductwork and the only CAUTION in this section refers to stagnant pockets of air with noble gases that will be introduced into the Containment atmosphere.

D. Incorrect. Plausible because this is a concern in ventilation changes that change air flow out of or into Containment, however, these fans recirculate air and alternating these fans does not change the air flow through the Equipment Hatch.

Technical Reference(s) SOP-801B, Step 5.1.3 CAUTION Attached w/ Revision: See
LO21.SYS.CL1, Page 8 Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basis for the precautions and limitations, and given major procedure steps relative to the Containment Ventilation system, **PLACE** them in the proper sequence for SOP-801, Containment Ventilation System.

Question Source: Bank ILOT8246
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 9
55.43 ______

Comments / Reference: SOP-801B, Step 5.1.3 CAUTION	Revision: 7

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. SOP-801B
CONTAINMENT VENTILATION SYSTEM	REVISION NO. 7 CONTINUOUS USE	PAGE 14 OF 48

[C] 5.1.3 Alternating Containment Recirculation Units

CAUTION:
- Alternating these cooling units may change indicated radiation levels inside containment due to mixing of noble gases from stagnant areas of air. Radiation levels reaching High Alarm on Containment Air Gaseous (2-RE-5503) OR Particulate Monitors (2-RE-5502) will cause a Containment Ventilation Isolation (CVI).
- IF CACRS Fans are reduced to only two in service, THEN CACRS Fans 1 and 2 should remain in service together OR Fans 3 and 4 should remain in service. This is to ensure BOTH of the CACRS Fans associated with at least one Condensate Measuring Tank are in service (reference ODA-308-3.4.15)

This section describes the steps to alternate running Containment Air Cooling AND Recirculation units.

Comments / Reference: LO21.SYS.CL1, Page 6	Revision: 05/02/11

Containment Air Cooling and Recirculation System

The Containment Air Cooling and Recirculation System is designed to circulate cool air throughout the Containment structure, equipment rooms, and cubicles. The system functions to remove all heat released to the Containment by the reactor, steam generators and related equipment during all normal plant operations and following a loss of offsite power. Heat is dissipated to the Ventilation Chilled Water System in order to maintain ambient temperature ≤120°F, thereby protecting equipment and structures. The system provides a supply of tempered air to the Control Rod Drive Mechanism Ventilation System and the Reactor Coolant Pipe Penetration Cooling System. The system consists of four (4) 33% capacity cooling units and fans, each capable of handling 65,000 cfm of air.

Each cooling unit, referred to as Containment Fan Coolers, consists of eight (8) cooling coils stacked two high in a rectangular array. The structure also supports a vane axial, direct drive fan. This arrangement allows warm air to be drawn through the coils by the fan, cooled, and delivered to the main supply plenum located on the 884’ elevation, which is connected to ductwork that supplies air throughout the Containment Building. Cool air is supplied to the open areas and the equipment rooms to ensure mixing of the Containment atmosphere. No return ductwork is installed. The warm air rises through openings and gratings in floors and cubicle walls. Two cooling units are located on the 860' level of the Containment, each with a fan mounted above it to blow cooled air up to the main supply plenum. Two units are located on the 905' level of the Containment with fans mounted below to supply cooled air down to the main supply plenum. Pneumatic dampers are installed at the outlet of each fan to prevent back flow through the standby unit.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	3	
Change: 2	Category	3	
	K/A	G 2.3.12	
Level of Difficulty: 3	Importance Rating	3.2	

Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: 73

Given the following conditions:

- Radiography is in progress in the Unit 2 Safeguards Building.
- When the source is exposed the dose rates in the area are rising to 1440 mrem/hr.
- The radiography is being performed in an open area of the Safeguards Building with NO enclosure.

Which of the following is the type of radiological area and what constitutes the minimum radiation monitoring required for an operator to enter the area?

A. Very High Radiation Area (VHRA).
Electronic Dosimeter.

B. Locked High Radiation Area (LHRA).
Electronic Dosimeter.

C. Very High Radiation Area (VHRA).
Alarming Electronic Dosimeter.

D. Locked High Radiation Area (LHRA).
Alarming Electronic Dosimeter.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that greater than 1000mr/hr is VHRA and that an electronic dosimeter is all that is required.
B. Incorrect. Plausible as the area meets requirements for a LHRA, no enclosure requires use of flashing lights, and DANGER signs and if thought that an electronic dosimeter may is all that is required.
C. Incorrect. Plausible if thought that greater than 1000mr/hr is VHRA, RAD key lock cannot be used as there is no enclosure, GRAVE DANGER is appropriate for VHRA but area is LHRA due to dose rate and alarming electronic dosimeter is the minimum required.
D. Correct. The area meets requirements for posting as a LHRA, no enclosure requires use of flashing lights, and DANGER signs. The alarming electronic dosimeter is the minimum required for entry.

Technical Reference(s): STA-660, Section 4.4; STA-660, Step 6.1.4 & 6.2.5; STA-660, Attachment 1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source:

Bank	NRC 2013 71	
Modified Bank		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2013

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	12
55.43	

Comments / Reference: STA-660, Section 4.4	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15 INFORMATION USE	Page 3 of 11

4.2 Dose Margin - The remaining allowable total effective dose equivalent an individual may receive during a specified monitoring period.

4.3 High Radiation Area (HRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 100 mrem in 1 hour at 30 centimeters (approximately 1 foot) from the radiation source or from any surface that the radiation penetrates.

4.4 Locked High Radiation Area (LHRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1000 milli-rem in 1 hour at 30 centimeters (approximately 1 foot) from the radiation source or from any surface that the radiation penetrates.

4.5 RAD Key – A mechanical key that provides access to Locked High Radiation Areas by the ability to open an associated RAD Lock.

4.6 RAD Lock – A lock used exclusively for controlling access to Locked High Radiation Areas.

4.7 Radiologically Significant ALARA Briefing – A documented briefing between Radiation Protection and participants prior to the commencement of work activities where radiological conditions are subject to frequent or rapid change. This briefing shall be performed prior to entry into a posted LHRA or VHRA.
[TS 5.7]

4.8 Electronic Dosimeter – A radiation monitoring device which continuously integrates the radiation dose rate and alarms when a preset integrated dose or dose rate is received.

4.9 Expected Dose – The dose that is expected for the duration of an entry into an area. The expected dose may be a dose calculated during job planning for all persons entering the area (e.g., Steam Generator channel head entries may have a dose setting of 750 mrem for the planned activities).

4.10 Stay Time – The length of time an individual may be allowed into an area based on radiation levels and remaining dose margin or expected dose, whichever is the most limiting.

4.11 Very High Radiation Area (VHRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in 1 hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

Comments / Reference: STA-660, Step 6.1.4	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15 INFORMATION USE	Page 5 of 11

6.1.2.1 For injury responses outside of a declared emergency, this requirement is met as long as Radiation Protection provides an escort into the affected area.
[CR-2009-001502]

6.1.3 Prior to entering a High Radiation Area, personnel shall receive an Area ALARA or Radiologically Significant ALARA briefing.
[TS 5.7]

6.1.3.1 For injury responses outside of a declared emergency, this requirement is met as long as Radiation Protection provides an escort into the affected area and emergency response personnel are advised of the dose rates in the area prior to entry.
[CR-2009-001502]

6.1.4 All entries shall be provided with or accompanied by one or more of the following:

- A radiation monitoring device which continuously indicates the radiation dose rate in the area, OR
- Electronic Dosimeter – Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, OR
- An individual qualified in radiation protection procedures equipped with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and should perform periodic radiation surveillance at the frequency specified by the RWP.

Comments / Reference: STA-660, Step 6.2.5	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15 INFORMATION USE	Page 6 of 11

[C] 6.2 Locked High Radiation Areas
[27374][10CFR20][REG GUIDE 8.38]

NOTE: RAD locks are used for controlling access to Locked High Radiation Areas only. Areas in the containment buildings which are normally Locked High Radiation Areas may have the locks left in place during a shutdown if changing the lock core is impractical (e.g., plant trip with expected shutdown limited and access to the locked areas is minimal).

6.2.1 In addition to the High Radiation Area requirements of Section 6.1 of this procedure, entrances to Locked High Radiation Areas shall be locked. Radiation Protection shall maintain positive control over each entry into a LHRA. 1

6.2.1.1 When a LHRA is unlocked, the boundary shall be guarded to prevent unauthorized entry. The individual assigned as a guard shall maintain constant surveillance and have the ability to warn others prior to them entering the posted area.

6.2.2 If a Locked High Radiation Area has no enclosure which can be locked and no enclosure can be reasonably constructed around it, then the area shall be barricaded and conspicuously posted with two flashing lights which are activated as a warning device.

6.2.3 When it is determined that an area is a Locked High Radiation Area, Radiation Protection should notify the Shift Manager.

6.2.4 Upon becoming a Locked High Radiation Area, Radiation Protection should replace the existing lock with a RAD lock. The status of the area should be verified in accordance with Attachment 1.
[ONE-98-001299]

[C] 6.2.5 Continuous Radiation Protection coverage shall be provided at all times when personnel are in the area. Alarming electronic dosimetry or telemetry may be used in lieu of continuous RP coverage if the dose rates are known and allowed for in the RWP.
[02302]

Comments / Reference: STA-660, Attachment 1	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15 INFORMATION USE	Page 11 of 11

ATTACHMENT 1
PAGE 1 of 1

LOCKED HIGH RADIATION AREA VERIFICATION

1. Radiation survey performed and documented in accordance with plant procedures.
2. Ensure the lock core has been changed appropriately ('RAD' or 'Very High RAD').
3. Status of area or room lock has been verified.
   [CR-2001-002958]
   a. Unlocking an area
      i. Ensure area is correct
      ii. Unlock the lock
      iii. Record on applicable Rad Key Tag
      iv. Perform a concurrent verification that the lock is unlocked
      v. Record on applicable Rad Key Tag
   b. Locking an area
      i. Ensure area is correct
      ii. Lock the lock
      iii. Ensure lock is securely locked
      iv. Perform a concurrent verification that the lock is locked
      v. Record on applicable Rad Key Tag
4. When area cannot be locked, ensure the area is barricaded, and a flashing light is located in close proximity to the area.
5. Ensure all entrances have been posted. In lieu of "CAUTION" signs, use "DANGER" sign for Locked High Radiation Areas and "GRAVE DANGER" sign for Very High Radiation Areas.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/6/2014	Tier	3	
Change: 3	Category	4	
	K/A	G 2.4.6	
Level of Difficulty: 2	Importance Rating	3.7	

Emergency Procedures/Plan: Knowledge of EOP mitigation strategies

Proposed Question: 74

Given the following conditions:

- Unit 1 is responding to a ruptured Steam Generator in accordance with EOP-3.0A, Steam Generator Tube Rupture.
- The ruptured Steam Generator has been identified and isolated.
- The Reactor Coolant System cooldown to target Core Exit Thermocouple temperature has been completed and required subcooling has been established.

Which of the following describes the next major action categories to be accomplished in EOP-3.0A, Steam Generator Tube Rupture to mitigate the tube rupture?

A. Terminate Safety Injection, and then Prepare to Cooldown to Cold Shutdown.

B. Terminate Safety Injection, and then Depressurize the Reactor Coolant System.

C. Depressurize the Reactor Coolant System, and then Terminate Safety Injection.

D. Depressurize the Reactor Coolant System, and then Prepare to Cooldown to Cold Shutdown.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because these are the two major action categories, in order, that are performed following the depressurization of the RCS. However, Depressurization of the RCS must be performed prior to attempting to Terminate Safety Injection.
B. Incorrect. Plausible because these are the two major action categories that are performed following the completion of the RCS cooldown. However, they are listed in reverse order.
C. Correct. IAW EOP-3.0A, depressurizing the RCS and terminating Safety Injection are the next two major action categories to be performed following completion of the RCS Cooldown.
D. Incorrect. Plausible because Depressurize the RCS is the next major action category which must be performed. However, the preparation for the Cooldown to Cold Shutdown cannot be completed until Safety injection termination has taken place.

Technical Reference(s): EOP-3.0A, Flowchart

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the analysis of a SGTR including the operator expected actions and associated times.

Question Source:

Bank: ______

Modified Bank: ______ (Note changes or attach parent)

New: X

Question History: Last NRC Exam ______

Question Cognitive Level:

Memory or Fundamental Knowledge: ______

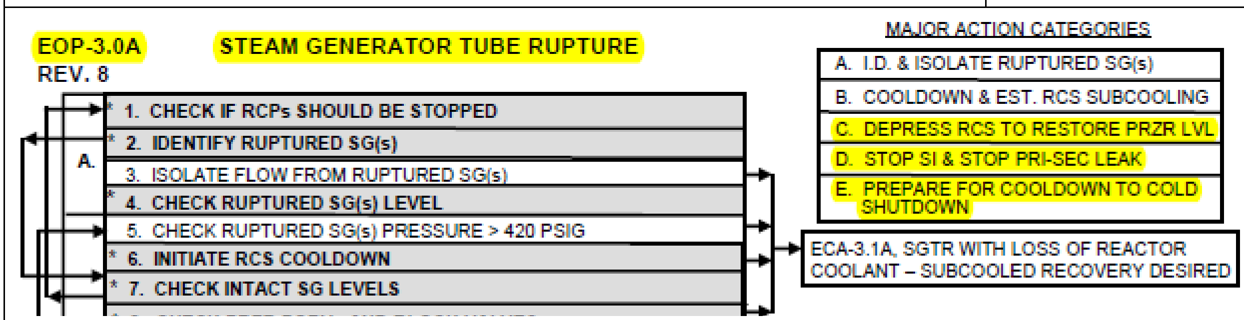
Comprehension or Analysis: X

10 CFR Part 55 Content:

55.41: 10

55.43: ______

Comments / Reference: EOP-3.0A, Flowchart	Revision: 8



Comments / Reference: EOP-3.0A, Attachment 6, Step 19 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 74 OF 103

ATTACHMENT 6
PAGE 21 OF 50

BASES

STEP 18: The RCS cooldown is designed to establish a 20°F subcooling margin, i.e., 20°F greater than uncertainties, in the primary system at the ruptured steam generator pressure. For steam generator tube rupture events, including multiple tube failures, with ECCS in service the primary pressure will stabilize at a value greater than the ruptured steam generator pressure. Consequently, at this stage of the recovery, the subcooling margin is expected to be greater than 20°F. If not, a loss of reactor coolant is suspected. In that case, the operator is directed to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, to stop ECCS pumps one at a time after it is demonstrated that the reduced ECCS flow is sufficient to maintain adequate coolant inventory.

For multiple tube failures, RCS pressure may temporarily decrease below the ruptured steam generator pressure during cooldown. However, pressure and subcooling should quickly increase when the cooldown is complete. The transition to ECA-3.1A is not necessary if subcooling increases sufficiently after the cooldown is complete.

STEP 19: After the cooldown is completed, ECCS flow will pressurize the RCS to an equilibrium condition where break flow equals ECCS flow. This equilibrium pressure will be somewhere between the ruptured steam generator pressure and the shutoff head of the ECCS pumps and increases with ECCS capacity. A major objective of EOP-3.0A is to bring the plant to a point where primary-to-secondary leakage will be stopped. However, the path one takes is important. The ideal path should increase coolant inventory and restore pressurizer level. Hence, the ideal path requires a depressurization of the RCS.

In some cases, pressurizer level may approach the upper tap (top of the indicating range) before RCS pressure is reduced to the ruptured steam generator pressure. This may be a symptom of a smaller tube failure, voiding in the upper head during natural circulation conditions, or injection of the SI accumulators. Depressurization of the RCS is terminated on high pressurizer level to prevent filling the pressurizer and loss of pressurizer pressure control. Following SI termination, pressurizer level decreases which further reduces RCS pressure to equilibrium with the ruptured steam generator. In some cases, such as a small tube failure in a high pressure SI plant, the pressurizer may be sufficiently full such that no depressurization of the RCS can be performed prior to ECCS termination.

On the other hand, for multiple tube failures or reduced ECCS capacity for a smaller tube failure, it may be necessary to decrease RCS pressure below that of the ruptured steam generator pressure in order to restore pressurizer level. In that case reverse flow, i.e., secondary-to-primary leakage, will supplement ECCS flow to restore

Comments / Reference: EOP-3.0A, Attachment 6, Step 19 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 75 OF 103

ATTACHMENT 6
PAGE 22 OF 50

BASES

pressurizer level. If pressure continued to be reduced to saturation, voiding in the primary system may result in an unreliable pressurizer level indication and delay SI termination. To avoid this, depressurization of the RCS is terminated if minimum RCS subcooling is reached.

Comments / Reference: EOP-3.0A Bases, Step 23	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 81 OF 103

ATTACHMENT 6
PAGE 28 OF 50

BASES

STEP 23: When the conditions for terminating ECCS flow are satisfied, ECCS flow must be terminated to stop primary-to-secondary leakage. This is done by stopping all ECCS pumps not needed for normal reactor coolant makeup. These pumps are placed in standby to ensure their availability in the event that ECCS flow must be reinitiated.

Primary-to-secondary leakage will continue with one CCP running until normal charging and letdown are established. Consequently, this and subsequent steps should be completed as quickly as possible to prevent steam generator overfill.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	3	
Change: 2	Category	4	
	K/A	G 2.4.32	
Level of Difficulty: 2	Importance Rating	3.6	

Emergency Procedures/Plan: Knowledge of operator response to loss of all annunciators

Proposed Question: 75

Given the following conditions:

- Unit 2 is responding to a loss of all Control Room Annunciators in accordance with ABN-740B, Control Room Annunciator System and Status Light Malfunction.
- Unit load is stable at 100%.
- All work on Unit 2 has been stopped.

Which of the following surveillances must be initiated while the annunciators are out of service?

OPT-303, Reactor Coolant System Water Inventory, ...

A. ...OPT-112B, Accident Monitoring Instrumentation Check, and OPT-302, Calculating Power Tilt Ratio.

B. ...OPT-112B, Accident Monitoring Instrumentation Check, and OPT-309, Unit Calorimetric.

C. ...OPT-309, Unit Calorimetric, and OPT-403, Axial Flux Difference.

D. ...OPT-302, Calculating Power Tilt Ratio, and OPT-403, Axial Flux Difference.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because OPT-302, QPTR is required, however OPT-112B, Accident Monitoring Instrumentation Checks is not required. It could be thought that OPT-112B performance is required because it is required by ABN-906, Plant Process Computer System Malfunction.

B. Incorrect. Plausible because OPT-112B and OPT-309 performance is required but it is required by ABN-906, Plant Process Computer System Malfunction.

C. Incorrect. Plausible because OPT-403, AFD is required, however OPT-309, Unit Calorimetric is not required. It could be thought that OPT-309 performance is required because it is required by ABN-906, Plant Process Computer System Malfunction.

D. Correct. IAW ABN-740B, Attachment 1, OPT-303, OPT-302 and OPT-403 are required during a loss of all annunciators.

Technical Reference(s) ABN-740B, Attachment 1 Attached w/ Revision: See Comments / Reference
ABN-906, Steps 2.3.2 RNO & 2.3.3 RNO

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-740B, Attachment 1	Revision: 1

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. ABN-740B
CONTROL ROOM ANNUNCIATOR SYSTEM AND STATUS LIGHT MALFUNCTION	REVISION NO. 1	PAGE 24 OF 60

ATTACHMENT 1
PAGE 1 OF 1

ANNUNCIATOR TECHNICAL SPECIFICATION CROSS REFERENCE

ALARM WINDOW(S)	TS/TR	SURVEILLANCE	FREQUENCY
2-ALB-2A 1.6,1.7,1.8,2.6,2.7,2.8	3.4.15, RCS Leakage Detection Systems	OPT-303	Each 24 hrs
2-ALB-2B 1.12,2.12,3.12,4.12	3.4.15, RCS Leakage Detection Systems	OPT-303	Each 24 hrs
2-ALB-6D			
4.10	TRS 13.2.33.1, Quadrant Power Tilt Ratio (QPTR) Alarm	OPT-302	Each 12 hrs during steady state operation when >50 RTP
4.11	TRS 13.2.32.1, Axial Flux Difference (AFD)	OPT-403	Once within 30 minutes AND 1 hour thereafter

Comments / Reference: ABN-906, Step 2.3.2 RNO	Revision: 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-906
PLANT PROCESS COMPUTER SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 4 OF 22

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- Upon Manual Plant Computer System (PCS) Failover, affected PCS screen(s) will turn red in color and the word FAILOVER or SAIPMS DOWN will appear until backup CPU is on-line (PCS functions disabled for approximately 45 seconds).
- During time Plant Computer is not operating, increased operator vigilance is required due to loss of many important monitoring functions performed by computer (e.g., alarms, thermal power calculations). Attachment 5 provides information for loss of various multiplexors.
- With restoration of the LEFM Calorimetric Program from a failed condition OR reboot of the Plant Computer from a failed condition, all "POWER LEFM" VALUES (1M, 15M, 30M, 1H, and 8H) on the POWERL Screen will be updated to the instantaneous value calculated by the LEFM Calorimetric Program. The resulting indication will remain as shown for the duration of the applicable time interval, which will not accurately reflect changes in the actual thermal power for the same duration. ODA-308-13.3.34-S01 provides guidance for the restoration of the LEFM.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify ALL Plant Computer CRT screens displaying Red background with word FAILOVER or SAIPMS DOWN.	Perform the following as applicable: a. IF at least ONE CRT is updating accurately, THEN use it to ensure operating parameters remain normal. b. Notify I&C Computer Group of existing problems.
☐ 2 Verify within TWO minutes of event initiation Control Room Satellite Display stations (SDS) - AT LEAST ONE PERFORMING INTENDED FUNCTIONS • Plant Computer CRT screens OPERATING AND • Plant computer Data - UPDATING	Perform the following: a. Notify I&C Computer group. b. Refer to Technical Requirements for limiting conditions of operation: • 13.2.32.1, AND begin logging AFD per OPT-403. • 13.2.33.1, within 12 hours, begin calculating QPTR per OPT-302. • 5.5.15-1, loss of Plant Computer Technical Specification Monitoring. • 13.3.34, Plant Calorimetric Measurement

Comments / Reference: ABN-906, Step 2.3.3 RNO	Revision: 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-906
PLANT PROCESS COMPUTER SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 5 OF 22

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	c. IF required trending was in progress, THEN ensure alternate method of documentation is initiated.

NOTE: During time Plant Process Computer is not operating, increased operator vigilance is required due to loss of many important monitoring functions performed by computer (e.g., alarms, thermal power calculations).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify Satellite Display Station(s) - UPDATING AS PROGRAMMED. • SPDS clock AND Main Control Board clock less than ONE minute apart. • Programmed data - UPDATING	Perform the following: a. IF Reactor Trip occurs, AND NO Satellite Display Station available, THEN manually determine critical safety functions as required by Emergency Operating Procedure in effect. b. IF any Satellite Display Station available, THEN monitor Critical Safety Functions on available Satellite Display Station. c. Notify I&C group of existing problems. d. Refer to Technical Specification 3.3.3 AND initiate standard LCOAR, as necessary. e. Perform applicable portions of OPT-112A/B as determined by Unit Supervisor.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		1
Change: 2	Group		1
	K/A	009 EA2.02	
Level of Difficulty: 3	Importance Rating		3.8

Small Break LOCA: Ability to determine and interpret the following as they apply to the small break LOCA: Possible leak paths

Proposed Question: 76

Given the following conditions:

- A Small Break Loss of Coolant Accident (LOCA) is in progress on Unit 2.
- A cooldown is in progress in accordance with EOS-1.2B, Post LOCA Cooldown and Depressurization.
- Containment pressure is 3.5 psig and slowly rising.
- One Control Rod failed to insert on the Reactor Trip.
- Reactor Coolant Pumps are stopped.
- Reactor Coolant System (RCS) pressure is 1585 psig and slowly lowering.
- Subcooled Margin is 10°F and becoming less subcooled.
- Pressurizer level is 94% and rising.
- The RVLIS 49" above the flange light is DARK and all other RVLIS lights are LIT.
- The Shift Technical Advisor reports the Inventory Critical Safety Function is YELLOW.

Which of the following describes the location of the Reactor Coolant System leak and the expected mitigation actions?

A. Pressurizer Steam Space.
Remain in EOS-1.2B, Post LOCA Cooldown and Depressurization and continue cooldown to less than 200°F.

B. Reactor Vessel Upper Head.
Remain in EOS-1.2B, Post LOCA Cooldown and Depressurization and continue cooldown to less than 200°F.

C. Pressurizer Steam Space.
Transition to FRI-0.3B, Response to Voids in Reactor Vessel, and prepare to start a Reactor Coolant Pump.

D. Reactor Vessel Upper Head.
Transition to FRI-0.3B, Response to Voids in Reactor Vessel, and prepare to start a Reactor Coolant Pump.

Proposed Answer: A

Explanation:

A. Correct. The rising Pressurizer level with voiding in the vessel head indicates a Pressurizer steam space leak exists and the correct action is to continue the cooldown to less than 200°F.
B. Incorrect. Plausible because the upper RVLIS light being DARK and a Control Rod failure to insert could be indication of a head leak, however, the Pressurizer level would not be rising. Remaining in EOS-1.2B and cooling down to less than 200°F is the correct action.
C. Incorrect. Plausible because the leak is in the Pressurizer steam space but transition to FRI-0.3B is not the correct action. The YELLOW path exists due to voids and Pressurizer level which could be thought the correct action to take, however, cooldown to less than 200°F is the correct action.
D. Incorrect. Plausible because the upper RVLIS light being DARK and a Control Rod failure to insert could be indication of a head leak however the Pressurizer level would not be rising. The YELLOW path exists due to voids and Pressurizer level which could be thought the correct action to take, however, cooldown to less than 200°F is the correct action.

Technical Reference(s): EOS-1.2B, Flowchart; FRI-0.3B, CSFST; FRI-0.3B, Step 1 CAUTION — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOS-1.2B, Post LOCA Cooldown and Depressurization.

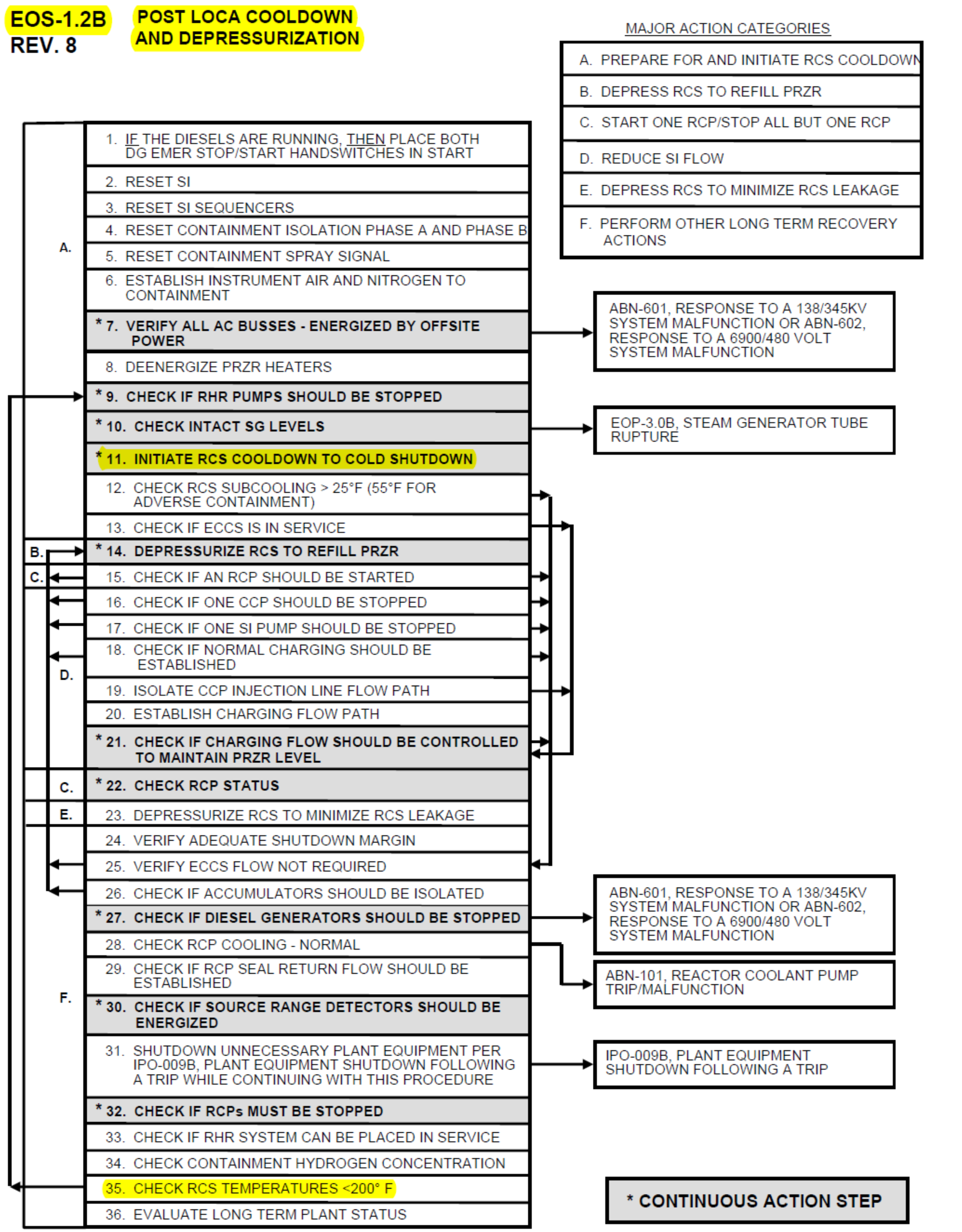
Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

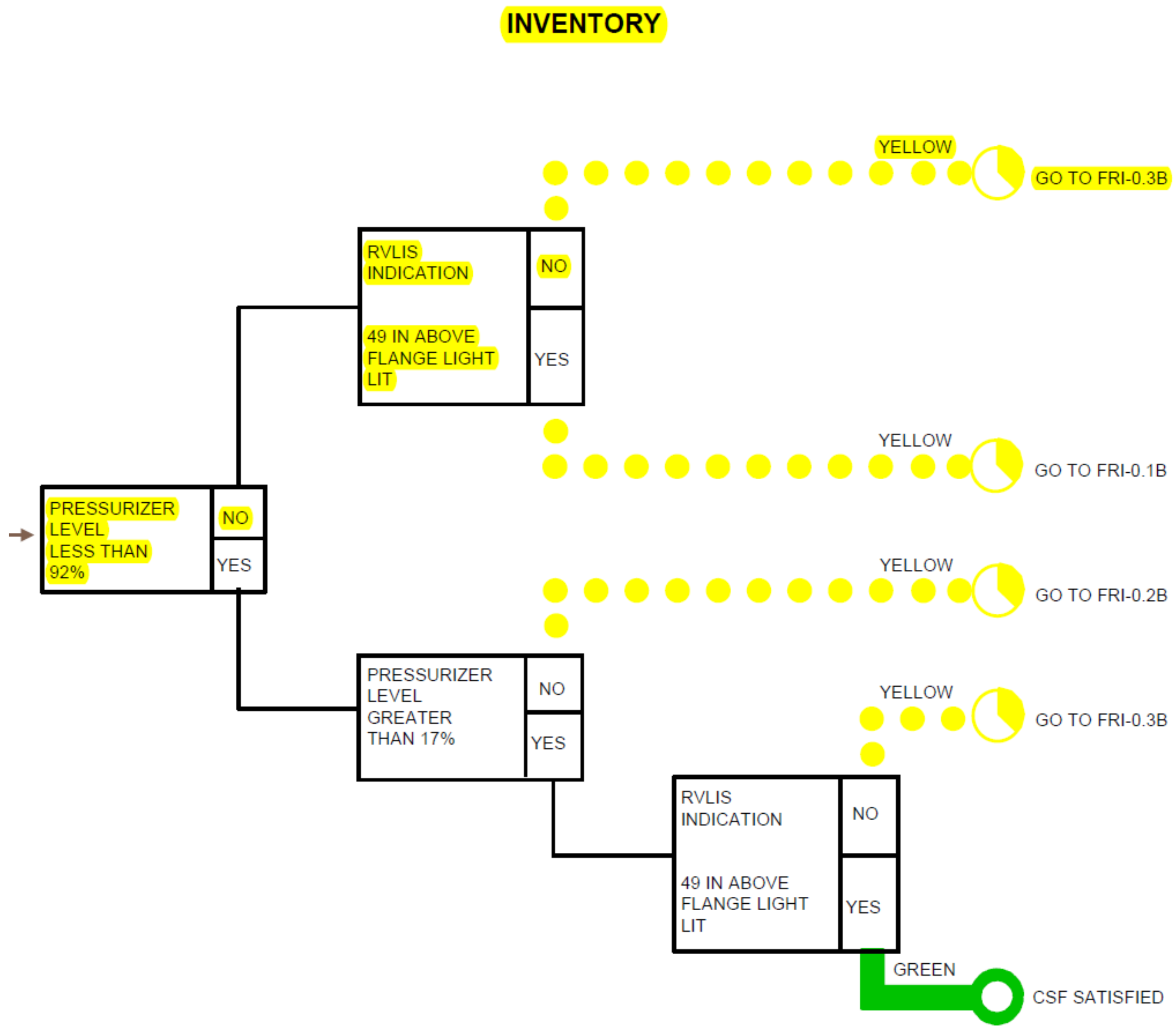
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____
55.43 5

Comments / Reference: EOS-1.2B, Flowchart	Revision: 8



Comments / Reference: FRI-0.3B, CSFST	Revision: 8



Comments / Reference: FRI-0.3B, Step 1 CAUTION	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRI-0.3B
RESPONSE TO VOIDS IN REACTOR VESSEL	REVISION NO. 8	PAGE 3 OF 44

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: If a controlled natural circulation cooldown is in progress and a void in the reactor vessel upper head is expected, this procedure should not be performed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check If ECCS Has Been Terminated: • SI pumps - ALL STOPPED • CCP injection line - ISOLATED	Return to procedure and step in effect.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		1
Change: 2	Group		1
	K/A	E04 EA2.01	
Level of Difficulty: 2	Importance Rating		4.3

LOCA Outside Containment: Ability to determine and interpret the following as they apply to the (LOCA Outside Containment): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: 77

Given the following conditions:

- A Unit 1 Reactor Trip and Safety Injection have occurred.
- The Safeguards Building area radiation monitors are in RED alarm.
- All Containment Building parameters are normal.
- ECA-1.2A, LOCA Outside Containment is in progress.
- After closing 1/1-8835, Safety Injection to Cold Leg 1 • 4 Injection Isolation Valve, Reactor Coolant System pressure is 1850 psig and rising with Emergency Core Cooling System flow lowering.

Which of the following describes the status of the Loss of Coolant Accident and the required procedure transition?

A. The LOCA is isolated.
Transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

B. The LOCA is isolated.
Transition to ECA-1.1A, Loss of Emergency Coolant Recirculation.

C. The LOCA is NOT isolated.
Transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

D. The LOCA is NOT isolated.
Transition to ECA-1.1A, Loss of Emergency Coolant Recirculation.

Proposed Answer: A

Explanation:

A. Correct. When 1/1-8835 is closed in Step 2 of ECA-1.2A and RCS pressure rises, this leads to a transition to EOP-1.0A at Step 3 of ECA-1.2A.

B. Incorrect. Plausible because the LOCA is isolated, however, a transition to ECA-1.1A is only required if RCS pressure is lowering (ECA-1.2A, Step 3 RNO).

C. Incorrect. Plausible because transition to EOP-1.0A is required, however, RCS pressure is rising so the LOCA is isolated.

D. Incorrect. Plausible if thought that transition to ECA-1.1A is required even when RCS pressure is rising due to the lack of inventory in the Containment.

Technical Reference(s) ECA-1.2A, Steps 2 & 3 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the proper transitions out of ECA-1.2, LOCA Outside Containment.

Question Source: Bank ILOT5963
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: ECA-1.2A, Step 2	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 3 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 1	Verify Proper Valve Alignment: a. RHRP 1 & 2 HL RECIRC ISOL VLVS - CLOSED • 1/1-8701A • 1/1-8702A • 1/1-8701B • 1/1-8702B b. RHR TO HL 2 & 3 INJ ISOL VLV - CLOSED • 1/1-8840 c. SI TO HL INJ ISOL VLVS - CLOSED • 1/1-8802A • 1/1-8802B	Manually close valve(s). <u>IF</u> valve(s) can <u>NOT</u> be manually closed, <u>THEN</u> locally close valve(s).
2	Identify And Isolate Break: a. Sequentially close and open the following valves and monitor for an RCS pressure increase: 1) RHR TO CL INJ ISOL VLVS: • 1/1-8809A • 1/1-8809B 2) SI to CL 1•4 INJ ISOL VLV • 1/1-8835	

Comments / Reference: ECA-1.2A, Step 3	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 4 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Check If Break Is Isolated:	
	a. RCS pressure - INCREASING	a. Go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	

-END-

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/24/2014	Tier		1
Change: 4	Group		1
	K/A	038 EA2.09	
Level of Difficulty: 4	Importance Rating		4.2

Steam Generator Tube Rupture: Ability to determine and interpret the following as they apply to a SGTR: Existence of natural circulation, using plant parameters

Proposed Question: 78

Given the following conditions:

- Unit 1 is responding to a Steam Generator Tube Rupture on Steam Generator 1-01 in accordance with EOS-3.1A, Post-SGTR Cooldown Using Backfill.
- All Reactor Coolant Pumps (RCP) were tripped on a loss of subcooling while performing EOP-0.0A, Reactor Trip or Safety Injection.
- EOP-0.0A, Reactor Trip or Safety Injection, Attachment 9, Post Event System Realignment is completed with systems reset or restored.
- The following indications exist:
  - Steam Generator 1-01 pressure is 785 psig and stable.
  - Steam Generator 1-01 level is 70% and stable.
  - Steam Generators 1-02, 1-03, & 1-04 pressures are 435 psig and lowering.
  - Steam Generators 1-02, 1-03 & 1-04 are ≈ 55% and stable.
  - Loop 1 Reactor Coolant Cold Leg temperature is 475°F and lowering.
  - Loops 2, 3, and 4 Reactor Coolant Cold Leg temperatures are 456°F and stable.
  - Loop 1 Reactor Coolant Hot Leg temperature is 490°F and lowering.
  - Loops 2, 3, and 4 Reactor Coolant Hot Leg temperatures are 476°F and lowering.
  - Reactor Coolant System pressure is 885 psig and lowering.
  - Highest reading Core Exit Thermocouple is 480°F and lowering.

Which of the following states the current natural circulation status and the action to be taken in accordance with EOS-3.1A, Post-SGTR Cooldown Using Backfill?

A. Natural Circulation exists.
Start a Reactor Coolant Pump.

B. Natural Circulation exists.
Do NOT start a Reactor Coolant Pump.

C. Natural Circulation does NOT exist.
Start a Reactor Coolant Pump.

D. Natural Circulation does NOT exist.
Do NOT start a Reactor Coolant Pump.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because IAW EOS-3.1A, Attachment 3, all indications support Natural Circulation flow. However, EOS-3.1A requires 60°F of subcooling in order to restart an RCP and only 42°F subcooling exists.
B. Correct. IAW EOS-3.1A, Attachment 3, all indications support natural circulation flow and subcooling is 42°F which is not sufficient for RCP restart.
C. Incorrect. Plausible if incorrectly included Loop 1 values in the determination of Natural Circulation. Additionally, it could be thought that the subcooling to determine existence of natural circulation also satisfies RCP restart criteria.
D. Incorrect. Plausible if incorrectly included Loop 1 values in the determination of Natural Circulation. However, conditions do not support RCP restart.

Technical Reference(s): EOS-3.1A, Attachment 3; EOS-3.1A, Steps 1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in EOP-3.0A, Steam Generator Tube Rupture.

Question Source:
- Bank ____
- Modified Bank ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 5

Comments / Reference: EOS-3.1A, Attachment 3	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-3.1A
POST-SGTR COOLDOWN USING BACKFILL	REVISION NO. 8	PAGE 18 OF 38

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- ☐ RCS subcooling - GREATER THAN 25°F (55°F FOR ADVERSE CONTAINMENT).
- ☐ SG pressures - STABLE OR DECREASING.
- ☐ RCS hot leg temperatures - STABLE OR DECREASING.
- ☐ Core exit TCs - STABLE OR DECREASING.
- ☐ RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE.

Comments / Reference: EOS-3.1A Step 1	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-3.1A
POST-SGTR COOLDOWN USING BACKFILL	REVISION NO. 8	PAGE 3 OF 38

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: Inadvertent criticality may occur following any natural circulation cooldown if the first RCP started is in the ruptured loop.

CAUTION: If RCP seal cooling had previously been lost, that affected RCP should not be started prior to a status evaluation.

NOTE: RCPs should be run in order of priority to provide normal PRZR spray (RCP 4, 1 then 2 or 3).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1	Check RCP Status:	
	a. RCP 4 - RUNNING	a. Start RCP(s) to provide normal PRZR spray: 1) Establish conditions for starting RCP(s) per Attachment 2. 2) IF RVLIS 49 IN above Flange Light NOT LIT, THEN perform the following: • Increase PRZR level greater than 90% (98% FOR ADVERSE CONTAINMENT). • Increase RCS subcooling greater than 60°F (95°F FOR ADVERSE CONTAINMENT). • Use PRZR heaters as necessary to saturate the pressurizer water.

Comments / Reference: EOS-3.1A, Step 1	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-3.1A
POST-SGTR COOLDOWN USING BACKFILL	REVISION NO. 8	PAGE 4 OF 38

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		3) Start RCP 4 per Attachment 2. IF RCP 4 can NOT be started, THEN start other RCP(s) per Attachment 2 as necessary to provide normal spray.
		IF RCP(s) can NOT be started, THEN refer to Attachment 3 to verify natural circulation.
		IF natural circulation NOT verified, THEN increase dumping steam.
* 2	Cycle PRZR Heaters As Necessary To Saturate PRZR Water At Ruptured SG(s) Pressure.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		1
Change: 3	Group		1
	K/A	040 G 2.1.7	
Level of Difficulty: 4	Importance Rating		4.7

Steam Line Rupture: Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Question: 79

Given the following conditions:

- A seismic event has occurred at Comanche Peak.
- Unit 1 has experienced a Reactor Trip and Safety Injection.
- A complete Main Steamline Isolation was verified in EOP-0.0A, Reactor Trip or Safety Injection.
- Transition has been made to EOP-2.0A, Faulted Steam Generator Isolation.
- The following parameters are observed:
  - Containment pressure is 15 psig and rising.
  - Reactor Coolant System (RCS) average temperature is 475°F and lowering.
  - RCS pressure is 1620 psig and lowering.
  - Pressurizer level is 32% and lowering.
  - Steam Generator (SG) 1-01 pressure is ≈ 15 psig.
  - SG 1-02, 1-03 & 1-04 pressures are ≈ 540 psig and lowering.
  - SG 1-01 has been isolated.
  - AFW flow to SGs 1-02, 1-03 and 1-04 is 180 gpm per SG.
  - SG 1-01 level is 20% wide range and rising.
  - SG 1-02, 1-03 & 1-04 levels are ≈ 55% narrow range and stable.

Which of the following is the first procedure transition required in accordance with EOP-2.0A, Faulted Steam Generator Isolation?

Transition to...

A. ...ECA-2.1A, Uncontrolled Depressurization of All Steam Generators.

B. ...EOP-3.0A, Steam Generator Tube Rupture.

C. ...EOS-1.1A, Safety Injection Termination.

D. ...EOP-1.0A, Loss of Reactor or Secondary Coolant.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the plant indications show pressure lowering in all intact steam generators, however the pressure is not decreasing in an uncontrolled manner based on the intact steam generators all being at the same pressure and at an expected pressure following blowdown of a single steam generator.

B. Correct. The rising level in the faulted steam generator and the lowering pressurizer level, RCS pressure and RCS temperature indicate that the faulted steam generator is also ruptured.

C. Incorrect. Plausible because subcooling is 100°F and a secondary heat sink exists however with RCS pressure lowering and pressurizer level less than 34% SI cannot be terminated.

D. Incorrect. Plausible because there would be no radiation monitor feedback for the faulted ruptured steam generator in this condition and the operator would have to diagnose that the faulted steam generator is not attaining dryout condition to avoid bypassing the transition to EOP-3.0A.

Technical Reference(s): EOP-2.0A, Steps 8 & 9; EOP-2.0A, Attachment 3, Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the symptoms for the entry conditions of EOP-2.0, Faulted Steam Generator Isolation.

Question Source:

Bank		
Modified Bank		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	
55.43	5

Comments / Reference: EOP-2.0A, Step 7 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 13 OF 14

ATTACHMENT 3
PAGE 5 OF 6

BASES

Since it may be difficult to sample a depressurized steam generator for activity, the operator should suspect a rupture if the steam generator does not dry out following isolation of feed to it. A faulted, ruptured steam generator will stay at some low pressure and continue to cool that loop and the RCS. In addition, the operator should suspect a rupture if following SG dryout, RCS inventory or pressure cannot be maintained and there is no indication of an RCS leak to containment. If the operator suspects that a faulted steam generator is not drying out and cannot confirm that it is ruptured by sampling because a sample cannot be drawn, the operator may request a check for radiation in the area of the break (if it is outside the containment) to confirm that a rupture exists. If it is not practical to locally check for radiation, the operator may conclude that the faulted generator is ruptured based on response of the faulted steam generator or the response of the RCS.

Optimal recovery in dealing with a steam generator tube rupture is provided in EOP-3.0A, STEAM GENERATOR TUBE RUPTURE.

Comments / Reference: EOP-2.0A, Step 2 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 9 OF 14

ATTACHMENT 3
PAGE 1 OF 6

BASES

CAUTION: During the attempt to determine the faulted loop(s), the operator must maintain at least one loop available for cooldown capability. Otherwise, RCS pressure and temperature will increase if all SGs are isolated.

CAUTION: During isolation of a faulted SG or secondary break, normal cooldown paths are likely to be isolated. Cooldown procedures may direct the operator to open one of the isolated cooldown paths resulting in a reinitiation of the event. However, in cases where an isolated SG is the only SG available, this SG can be unisolated and used for RCS cooldown.

STEP 1: Since this procedure is entered in response to a diagnosed SG fault, the main steamline isolation valves should have previously received a "CLOSE" signal. If not, or if the valves failed to close, the operator is instructed to manually close the valves. The main steam isolation bypass valves are sealed closed with position indication on the control board. All steamlines receive the isolation signal and, therefore, all SGs MSIVs are checked closed. The MSIVs are checked to be closed in this step in an attempt to isolate the break and to isolate the SGs from each other.

The action to verify Main Steam isolation valve position is intended to include the actions to verify the Main Steam isolation bypass valves closed, in the event the bypass valves have been opened during startup operation (e.g., Main Steamline warmup).

STEP 2: Any cooldown operations that are performed as subsequent recovery actions will require at least one nonfaulted SG. If all SG pressures are decreasing in an uncontrolled manner, this indicates a failure affecting all SGs. Recovery actions, in this case, should be performed using ECA-2.1A, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, since feedwater flow will be necessary to a faulted SG and normal level control should not be used.

"Uncontrolled" means not under the control of the operator and incapable of being controlled by the operator using available equipment. The intent of this step is not to identify a Faulted Steam Generator based on a decreasing pressure due to an RCS cooldown (or other known cause) even though it may not be under the control of the operator. If the rate at which pressure is

Comments / Reference: EOP-2.0A, Step 2 Bases continued	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 10 OF 14

ATTACHMENT 3
PAGE 2 OF 6

BASES

decreasing is small or the cause is known, it should not be considered DECREASING IN AN UNCONTROLLED MANNER. Non-faulted SG pressures may decrease slightly due to RCS cooldown from a faulted SG. This non-faulted SG pressure response can be controlled by the operator when the faulted SG depressurizes.

Comments / Reference: EOP-2.0A, Step 8	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 5 OF 14

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	Check Secondary Radiation:	
	a. Request periodic activity samples of all SGs.	
	b. Check available secondary radiation monitors - NORMAL • Condenser Off Gas radiation (COG-182, 1RE-2959) • Main steamline radiation (MSL-178 through 181, 1RE-2325 through 2328) • SG blowdown sample radiation (SGS-164, 1RE-4200)	b. Go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.
8	Check If ECCS Flow Should Be Reduced:	
	a. Secondary heat sink: • Total AFW flow to intact SGs - GREATER THAN 460 GPM -OR- • Narrow range level in at least one intact SG - GREATER THAN 43%(50% FOR ADVERSE CONTAINMENT)	a. <u>IF</u> neither condition satisfied, <u>THEN</u> go to Step 9.
	b. RCS subcooling - GREATER THAN 25°F(55°F FOR ADVERSE CONTAINMENT)	b. Go to Step 9
	c. RCS pressure - STABLE <u>OR</u> INCREASING	c. Go to Step 9.
	d. PRZR level - GREATER THAN 13%(34% FOR ADVERSE CONTAINMENT)	d. Go to Step 9.
	e. Go to EOS-1.1A, SAFETY INJECTION TERMINATION, Step 1.	

Comments / Reference: EOP-2.0A, Step 9	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 6 OF 14

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	Go To EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	

-END-

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		1
Change: 2	Group		1
	K/A	054 AA2.03	
Level of Difficulty: 4	Importance Rating		4.2

Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater: Conditions and reasons for AFW pump startup

Proposed Question: 80

Which of the following would result in the start of the Unit 2 Motor Driven Auxiliary Feedwater (AFW) Pumps and the Technical Specification Bases for the AFW Pump startup?

A. Narrow Range Level in ONE Steam Generator lowering to 37% following a Reactor Trip.
Ensure that at least ONE Steam Generator is available with water to act as a heat sink for the Reactor.

B. Trip of Main Feedwater Pump 2A at 18% Reactor Power.
Ensure that at least ONE Steam Generator is available with water to act as a heat sink for the Reactor.

C. Narrow Range Level in ONE Steam Generator lowering to 37% following a Reactor Trip.
Ensure that at least TWO Steam Generators are available with water to act as a heat sink for the Reactor.

D. Trip of Main Feedwater Pump 2A at 18% Reactor Power.
Ensure that at least TWO Steam Generators are available with water to act as a heat sink for the Reactor.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible at a single SG lowering to 37% on Unit 1 would start both Motor Driven AFW Pumps, however, on Unit 2 the level would need to lower to 35.4%. According to Technical Specification (TS) LCO 3.3.2 Bases, the reason for the AFW Pump start is to ensure at least one SG is available to act as a heat sink.
B. Correct. At 18% power only one Main Feedwater Pump would be in operation, however, the other Main Feedwater Pump would either be in a tripped condition or the trip oil pressure switches isolated to indicate a trip to the RPS. Thus, a trip of the operating Main Feedwater Pump would start both motor driven AFW Pumps. According to TS LCO 3.3.2 Bases, the reason for the AFW pump start is to ensure at least one SG is available to act as a heat sink.
C. Incorrect. Plausible at a signal SG lowering to 37% on Unit 1 would start both Motor Driven AFW Pumps, however on Unit 2 the level would need to lower to 35.4%. According to TS LCO 3.3.2 Bases, the reason as stated is incorrect in that the start signal only ensures at least ONE SG is available and not TWO.
D. Incorrect. Plausible because the signal for starting the Motor Driven AFW Pumps is correct, however, the reason as stated is incorrect in that the start signal only ensures at least ONE SG is available and not TWO.

Technical Reference(s): Technical Specification LCO 3.3.2 Bases; IPO-003B, Step 4.1.17 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Auxiliary Feedwater System including Technical Specifications, TRM and ODCM.

Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New	X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments / Reference: IPO-003B, Step 4.1.17	Revision: 19

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. IPO-003B
POWER OPERATIONS	REVISION NO. 19 CONTINUOUS USE	PAGE 11 OF 222

4.1.17 Automatic start of the Motor Driven AFW Pumps on the trip of both Main Feedwater Pumps is required Operable in MODES 1 and 2 (TS 3.3.2, Table 3.3.2-1 Function 6.g) to ensure a supply of water to at least one SG for heat sink availability. In MODE 1 or 2 with one MFP supplying flow to the SGs (AFW pumps stopped), the second MFP must remain tripped or have the trip oil pressure switches isolated to ensure compliance with TS 3.3.2. (CR-2010-000638)

Comments / Reference: Technical Specification LCO 3.3.2 Bases	Revision: 68

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

decay heat removal. During a loss of offsite power, to both safety related busses feeding the motor driven AFW pumps, the loss of power to the bus feeding the turbine driven AFW pump valve control motor will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. In addition, once the diesel generators are started and up to speed, the motor driven AFW pumps will be sequentially loaded onto the diesel generator busses.

Functions 6.a through 6.e must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level-Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level-Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

f. Not Used

g. Auxiliary Feedwater - Trip of All Main Feedwater Pumps

A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. Each turbine driven MFW pump is equipped with two pressure switches on the oil line for the speed control system. A Train "A" and a Train "B" sensor is on each MFW pump. The Train "A(B)" trip signals from both MFW pumps are required to actuate the Train "A(B)" motor-driven auxiliary feedwater pump. A trip of all MFW pumps starts the motor driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

Comments / Reference: Technical Specification 3.3.2 Bases	Revision: 68

ESFAS Instrumentation
B 3.3.2

Table B 3.3.2-1 (Page 2 of 3)
ESFAS Trip Setpoints

FUNCTION	NOMINAL TRIP SETPOINT
4. Steam Line Isolation	
a. Manual Initiation	NA
b. Automatic Actuation Logic and Actuation Relays	NA
c. Containment Pressure - High 2	6.2 psig
d. Steam Line Pressure	
(1) Low	605 psig $\hat{o}_1 \geq 10$ seconds $\hat{o}_2 \leq 5$ seconds
(2) Negative Rate - High	100 psi $\hat{o} \geq 50$ seconds
5. Turbine Trip and Feedwater Isolation	
a. Automatic Actuation Logic and Actuation Relays	NA
b. SG Water Level - High-High (P-14)	84% NR (Unit 1) 81.5% NR (Unit 2)
c. Safety Injection	See Function 1.
6. Auxiliary Feedwater	
a. Automatic Actuating Logic and Actuation Relays (SSPS)	NA
b. Not Used	
c. SG Water Level - Low-Low	38% NR (Unit 1) 35.4% NR (Unit 2)

Comments / Reference: Technical Specification 3.3.2 Bases	Revision: 68

ESFAS Instrumentation
B 3.3.2

Table B 3.3.2-1 (Page 3 of 3)
ESFAS Trip Setpoints

FUNCTION	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)	
g. Trip of All Main Feedwater Pumps	NA
h. Not Used.	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 03/25/2014	Tier		1
Change: 4	Group		1
	K/A	057 G.2.4.1	
Level of Difficulty: 3	Importance Rating		4.8

Loss of Vital AC Electrical Instrument Bus: Emergency Procedures/Plan: Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: 81

Given the following conditions:

- Unit 2 is at 20% power and the following annunciator just alarmed.
- 2-ALB-10B, Window 1.15 – 118V INV IV2EC1 TRBL.
- The BOP reports that Instrument Bus 2EC1 is de-energized.

Which of the following describes the correct action and Technical Specification entry?

A. Trip the Reactor and enter EOP-0.0B, Reactor Trip or Safety Injection.
LCO 3.8.1, AC Sources – Operating.

B. Trip the Reactor and enter EOP-0.0B, Reactor Trip or Safety Injection.
LCO 3.7.3 Feedwater Isolation Valves (FIVs) and Feedwater Control Valves (FCVs) and Associated Bypass Valves.

C. Restore power to bus 2EC1 per ABN-603, Loss of Protection or Instrument Bus.
LCO 3.8.1, AC Sources – Operating.

D. Restore power to bus 2EC1 per ABN-603, Loss of Protection or Instrument Bus.
LCO 3.7.3 Feedwater Isolation Valves (FIVs) and Feedwater Control Valves (FCVs) and Associated Bypass Valves.

Proposed Answer: A

Explanation:

A. Correct. Feedwater isolates due to the loss of power on 2EC1 so a manual reactor trip is required prior to receiving an automatic reactor trip on SG low level. TS 3.8.1 entry is required due to loss of DG automatic emergency start capability and loss of power to the associated BO sequencer.
B. Incorrect. Plausible because Feedwater isolates due to the loss of power on 2EC1 so a manual reactor trip is required prior to receiving an automatic reactor trip on SG low level. On Unit 2 feedwater isolates on a loss of 2EC1 power so it could be thought that LCO 3.7.3 is the correct TS entry however feedwater is isolated which is it required position for TS LCO 3.7.3.
C. Incorrect. Plausible because Unit 1 does not receive a feedwater isolation signal on loss of power to 1EC1. So based on the unit difference entering ABN-603 could be considered the correct action. Entering TS LCO 3.8.1 is the correct TS entry due to loss of DG automatic emergency start capability and loss of power to the associated BO sequencer.
D. Incorrect. Plausible because Unit 1 does not receive a feedwater isolation signal on loss of power to 1EC1. So based on the unit difference entering ABN-603 could be considered the correct action. On Unit 2 feedwater isolates on a loss of 2EC1 power so it could be thought that LCO 3.7.3 is the correct TS entry however feedwater is isolated which is it required position for TS LCO 3.7.3.

Technical Reference(s): ABN-603, Steps 3.1, 3.3.1 & 3.3.2
Technical Specification LCO 3.8.1
Technical Specification LCO 3.7.3

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken, both initial and subsequent, for ABN-604, Loss of Non-1E Instrument Bus.

Question Source:
Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments / Reference: ABN-603, Step 3.1	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 16 OF 34

3.0 LOSS OF INSTRUMENT BUS

3.1 Symptoms

a. The affected inverter trouble alarm:

- 118V INV IVuEC1 TRBL (10B-1.15)
- 118V INV IVuEC2 TRBL (10B-2.15)
- 118V INV IVuEC3 TRBL (10B-2.18)
- 118V INV IVuEC4 TRBL (10B-3.18)
- 118V INV IVuEC1/3 TRBL (10B-1.18)
- 118V INV IVuEC2/4 TRBL (10B-4.18)

b. The associated bus instruments alarming or failing (see Attachments 3 and 4):

- uEC1 from IVuEC1
- uEC2 from IVuEC2
- uEC5 from IVuEC3
- uEC6 from IVuEC4

NOTE: **(Unit 2 only)** On loss of 2EC1 or 2EC2, the FWIVs close due to loss of water hammer interlocks and the FPBVs open. During low power operations, this could cause overheating of the containment penetrations. 2-ALB-8A, 1.5, 2.5, 3.5 and 4.5 contain actions should this occur.

c. **(Unit 2 only)** A feed isolation will occur, FWIVs close (loss of 2EC1 or 2EC2).

d. A DG 86-2 lockout relay (loss of uEC1 or uEC2) will prevent diesel start on loss of power.
The DG will not emergency start due to loss of power to its emergency start relay. |
The diesel generators can be manually started in the emergency mode if needed. |

e. If diesel running due to a loss of offsite power, a loss of uEC1 or uEC2 will restore normal trips and stop the diesel due to a DG 86-2 lockout relay.

f. Thermal barrier return isolation (u-HV-4696) closes (loss of uEC1)

g. During the period uEC1 or uEC2 is powered from bypass power, the Black Out Sequencer is inoperable per TS 3.8.1 |
|

3.2 Automatic Actions

None

Comments / Reference: ABN-603, Step 3.3.1 & 3.3.2	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 17 OF 34

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Check Unit status.	
☐ a. Verify Unit - IN MODE 5 OR 6	a. GO TO Step 2.
☐ b. Verify NONE of the following - IN PROGRESS: • Core alterations • Positive reactivity addition of ANY type. • Movement of irradiated fuel assemblies	b. Perform the following: 1) Stop operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. 2) Suspend any core alterations OR fuel movement in progress.

NOTE:
- If uEC1 or uEC2 are powered from alternate power, the respective sequencer is INOPERABLE and, upon loss of power, the associated DG will not start due to an 86-2 lockout relay.
- It may be necessary to transfer control of Trn B MDAFW and TDAFW SG flow control valves from RSP to Control Room after power restored.

☐ 2 Dispatch an Operator to reenergize the affected instrument bus by moving the manual transfer switch to the alternate power supply (bottom of instrument panel).

☐ 3 INITIATE actions to place the swing inverter in service per SOP-607A/B "118 VAC DISTRIBUTION SYSTEM AND INVERTERS"

Comments / Reference: Technical Specification LCO 3.8.1	Amendment: 161

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

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

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

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

APPLICABILITY: MODES 1, 2, 3, and 4

------------------------------NOTE------------------------------
One DG may be synchronized with the offsite power source under administrative controls for the purpose of surveillance testing.

Comments / Reference: Technical Specification LCO 3.7.3	Amendment: 161

3.7 PLANT SYSTEMS

3.7.3 Feedwater Isolation Valves (FIVs) and Feedwater Control Valves (FCVs) and Associated Bypass Valves

LCO 3.7.3 Four FIVs, four FCVs, and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when FIV, FCV or associated bypass valve is either closed and de-activated or isolated by a closed manual valve.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more FIVs inoperable.	A.1 Close or isolate FIV. <u>AND</u> A.2 Verify FIV is closed or isolated.	72 hours Once per 7 days
B. One or more FCVs inoperable.	B.1 Close or isolate FCV. <u>AND</u> B.2 Verify FCV is closed or isolated.	72 hours Once per 7 days

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2013	Tier		1
Change: 3	Group		2
	K/A	005 AA2.03	
Level of Difficulty: 3	Importance Rating		4.4

Inoperable/Stuck Control Rod: Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

Proposed Question: 82

Given the following conditions:

- Unit 2 is at 100% power.
- OPT-106B, Control Rod Exercise is in progress.
- While testing Shutdown Bank A, rods D14 and P12 stuck at 220 steps and will not move.
- The PROMPT Team and System Engineering reported that there is not an electrical problem and that the two control rods are mechanically bound.

Which of the following is the required Technical Specification ACTION?

A. Initiate boration to restore Shutdown Margin greater than or equal to 1.3% Δk/k within 15 minutes AND restore Shutdown Bank A to within limits within 2 hours.

B. Verify Shutdown Margin greater than or equal to 1.3% Δk/k within 1 hour AND be in MODE 3 within 6 hours.

C. Initiate boration to restore Shutdown Margin greater than or equal to 1.3% Δk/k within 15 minutes OR restore Shutdown Bank A to within limits within 2 hours.

D. Verify Shutdown Margin greater than or equal to 1.3% Δk/k within 1 hour OR be in MODE 3 within 6 hours.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because a loss of SDM in accordance with Technical Specification LCO 3.1.1 requires boration to restore SDM within 15 minutes but Technical Specification LCO 3.1.1 does not apply. Also it could be thought that 220 steps on rods D14 and P12 are not within the limits of Technical Specification LCO 3.1.5 however the limits are 218 to 231 steps.

B. Correct. In accordance with Technical Specification LCO 3.1.4 SDM must be verified within COLR limits within 1 hour and the Unit must be in MODE 3 within 6 hours.

C. Incorrect. Plausible because it could be thought that the 15 minute requirement of Technical Specification LCO 3.1.1 applies however it does not apply. Also it could be thought that 220 steps on rods D14 and P12 are not within the limits of Technical Specification LCO 3.1.5 however the limits are 218 to 231 steps and that either action would meet the Technical Specification action requirements.

D. Incorrect. Plausible because it could be thought that verifying SDM or placing the unit in MODE 3 would satisfy Technical Specification LCO 3.1.4 actions however both must be accomplished.

Technical Reference(s): ABN-712, Step 2.3.7; Technical Specification LCO 3.1.1; Technical Specification LCO 3.1.4; Technical Specification LCO 3.1.5

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: ____

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41 ____
55.43 2

Comments / Reference: ABN-710, Step 2.3.7	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 7 OF 52

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 7 Verify Affected Rod(s) - TRIPPABLE • Affected rods capable of motion (i.e. not stuck or mechanically bound). OPT-106A/B may be used to make this determination if required.	Perform the following per TS 3.1.4: a. Within 1 hour verify SDM to be within limits provided in the COLR OR initiate boration to restore SDM within limit. AND Within 6 hours, place unit in HOT STANDBY per IPO-003A/B. b. Initiate LCOAR, as necessary.

Comments / Reference: Technical Specification LCO 3.1.1	Amendment: 161

SDM
3.1.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5

-----NOTE-----
While this LCO is not met, entry into MODE 5 from MODE 6 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

Comments / Reference: Technical Specification LCO 3.1.4	Amendment: 161

Rod Group Alignment Limits
3.1.4

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

Comments / Reference: Technical Specification LCO 3.1.5	Amendment: 161

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with any control bank not fully inserted.

------------------------------------NOTE------------------------------------
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM to be within the limits provided in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Restore shutdown banks to within limits.	1 hour 1 hour 2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		1
Change: 2	Group		2
	K/A	028 AA2.08	
Level of Difficulty: 3	Importance Rating		3.5

Pressurizer Level Control Malfunction: Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: PZR level as a function of power level

Proposed Question: 83

Given the following conditions:

- Unit 1 is at 20% ramping to 100% power.
- Rod Control is in MANUAL.
- Average T_{AVE} is 563°F.
- Pressurizer level is 32%.
- Loop 1 T_{AVE} fails to 630°F.

Assuming no operator action is taken, which of the following indicates final Pressurizer level and the applicable Technical Specification LCO?

Pressurizer level is…

A. …53%. Technical Specification LCO 3.3.1, Condition M for Pressurizer Level.

B. …60%. Technical Specification LCO 3.3.1 Condition M for Pressurizer Level.

C. …53%. Technical Specification LCO 3.3.1 Condition E for OT N16 and OP N16.

D. …60%. Technical Specification LCO 3.3.1 Condition E for OT N16 and OP N16.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 53% PRZR level is the correct final PRZR level, however, PRZR level Technical Specification (TS) LCO 3.3.1, Condition M is not entered, LCO 3.3.1 Condition E is entered for the failed T_{AVE} channel.

B. Incorrect. Plausible because it could be thought that PRZR level would fail to maximum level due to a single T_{AVE} channel failing high, however, the PRZR level program uses AVE T_{AVE}, <u>not</u> high T_{AVE}. It could be thought that PRZR level TS requires entry but it does not, TS LCO 3.3.1 Condition E for the failed T_{AVE} channel must be entered.

C. Correct. From 0 to 100% power T_{AVE} rises from 557°F to 585.4°F. PRZR programmed level rises from 25% to 60% from 0 to 100% power due to AVE T_{AVE} input to the PRZR level controller. When Loop 1 T_{AVE} fails to 630°F, AVE T_{AVE} goes to 580°F. 580°F AVE Tave is equivalent to 80% power. PRZR program level changes 0.35% for every 1% power change. Therefore, 0.35% x 80% + 25% = 53% PRZR level. TS LCO 3.3.1, Condition E must be entered for the T_{AVE} failure.

D. Incorrect. Plausible because the correct TS condition is 3.3.1 Condition E the correct TS entry, however, final PRZR level will be 53% not 60% because PRZR programmed level uses AVE T_{AVE} <u>not</u> high T_{AVE} for input.

Technical Reference(s): ABN-704, Sections 2.1 & 2.2; ABN-704, Attachment 3; TDM-301, Programmed T_{AVE} Curve; Technical Specification Table 3.3.1-1

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Pressurizer Level Control System including Technical Specifications, TRM and ODCM.

Question Source:

Bank		
Modified Bank		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	
55.43	2

Comments / Reference: ABN-704, Section 2.1	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-704
Tc/N-16 INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 3 OF 14

2.0 Tc/N-16 Instrumentation Malfunction

2.1 Symptoms

a. Annunciator Alarms

- ANY N16 DEV HI/LO (5C-1.5)
- 1 of 4 OT N16 HI (5C-2.5)
- ANY Tave DEV HI/LO (5C-3.5)
- 1 of 4 OP N16 HI (5C-2.6)
- AVE Tave-Tref DEV (6D-1.10)
- AVE Tave HI (6D-2.10)
- 1 of 4 Tave LO-LO (6D-3.10)
- Tref-AUCT LO TAVE MISMATCH (6D-3.13)
- AUCT TAVE LO (6D-4.13)
- 1 of 4 OT N16 ROD STOP & TURB RUNBACK (6D-3.14)
- 1 of 4 OP N16 ROD STOP & TURB RUNBACK (6D-2.13)

b. Plant Indications

1) One Tc channel higher or lower than the other three.

- u-TI-411A, CL 1 TEMP (NR) CHAN I
- u-TI-421A, CL 2 TEMP (NR) CHAN II
- u-TI-431A, CL 3 TEMP (NR) CHAN III
- u-TI-441A, CL 4 TEMP (NR) CHAN IV

2) One Tave channel higher or lower than the other three.

- u-TI-412, RC LOOP 1 Tave CHAN I
- u-TI-422, RC LOOP 2 Tave CHAN II
- u-TI-432, RC LOOP 3 Tave CHAN III
- u-TI-442, RC LOOP 4 Tave CHAN IV

Comments / Reference: ABN-704, Section 2.2	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-704
Tc/N-16 INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 4 OF 14

2.1 b. 4) One OTN16 setpoint higher or lower than the other three.

- u-TI-411B, RC LOOP 1 OT N16 SETPOINT CHAN I
- u-TI-421B, RC LOOP 2 OT N16 SETPOINT CHAN II
- u-TI-431B, RC LOOP 3 OT N16 SETPOINT CHAN III
- u-TI-441B, RC LOOP 4 OT N16 SETPOINT CHAN IV

5) One OPN16 setpoint higher or lower than the other three.

- u-NI-411A, RC LOOP 1 OP N16 SETPOINT CHAN I
- u-NI-421A, RC LOOP 2 OP N16 SETPOINT CHAN II
- u-NI-431A, RC LOOP 3 OP N16 SETPOINT CHAN III
- u-NI-441A, RC LOOP 4 OP N16 SETPOINT CHAN IV

2.2 Automatic Actions

a. Any failure that results in an increased Loop Tave will cause Average Tave to be higher with the following actions:

- Rapid control rod insertion due to Tave-Tref mismatch if in AUTO.
- Steam dumps will open if armed with a C-7 (loss of load).
- Pressurizer reference level increase (to a maximum of 60%) with charging flow increase when in auto.

Comments / Reference: ABN-704, Attachment 3	Revision: 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-704
Tc/N-16 INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 11 OF 14

ATTACHMENT 3
PAGE 1 OF 4

ANNUNCIATOR ALARMS AND TRIP STATUS LIGHTS

LOOP 1, PROT I, CH 0411B

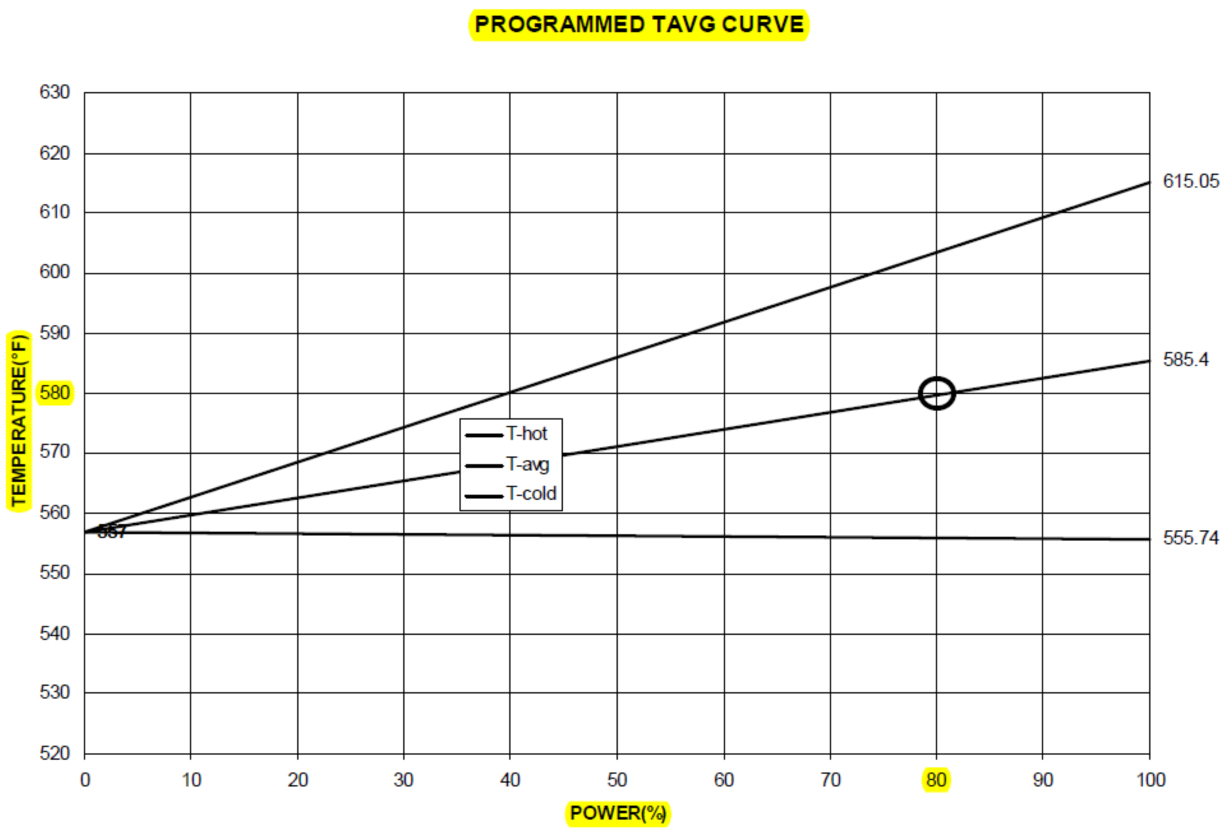
	ALARM	ANN. WINDOW	PANEL
☐	1 OF 4 OT N16 HI	2.5	ALB-5C
☐	1 OF 4 OP N16 HI	2.6	ALB-5C
☐	ANY TAVE DEV HI/LO	3.5	ALB-5C
☐	1 OF 4 OP N16 ROD STOP & TURB RUNBACK	2.13	ALB-6D
☐	1 OF 4 TAVE LO-LO	3.10	ALB-6D
☐	1 OF 4 OT N16 ROD STOP & TURB RUNBACK	3.14	ALB-6D
☐	* ANY N16 DEV HI/LO	1.5	ALB-5C

	TRIP STATUS	STATUS INDICATOR	TRIP STATUS PANEL
☐	RC LOOP 1 TAVE LO TB-412G	1.1	TSLB-3
☐	RC LOOP 1 OT N16 TB-411C	1.8	TSLB-5
☐	RC LOOP 1 OP N16 JB-411D	1.9	TSLB-5
☐	RC LOOP 1 TAVE LO-LO TB-412D	1.9	TSLB-9
☐	OT N16 ROD STOP & TURB RUNBACK TB-411D	1.4	TSLB-9
☐	OP N16 ROD STOP & TURB RUNBACK JB-411C	1.5	TSLB-9

Comments / Reference: TDM-301A, Programmed T_{AVE} Curve	Revision: 10

CPNPP TECHNICAL DATA MANUAL	UNIT 1	PROCEDURE NO. TDM-301A
RCS TEMPERATURE & PRESSURE LIMITS	REVISION NO. 10 INFORMATION USE	PAGE 5 OF 7

PROGRAMMED TAVE CURVE



Comments / Reference: Technical Specification Table 3.3.1-1	Amendment: 161

RTS Instrumentation
3.3.1

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
5. Source Range Neutron Flux	2(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3(b), 4(b), 5(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps
6. Overtemperature N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1(q)(r)
7. Overpower N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 112.8% RTP (q)(r)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/26/2014	Tier		1
Change: 3	Group		2
	K/A	W/E06 EA2.2	
Level of Difficulty: 3	Importance Rating		4.1

Degraded Core Cooling: Ability to determine and interpret the following as they apply to the Degraded Core Cooling: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments to degraded core cooling

Proposed Question: 84

Given the following conditions:

- Unit 1 is responding to a large break Loss of Coolant Accident (LOCA).
- Cold leg recirculation alignment is in progress in accordance with EOS-1.3A, Transfer to Cold Leg Recirculation.
- The Reactor Operator is closing 1/1-8812A, RWST TO RHRP 1 SUCT VLV and 1/1-8812B, RWST TO RHRP 2 SUCT VLV.
- The Shift Technical Advisor announces that a loss of Critical Safety Function exists on Core Cooling due to Core Exit Thermocouple (CET) readings on six CETs as follows;
    - CET A06 = 763°F
    - CET J12 = 781°F
    - CET N08 = 792°F
    - CET G08 = 794°F
    - CET R10 = 787°F
    - CET L04 = 767°F

Applying the Emergency Response Guidelines "rules of usage" which of the following describes the required procedural actions?

A. Immediately transition to FRC-0.1A, Response to Inadequate Core Cooling and complete alignment of ECCS for cold leg recirculation after completing FRC-0.1A.

B. Immediately transition to FRC-0.2A, Response to Degraded Core Cooling and complete alignment of ECCS for cold leg recirculation after completing FRC-0.2A.

C. Complete alignment of ECCS for cold leg recirculation then transition to FRC-0.1A, Response to Inadequate Core Cooling.

D. Complete alignment of ECCS for cold leg recirculation then transition to FRC-0.2A, Response to Degraded Core Cooling.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that immediate transition due to inadequate core cooling is required this is partially based on the belief that a RED path exists but CET temperatures must be greater than 1200°F for the RED path.

B. Incorrect. Plausible because it could be thought that immediate transition is required due to the degraded core cooling (FRC-0.2A) but the NOTE prior to Step 1 in EOS-1.3A requires completion of ECCS cold leg Recirc alignment before addressing the FRC status tree.

C. Incorrect. Plausible because alignment of ECCS cold leg Recirc must be completed prior to addressing the FRC status tree but FRC-0.1A requires CET temperatures to be greater than 1200°F.

D. Correct. The NOTE prior to Step 1 of EOS-1.3A states that FRGs should not be implemented until after completion of step 3 (ECCS Cold Leg Recirc alignment). FRC-0.2A is the correct transition after aligning ECCS for cold leg Recirc due to CETs greater than 750°F but less than 1200°F.

Technical Reference(s) EOS-1.3A, Step 1 NOTE
FRC-0.1A, Core Cooling CSFST
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:

Question Source: Bank
Modified Bank (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

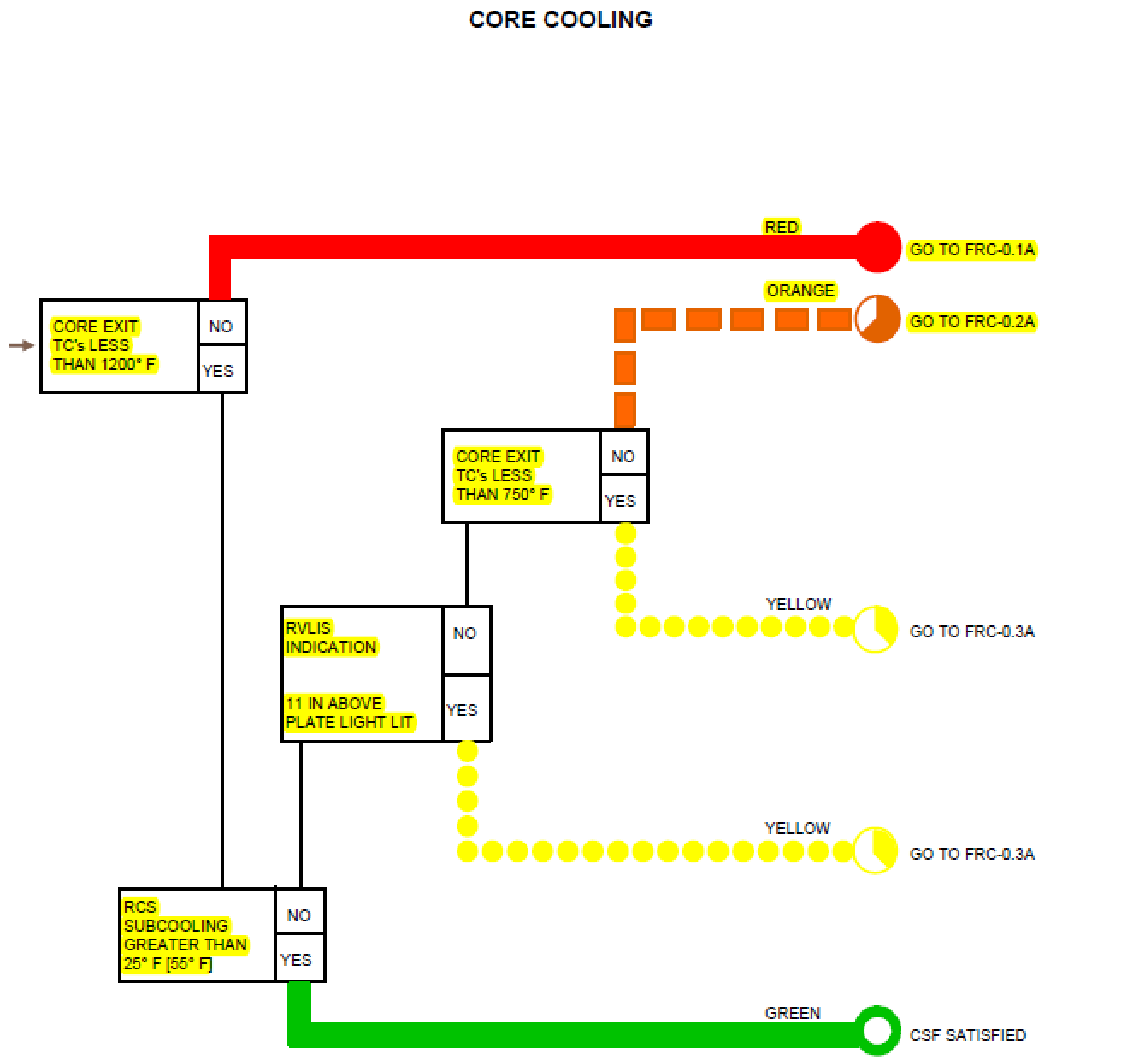
Comments / Reference: EOS-1.3A, Step 1 NOTE	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 3 OF 54

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: Steps 1 through 3 should be performed without delay. FRGs should not be implemented prior to completion of these steps.

Comments / Reference: FRC-0.1A, Core Cooling CSFST	Revision: 8



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		1
Change: 4	Group		2
	K/A	067 AA2.15	
Level of Difficulty: 2	Importance Rating		3.9

Plant Fire on Site: Ability to determine and interpret the following as they apply to the Plant Fire on Site: Requirements for establishing a fire watch

Proposed Question: 85

Given the following conditions:

- Unit 2 is MODE 3 preparing for refueling.
- RCS temperature is 410°F with a cooldown in progress.
- The Unit 2 Safeguards NEO reports that Door S2-18, to the TDAFWP 2-01 Room is ajar and will NOT latch closed.
- Door S2-18 is rated a 3-hour fire door.
- Fire detection status on the 790' elevation of the Unit 2 Safeguards Building has been verified OPERABLE.

Which of the following are the MINIMUM required Fire Protection actions and status of the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) operability, in accordance with STA-738, Fire Protection Systems/Equipment Impairments and Technical Specification LCO 3.7.5, Auxiliary Feedwater (AFW) System?

A. Establish a continuous fire watch with backup fire suppression within 1 hour.
The TDAFWP remains OPERABLE.

B. Establish a continuous fire watch with backup fire suppression within 1 hour.
The TDAFWP is inoperable.

C. Establish an hourly fire watch patrol of Door S2-18 within 1 hour.
The TDAFWP remains OPERABLE.

D. Establish an hourly fire watch patrol of Door S2-18 within 1 hour.
The TDAFWP is inoperable.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because in accordance with STA-738, Attachment 8.A 7), with Door S2-18 impaired, establish a continuous fire watch on one side of impaired fire rated assembly within 1 hour OR verify operability of fire detection on one side of impaired fire rated assembly and establish an hourly fire watch. The establishment of the continuous fire watch would meet this requirement, however, the question asks for the minimum required and requiring backup fire suppression is not needed for this impairment as it would be for an impairment to a Spray and/or Sprinkler System. TDAFWP remains OPERABLE per TS LCO 3.7.5 due to compensatory actions of STA-738.

B. Incorrect. Plausible because in accordance with STA-738, Attachment 8.A 7), with Door S2-18 impaired, establish a continuous fire watch on one side of impaired fire rated assembly within 1 hour OR verify operability of fire detection on one side of impaired fire rated assembly and establish an hourly fire watch. The establishment of the continuous fire watch would meet this requirement, however, the question asks for the minimum required and requiring backup fire suppression is not needed for this impairment as it would be for an impairment to a Spray and/or Sprinkler System. Additionally, the TDAFWP remains OPERABLE per TS LCO 3.7.5 due to compensatory actions of STA-738.

C. Correct. In accordance with STA-738, Attachment 8.A 7), with Door S2-18 impaired, establish a continuous fire watch on one side of impaired fire rated assembly within 1 hour OR verify operability of fire detection on one side of impaired fire rated assembly and establish an hourly fire watch. As the operability of the fire detection has been verified, the one hour fire watch patrol satisfies the compensatory measure. TDAFWP remains OPERABLE per TS LCO 3.7.5 due to compensatory actions of STA-738.

D. Incorrect. Plausible because in accordance with STA-738, Attachment 8.A 7), with Door S2-18 impaired, establish a continuous fire watch on one side of impaired fire rated assembly within 1 hour OR verify operability of fire detection on one side of impaired fire rated assembly and establish an hourly fire watch. As the operability of the fire detection has been verified, the one hour fire watch patrol satisfies the compensatory measure. However, the TDAFWP remains OPERABLE per TS LCO 3.7.5 due to compensatory actions of STA-738.

Technical Reference(s) STA-738, Attachment 8.A Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: STA-738, Attachment 8.A

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source: Bank ______
Modified Bank ILOT8068 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 1

Comments / Reference: STA-738, Attachment 8.A	Revision: 6

CPSES STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-738
FIRE PROTECTION SYSTEMS/EQUIPMENT IMPAIRMENTS	REVISION NO. 6	PAGE 17 OF 45

ATTACHMENT 8.A
PAGE 5 OF 6

GUIDELINES FOR COMPENSATORY MEASURES

[C] 6) YARD FIRE HYDRANTS/FIRE HOSE HOUSES

With one or more of the yard fire hydrants or associated hydrant hose houses listed in Attachment 8.E inoperable, within 1 hour provide sufficient additional lengths of 2-1/2 inch diameter hose located at an adjacent OPERABLE fire hydrant to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide additional fire hose within 24 hours. It is also acceptable to provide suppression coverage to the area left unprotected by an inoperable yard fire hydrant or hydrant hose house by using a yard fire hydrant not listed in Attachment 8.E, if that fire hydrant can provide an equivalent quantity of water and pressure.

[C] 7) FIRE RATED ASSEMBLIES

With one or more of the fire rated assemblies (fire dampers, fire walls, fire doors, penetration seals, thermolag and radiant energy shield) listed in Attachment 8.F impaired or inoperable; establish within 1 hour a continuous fire watch on one side of the affected assembly, or verify operability of the fire detection on at least one side of the impaired/inoperable fire rated assembly and establish an hourly fire watch patrol.

For all other rated assemblies (i.e., Risk Management areas), compensatory measures should be determined by the Fire Protection Supervisor.

Comments / Reference: STA-738, Attachment 8.A	Revision: 6

CPSES STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-738
FIRE PROTECTION SYSTEMS/EQUIPMENT IMPAIRMENTS	REVISION NO. 6	PAGE 15 OF 45

ATTACHMENT 8.A
PAGE 3 OF 6

GUIDELINES FOR COMPENSATORY MEASURES

(5) Suppression and Detection System Inoperable - Within one hour establish a continuous fire watch.

3) b) With one or more of the required spray and/or sprinkler systems in the Diesel Generator Building inoperable, establish an hourly roving fire watch patrol within one hour.

c) With one or more of the required Spray and/or Sprinkler Systems listed in Attachment 8.C inoperable, establish a continuous fire watch with backup fire suppression equipment within 1 hour. For Zone V radiation areas, the area shall be inspected at least once per 8 hours, with backup fire suppression equipment established within 1 hour for the inoperable system.

Original Question: ILOT8068

Given the following conditions:

- Unit 2 is in the process of being shutdown for refueling.
- RCS temperature is 410°F with a cooldown in progress.
- The Unit 2 Safeguards NEO reports that Door S2-18, to the TDAFWP 2-01 Room is ajar and will NOT latch closed.
- Door S2-18 is rated a 3-hour fire door.

Which of the following Fire Protection actions and Technical Specification conditions are applicable?

A. Establish a continuous fire watch on either side of Door S2-18 within 1 hour, all three (3) Auxiliary Feedwater trains remain OPERABLE per TS LCO 3.7.5.

B. Establish a continuous fire watch on either side of Door S2-18 within 1 hour, one (1) Auxiliary Feedwater train is INOPERABLE per TS LCO 3.7.5.

C. Verify operability of fire detection on either side of Door S2-18 within 3 hours, all three (3) Auxiliary Feedwater trains remain OPERABLE per TS LCO 3.7.5.

D. Verify operability of fire detection on either side of Door S2-18 within 3 hours, one (1) Auxiliary Feedwater train is INOPERABLE per TS LCO 3.7.5.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		2
Change: 2	Group		1
	K/A	003 A2.05	
Level of Difficulty: 2	Importance Rating		2.8

Reactor Coolant Pump System: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leak off flows

Proposed Question: 86

Given the following conditions:

- Unit 1 is at 100% power when the following are observed:
  - 1-PI-115, VCT PRESS indicates 68 psig.
  - 1-ALB-6A, Window 1.5 – VCT PRESS HI/LO alarms.
  - Volume Control Tank (VCT) level is 52% and stable.

Which of the following describes the effect of the rising VCT pressure on Reactor Coolant Pump seal 1 leakoff flow and what procedure is used to mitigate the malfunction?

Reactor Coolant Pump 1 seal leakoff flow...

A. ...increases; Refer to ALM-0061A, Alarm Procedure 1-ALB-6A.

B. ...increases; Refer to ABN-105, Chemical and Volume Control System Malfunction.

C. ...decreases; Refer to ALM-0061A, Alarm Procedure 1-ALB-6A.

D. ...decreases; Refer to ABN-105, Chemical and Volume Control System Malfunction.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because seal 2 leakoff would increase and ALM-0061A is the correct procedure.

B. Incorrect. Plausible because seal 2 leakoff would increase, but ABN-105 is not the correct procedure. A misconception could exist that the CVCS Malfunction ABN would provide the necessary procedural guidance.

C. Correct. When VCT pressure increases RCP 1 seal leakoff flow decreases. ALM-0061A directs the operator to SOP-103 which is used to lower VCT pressure.

D. Incorrect. Plausible because seal 1 leakoff will decrease, but ABN-105 is not the correct procedure. A misconception could exist that the CVCS Malfunction ABN would provide the necessary procedural guidance.

Technical Reference(s) ALM-0061A, 1-ALB-6A, Window 1.5 Attached w/ Revision: See

ABN-105, Section 1.0 Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrumentation while responding to a Reactor Coolant Pump System malfunction.

Question Source:	Bank		
	Modified Bank	ILOT5838	(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments / Reference: ABN-105, Section 1.0	Revision: 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-105
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 2 OF 41

1.0 APPLICABILITY

This procedure describes the actions required in the event of a failure or malfunction of the CVCS and applies to MODES 1, 2, 3, 4, and 5 only.

This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for this symbol to obtain the unit specific equipment number. (Example u-FK-121 represents 1-FK-121 for Unit 1 and 2-FK-121 for Unit 2.)

- Section 2.0 - Pressurizer Level Decreasing Below Program Level
- Section 3.0 - Charging Pump Trip
- Section 4.0 - Pressurizer Level Increasing Above Program Level
- Section 5.0 - Loss of Letdown
- Section 6.0 - Reactor Makeup System Malfunction
- Section 7.0 - Gas Binding/Cavitation of Charging Pumps
- Section 8.0 - Dilution Anomaly

Comments / Reference: ALM-0061A, 1-ALB-6A, Window 1.5	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 17 OF 79

ANNUNCIATOR NOM./NO.: **VCT PRESS HI/LO** **1.5**

PROBABLE CAUSE:

1-LK-112C, VCT LVL CTRL malfunction
1/1-PCV-115, VCT GWPS ISOL VLV malfunction
VCT pressure regulator malfunction
Degassing the reactor coolant system

AUTOMATIC ACTIONS:

1/1-PCV-115, VCT GWPS ISOL VLV closes on low VCT pressure.

> NOTE: 1/1-PCV-115 closes on low VCT pressure (14 psig), low waste gas compressor suction pressure (-0.5 psig), or if both hydrogen recombiner oxygen valves are closed. 1/1-PCV-115 fails closed on loss of air or power. 1-8120, VCT 1-01 TO REHUT RLF VLV set pressure is 75 psig. 1-PCV-115, 1-TC-09 FB 2 Fuse 13 and 15.

OPERATOR ACTIONS:

> NOTE: Normal position for 1/1-LCV-112A is VCT.

1. Monitor 1-PI-115, VCT PRESS.
   A. If pressure is low, ensure 1/1-PCV-115, VCT GWPS ISOL VLV is closed.

[C] 2. Monitor other indication for possible charging pump gas intrusion such as:
   - window 1.8 - PDP SUCT STAB LVL HI-HI
   - window 3.4 - CHRG FLO HI/LO
   - window 4.5 - VCT LVL LO-LO
   - CHRG PMP pressure or flow oscillations

   A. If gas intrusion is indicated, refer to ABN-105 for Gas Binding/Cavitation of Charging Pumps .

3. Verify VCT level is between 46% and 56% on 1-LI-112A, VCT LVL and 1-LI-185, VCT LVL.
   A. If level is low, ensure automatic makeup is initiated per SOP-104A.
   B. If level is high, ensure 1/1-MU, RCS MU MAN ACT is off and place 1/1-LCV-112A, VCT LVL CTRL VLV in HUT to lower VCT level.
   C. When VCT level has lowered to desired level, place 1/1-LCV-112A, VCT LVL CTRL VLV in VCT.

4. Verify VCT pressure is being maintained per Chemistry's request on 1-PI-115, VCT PRESS.
   A. IF VCT pressure CANNOT be maintained, THEN dispatch an operator to SFGD 832' VCT VLV RM to verify the in service pressure regulator is set correctly.
      - 1-8155, N_2 TO VCT SPLY
      - 1-8156, H_2 TO VCT SPLY

   B. If VCT pressure is high, control VCT pressure per SOP-103A.

Original Question: CPNPP Exam Bank ILOT5838

Volume Control Tank pressure lowers and ALB-6B, Window 1.5, VCT PRESS HI/LO, alarms.

Which of the following describes the effect of the lowering pressure on Reactor Coolant Pump seal #1 leakoff flow and what procedure should be used to mitigate the malfunction?

A. Seal leakoff flow increases; ALM-0061A, ALARM PROCEDURE ALB-6B, should be used.

B. Seal leakoff flow increases; SOP-108A, REACTOR COOLANT PUMP, should be used.

C. Seal leakoff flow decreases; ALM-0061A, ALARM PROCEDURE ALB-6B, should be used.

D. Seal leakoff flow decreases; SOP-108A, REACTOR COOLANT PUMP, should be used.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 02/27/2014	Tier		2
Change: 3	Group		1
	K/A	005 G 2.2.25	
Level of Difficulty: 2	Importance Rating		4.2

Residual Heat Removal System: Emergency Procedures/Plan: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits

Proposed Question: 87

Given the following conditions:

- Unit 1 is in MODE 3 with Reactor Startup preparations in progress.
- Valve lineups are being performed and it has been determined that 1-HCV-606, RHR HX 1-01 FLO CTRL VLV has a broken instrument air supply line.

Which of the following explains the condition of the Train A Emergency Core Cooling System (ECCS) as a result of the Residual Heat Removal (RHR) System malfunction?

Train A ECCS is...

A. ...OPERABLE because 1-HCV-606 fails CLOSED on a loss of instrument air.

B. ...INOPERABLE because 1-HCV-606 fails CLOSED on a loss of instrument air.

C. ...OPERABLE because 1-HCV-606 fails OPEN on a loss of instrument air.

D. ...INOPERABLE because 1-HCV-606 fails OPEN on a loss of instrument air.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Train A ECCS is OPERABLE, however 1-HCV-606 does not fail closed which would make the flowpath inoperable.
B. Incorrect. Plausible because if 1-HCV-606 did fail closed Train A ECCS flowpath would be inoperable.
C. Correct. Train A ECCS is OPERABLE because its flowpath is capable of injecting cooled water from the RWST or containment sump into the RCS.
D. Incorrect. Plausible because 1-HCV-606 fails open, however the flowpath remains OPERABLE and this makes the train OPERABLE.

Technical Reference(s) Technical Specification LCO 3.5.2 Bases Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Residual Heat Removal System

including Technical Specifications, TRM and ODCM.

Question Source:	Bank	ILOT1877	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments / Reference: Technical Specification LCO 3.5.2 Bases	Revision: 68

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After several hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and

Comments / Reference: Technical Specification LCO 3.5.2 Bases	Revision: 68

ECCS - Operating
B 3.5.2

BASES

BACKGROUND (background)

the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines in a common header and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Throttle valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs. The throttle valves also protect the SI pumps and centrifugal charging pumps from exceeding runout flow rates.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/14	Tier		2
Change: 2	Group		1
	K/A	008 A2.09	
Level of Difficulty: 3	Importance Rating		2.8

Component Cooling Water System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Results of excessive exit temperature from the letdown cooler, including the temperature effects on ion-exchange resins

Proposed Question: 88

Given the following conditions:

- Unit 1 is in MODE 1 at 100% power.
- The following Annunciators are in alarm:
  - 1-ALB-06A, Window 1.3 – LTDN HX OUT TEMP HI.
  - 1-ALB-06A, Window 2.3 – LTDN HX NORM OUT FLO DIVERT.
- 1-TK-130, LTDN HX OUT TEMP CTRL was placed in MANUAL and output stopped at 12%.
- 1-TK-130, LTDN HX OUT TEMP CTRL output CANNOT be adjusted further.
- Alarm Response Procedure field actions have not reduced Letdown Heat Exchanger outlet temperature.
- 1-FI-132, LTDN FLO indicates 130 gpm.
- 1-FI-121A, CHRG FLO indicates 142 gpm.

Which of the following are the required actions for this condition and appropriate procedure entry?

Ensure 1/1-TCV-129, LTDN DIVERT VLV, is diverted to the VCT since Letdown Heat Exchanger outlet temperature has risen to greater than or equal to…

A. ...135°F. Isolate Letdown to stop boration and enter ABN-105, Chemical and Volume Control System Malfunctions.

B. ...155°F. Isolate Letdown to stop boration and enter ABN-105, Chemical and Volume Control System Malfunctions.

C. ...135°F. Isolate Letdown to stop dilution and enter ABN-502, Component Cooling Water System Malfunctions.

D. ...155°F. Isolate Letdown to stop dilution and enter ABN-502, Component Cooling Water System Malfunctions.

Proposed Answer: A

Explanation:

A. Correct. 1/1-TCV-129 automatically diverts to the VCT when 1-TI-130, LTDN HX OUT TEMP reaches ≥ 135°F. ALM-0061A, Window 1.3 – LTDN HX OUT TEMP HI, directs isolating Letdown which stops any boration in progress and then directs entry into ABN-105 for loss of Letdown.

B. Incorrect. Plausible because 155°F is the setpoint for automatic divert on BTRS inlet temperature, however, 1/1-TCV-129 automatically diverts to the VCT when 1-TI-130, LTDN HX OUT TEMP reaches ≥ 135°F. Additionally, ALM-0061A, Window 1.3 – LTDN HX OUT TEMP HI, directs isolating Letdown which stops any boration in progress and then directs entry into ABN-105 for loss of Letdown.

C. Incorrect. Plausible because 1/1-TCV-129 automatically diverts to the VCT when 1-TI-130 reaches ≥ 135°F. It could be thought that warmer Letdown would lead to dilution but it leads to boration and it could be thought that ABN-502 would correct the temperature of the Letdown stream.

D. Incorrect. Plausible because the procedural action is correct, however, 155°F is the setpoint for automatic divert on BTRS inlet temperature. 1/1-TCV-129 automatically diverts to the VCT when 1-TI-130 reaches ≥ 135°F. It could be thought that warmer Letdown would lead to dilution but it leads to boration and it could be thought that ABN-502 would correct the temperature of the Letdown stream.

Technical Reference(s) ALM-0061A, 1-ALB-6A, Window 1.3 Attached w/ Revision: See Comments / Reference
LO21.SYS.CS1, Pages 61 & 62

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrumentation while responding to a Component Cooling Water System malfunction.

Question Source: Bank ILOT7379
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: ALM-0061A, 1-ALB-6A, Window 1.3	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 11 OF 79

ANNUNCIATOR NOM./NO.: **LTDN HX OUT TEMP HI** **1.3**

PROBABLE CAUSE:

Inadequate charging flow
CCW system malfunction
1-TK-130 LTDN HX OUT TEMP CTRL malfunction
1-TCV-4646 LTDN HX OUT TEMP CTRL VLV malfunction
Excessive letdown flow

NOTE: 1-TCV-4646 fails open on loss of air or loss of power to Process Control Cabinet 08.

AUTOMATIC ACTIONS: None

NOTE: 1/1-TCV-129, LTDN DIVERT VLV diverts flow to the VCT if letdown temperature is >135°F or BTRS demineralizer inlet temperature is >155°F.

OPERATOR ACTIONS:

1. Monitor 1-TI-130, LTDN HX OUT TEMP.

   A. If temperature increases to ≥135°F, ensure 1/1-TCV-129, LTDN DIVERT VLV is diverted to the VCT.

Comments / Reference: ALM-0061A, 1-ALB-6A, Window 1.3	Revision: 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 13 OF 79

ANNUNCIATOR NOM./NO.: **LTDN HX OUT TEMP HI** **1.3**

OPERATOR ACTIONS: (CONTINUED)

6. Ensure 1-TI-130, LTDN HX OUT TEMP is maintained <125°F.

   A. If temperature CANNOT be maintained, isolate letdown and refer to ABN-105 for Loss of Letdown.

      - 1/1-8149A, LTDN ORIFICE ISOL VLV (45 GPM)
      - 1/1-8149B, LTDN ORIFICE ISOL VLV (75 GPM)
      - 1/1-8149C, LTDN ORIFICE ISOL VLV (75 GPM)

Comments / Reference: LO21.SYS.CS1, Pages 61 & 62	Revision: 04/18/11

Boric acid is a weak acid which occurs in varied ionic forms. The most important forms are the monoborate [$B(OH)_3$] and triborate [$B_3(OH)_{10}$] ions. The existence of these forms is affected by temperature, initial overall boron concentration and, to a small extent, pH. When a resin bed is initially saturated, either a mono-borate or tri-borate ion can take the place of an OH^- resin site.

The relative concentration of mono and tri-borate ions in solution changes as letdown temperature is changed. If letdown temperature decreases, the boric acid in solution tends to form more of the tri-borate ions. When these ions approach an anion resin site holding a mono-borate ion, the two ions exchange since the tri-borate ions are at a higher concentration. This results in a dilution of boron from the reactor coolant system. Similarly, if letdown temperature increases, more mono-borate ions are formed and the resin will release tri-borate ions in exchange for mono-borate ions, resulting in a boration of the reactor coolant system.

The resulting effect from raising letdown temperature is easily remembered by recognizing that it has the same reactivity effect on the core as raising the temperature of the reactor coolant system when operating at full reactor power. If coolant temperature is raised, the change in temperature adds negative reactivity to the core. Likewise, if letdown temperature is raised, the resulting change in boron concentration out of the letdown mixed bed ion exchanger will add negative reactivity to the core.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		2
Change: 3	Group		1
	K/A	026 A2.05	
Level of Difficulty: 2	Importance Rating		4.1

Containment Spray System: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of chemical addition tanks to inject

Proposed Question: 89

Given the following conditions:

- Unit 1 is responding to a Large Break Loss of Coolant Accident (LOCA) in accordance with EOP-0.0A, Reactor Trip or Safety Injection.
- Containment pressure is 35 psig and lowering.
- While performing Step 7 of EOP-0.0A the following is observed:
  - 1-HS-4754, CHEM ADD TK DISCH VLV is CLOSED.
  - 1-HS-4755, CHEM ADD TK DISCH VLV is CLOSED.
  - 1-HS-4752, CHEM ADD TK DISCH VLV is OPEN.
  - 1-HS-4753, CHEM ADD TK DISCH VLV is OPEN.
  - Containment Spray Chemical Additive Tank level is 93% and stable.

Which of the following describes the expected action at EOP-0.0A, Reactor Trip or Safety Injection, Step 7 and the Technical Specification Bases for this action?

A. Manually OPEN 1-HS-4754 and 1-HS-4755.
Make spray more acidic to enhance iodine absorption and retention.

B. Manually OPEN 1-HS-4754 and 1-HS-4755.
Make spray more alkaline to enhance iodine absorption and retention.

C. Locally OPEN 1-LV-4754 and 1-LV-4755.
Make spray more acidic to enhance iodine absorption and retention.

D. Locally OPEN 1-LV-4754 and 1-LV-4755.
Make spray more alkaline to enhance iodine absorption and retention.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the valves should be manually opened however, NaOH will make spray more alkaline not more acidic.
B. Correct. EOP-0.0A, Step 7 RNO says to manually align valves and refer to Attachment 6 as necessary. TS bases for LCO 3.6.7, Spray Additive System states that making the spray more alkaline will enhance iodine absorption and retention.
C. Incorrect. Plausible because the valves should be opened but EOP-0.0A, Step 7 states to manually align valves not locally open valves and adding NaOH will make spray more alkaline not more acidic.
D. Incorrect. Plausible because the valves should be opened but EOP-0.0A, Step 7 states to manually align valves not locally open valves and adding NaOH will make spray more alkaline.

Technical Reference(s): EOP-0.0A, Step 7 & Attachment 6; Technical Specification LCO 3.6.7 Bases — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Containment Spray System.

APPLY the administrative requirements of the Containment Spray System including Technical Specifications, TRM and ODCM.

Question Source:

Bank		
Modified Bank		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	
55.43	2

Comments / Reference: EOP-0.0A, Step 7	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 6 OF 117

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 7	Verify Containment Spray Not Required:	
	a. Containment pressure - HAS REMAINED LESS THAN 18.0 PSIG • 1-ALB-2B window 1-8, CS ACT - NOT ILLUMINATED -AND- • 1-ALB-2B window 4-11, CNTMT ISOL PHASE B ACT - NOT ILLUMINATED -AND- • Containment Pressure - LESS THAN 18.0 PSIG	a. Perform the following: 1) Verify Containment Spray AND Phase B Actuation initiated. IF NOT, THEN manually actuate. 2) Verify appropriate MLB indication for CNTMT SPRAY (BLUE WINDOWS) AND PHASE B (ORANGE WINDOWS). IF valves NOT aligned, THEN manually align valve(s) as appropriate. (Refer to Attachment 6 as necessary). 3) Verify containment spray flow. 4) Ensure CHEM ADD TK DISCH VLVs - OPEN • 1-HS-4752 • 1-HS-4753

Comments / Reference: EOP-0.0A, Attachment 6	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 40 OF 117

ATTACHMENT 6
PAGE 1 OF 2

CONTAINMENT SPRAY/PHASE B ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	POSITION	ESFAS TRAIN	MLB LOCATION
☐ CB-02	1-HS-4754	CHEM ADD TK DISCH VLV	OPEN	A	1-MLB-4A3/1.9
☐ CB-02	1-HS-4776	CS HX 1 OUT VLV	OPEN	A	1-MLB-4A3/2.7
☐ CB-02	1-HS-4772-1	CSP 1 RECIRC VLV	CLOSED	A	1-MLB-4A3/1.6
☐ CB-02	1-HS-4772-2	CSP 3 RECIRC VLV	CLOSED	A	1-MLB-4A3/2.6
☐ CB-02	1-HS-4755	CHEM ADD TK DISCH VLV	OPEN	B	1-MLB-4B3/1.9

Comments / Reference: Technical Specification LCO 3.6.7 Bases	Revision: 68

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. When NaOH is added to the spray, a pH value of greater than or equal to 7.1 of the solution recirculated from the containment sump is ensured. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to two containment spray pumps and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The spray additive system, including the eductors, is designed to ensure the contents of the Chemical Additive Tank is injected into containment given any single active failure. Consequently, in the short term, the pH of a train of spray can vary from acidic (pH of approximately 4.5) to strong basic. The low spray pH can only occur during injection prior to switchover to recirculation. The equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7 and adequate spray pH for long term iodine retention is achieved with the onset of the spray recirculation mode. The high spray pH can only occur after switchover to recirculation from the sump when spray additive is added to recirculated sump water.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		2
Change: 2	Group		1
	K/A	061 G 2.4.18	
Level of Difficulty: 4	Importance Rating		4.0

Auxiliary/Emergency Feedwater System: Emergency Procedures/Plan: Knowledge of the specific bases for EOPs

Proposed Question: 90

Given the following conditions:

- Unit 2 is responding to a Pressurized Thermal Shock condition in accordance with FRP-0.1B, Response to Imminent Pressurized Thermal Shock Condition.
- While checking Reactor Coolant System (RCS) Cold Leg temperatures, all four loops indicate 170°F and lowering.
- All four Steam Generators (SG) are faulted.

Which of the following describes the expected action with regard to Auxiliary Feedwater (AFW) flow and bases for the action in accordance with FRP-0.1B?

A. Control AFW flow at 100 gpm to each SG.
Minimize the effects of the RCS cooldown due to secondary depressurization.

B. Control AFW flow at 100 gpm to each SG.
Minimize thermal shock to Steam Generator components due to dryout and subsequent feeding.

C. Control total AFW flow at a minimum of 460 gpm.
Minimize the effects of the RCS cooldown due to secondary depressurization.

D. Control total AFW flow at a minimum of 460 gpm.
Minimize thermal shock to Steam Generator components due to dryout and subsequent feeding.

Proposed Answer: A

Explanation:

A. Correct. AFW flow is throttled to 100 gpm per SG to minimize the effects of RCS cooldown while the secondary depressurizes.

B. Incorrect. Plausible because 100 gpm is correct flow but SG dryout concern is from ECA-2.1B and is used to control thermal shock to the SG not the reactor vessel.

C. Incorrect. Plausible because it could be thought that 460 gpm is required for secondary heat sink in FRP-0.1B and that 100 gpm is only applicable to ECA-2.1B. Also, the reason for the action is correct.

D. Incorrect. Plausible because it could be thought that 460 gpm is required for secondary heat sink in FRP-0.1B and that 100 gpm is only applicable to ECA-2.1B. Also it could be thought that SG dryout is concern in FRP-0.1B based on thermal shock but ECA-2.1B addresses SG thermal shock not reactor vessel thermal shock.

Technical Reference(s): FRP-0.1B, Attachment 4, Step 2 Bases; ECA-2.1B, Attachment 4, Step 2 CAUTION — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the operator actions, including all cautions, notes, RNOs, and bases associated with FRP-0.1, Response to Imminent Pressurized Thermal Shock Condition.

Question Source:
Bank ____
Modified Bank ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41 ____
55.43 5

Comments / Reference: FRP-0.1B, Attachment 4, Step 2 Bases	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRP-0.1B
RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	REVISION NO. 8	PAGE 32 OF 53

ATTACHMENT 4
PAGE 2 OF 23

BASES

CAUTION: If the turbine-driven AFW pump is the only operable source of feed flow to the steam generators (e.g., MD AFW pumps are not capable of providing feed flow to the SGs), then isolation of its steam supply line may degrade system conditions and result in a transition to FRH-0.1B. Therefore, this isolation must not be performed.

STEP 2: Cold leg temperature is the best available indication of vessel downcomer temperature. It is important to terminate, if possible, any cooldown in progress to limit the extent of possible vessel damage due to excessive thermal stresses.

The action to verify Main Steam isolation valve position is intended to include the actions to verify the Main Steam isolation bypass valves closed, in the event the bypass valves have been opened during startup operation (e.g., Main Steamline warmup).

If the RCS cold leg temperatures are decreasing the operator is instructed to eliminate any secondary-side or RHR System instigated RCS cooldown. The items checked in this step are in a preferred order such that the most probable causes of the cooldown are checked first. Therefore, any valves that dump steam are verified to be closed. Next, any cooldown from the RHR System is terminated. A cooldown caused by overfeeding the intact SGs is stopped by controlling AFW flow consistent with minimum secondary heat sink requirements. The operator checks for any faulted SGs and isolates them. Finally, if a faulted SG is necessary for RCS temperature control or if all SGs are faulted, AFW flow to those SGs is controlled at a minimum measurable value to minimize the effects of the RCS cooldown due to the secondary side depressurization.

Comments / Reference: ECA-2.1B, Attachment 4, Step 2 CAUTION	Revision: 8

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-2.1B
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 43 OF 72

ATTACHMENT 4
PAGE 2 OF 31

BASES

CAUTION: If AFW flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable AFW flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

NOTE: This note advises the operator to monitor RCS boron concentration to verify adequate shutdown margin during the cooldown to cold shutdown. Note that since ECCS was in service, RCS boron concentration is expected to be sufficient.

Periodic samples should be taken to monitor shutdown margin, however the operator should not wait for the sample results.

STEP 2: Depending upon the size of the effective break areas for the steam generators, the cooldown rate experienced after reactor trip could exceed 100°F/hr. A reduction of AFW flow to the steam generators has three primary effects:

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		2
Change: 2	Group		2
	K/A	015 A2.01	
Level of Difficulty: 3	Importance Rating		3.9

Nuclear Instrumentation System: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation

Proposed Question: 91

Given the following conditions:

- Unit 1 is operating at 100% power when the following alarms are received:
  - 1-ALB-5C, Window 2.5 – 1 OF 4 OT N16 HI.
  - 1-ALB-6D, Window 1.4 – RX > 50% PWR UP DET FLUX DEV HI.
  - 1-ALB-6D, Window 3.4 – PR CHAN DEV.
  - 1-ALB-6D, Window 3.14 – 1 OF 4 OT N16 ROD STOP & TURB RUNBACK.
  - 1-ALB-6D, Window 4.10 – QUADRANT PWR TILT.
- Rod Control is in AUTOMATIC.
- Control Bank D rods remain at 215 steps.
- ABN-703, Power Range Instrumentation Malfunction is in progress.

Which of the following describes the cause of the alarms and the expected procedural response?

A Power Range nuclear instrument channel...

A. ...upper detector has failed low.
Place control rods in MANUAL and withdraw rods to restore T_{AVE} - T_{REF} deviation.

B. ...upper detector has failed low.
Verify QPTR is within limit in 12 hours.

C. ...upper detector has failed high.
Place control rods in MANUAL and withdraw rods to restore T_{AVE}-T_{REF} deviation.

D. ...upper detector has failed high.
Verify QPTR is within limit in 12 hours.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the upper detector has failed low based on control bank D rod position with rod control in automatic. Rods are placed in MANUAL but not to restore T_{AVE}-T_{REF} deviation, there is no deviation because rods do not move when the instrument fails low.

B. Correct. With Control Bank D rods at 215 steps with rod control in automatic indicates the upper detector failed low. Rods are placed in MANUAL but not to restore T_{AVE}-T_{REF} deviation. QPTR surveillance is required 12 hours after the detector failure when above 75% power.

C. Incorrect. Plausible because placing rods in MANUAL and restoring T_{AVE}-T_{REF} deviation would be required if the detector failed high however based on control rod response the detector has failed low.

D. Incorrect. Plausible because verifying QPTR in 12 hours is required but the detector has failed low not high based on control rod response.

Technical Reference(s): ABN-703, Section 2.1 & 2.2; ABN-703, Steps 2.3.1, 2.3.2, & 2.3.7; Technical Specification LCO 3.2.4; OPT-302, Step 5.2.3

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Power Range Instrumentation Malfunction in accordance with ABN-703, Power Range Nuclear Instrument Malfunction.

Question Source:
Bank ______
Modified Bank ILOT8203 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ______
55.43 2

Comments / Reference: ABN-703, Section 2.1	Revision: 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 3 OF 23

2.0 POWER RANGE INSTRUMENTATION MALFUNCTION

2.1 Symptoms

a. Annunciator Alarms

- 1 OF 4 OT N-16 HI (5C-2.5)
- 1 OF 4 HI SETPT PR FLUX HI (6D-1.3)
- 1 OF 4 LO SETPT PR FLUX HI (6D-2.3)
- 1 OF 4 PR FLUX RATE HI (6D-3.3)
- PR HI VOLT FAIL (6D-4.3)
- RX ≥50% PWR UP PR DET FLUX DEV HI (6D-1.4)
- RX ≥50% PWR LOW PR DET FLUX DEV HI (6D-2.4)
- PR CHAN DEV (6D-3.4)
- QUADRANT PWR TILT (6D-4.10)
- OP HI FLUX ROD STOP C-2 (6D-2.14)
- 1 OF 4 OT N-16 ROD STOP & TURB RUNBACK (6D-3.14)

Comments / Reference: ABN-703, Section 2.2	Revision: 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 4 OF 23

2.1 b.

- Upscale, downscale, or erratic indication of the PERCENT FULL POWER or the upper or lower MICROAMPERES DETECTOR CURRENT meters on the nuclear instrumentation cabinet drawers for the failed channel.
- Lighting of the POSITIVE RATE TRIP lights on the nuclear instrumentation cabinet drawer for the failed channel, if the failure caused a rate of change of greater than or equal to 5% within 2 seconds.
- Lighting of the CHANNEL DEVIATION light on the comparator and rate drawer.

2.2 Automatic Actions

NOTE:	The power range channels are designed with coincidence requirements for operational reliability. For that reason, an individual channel failure will cause an annunciator alarm and the OP HI FLUX ROD STOP C-2 with 1/4 channels at 103% of full power. No other safety system actuations will occur due to coincidence requirements.

- IF a power range channel fails HIGH while the rod control system is in automatic, THEN control rods will be rapidly inserted.
- A power range channel failure LOW will cause no control response.

Comments / Reference: ABN-703, Step 2.3.1 & 2.3.2	Revision: 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 5 OF 23

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify rapid control rod insertion - NOT REQUIRED a. Reactor and Turbine Power - MATCHED -AND- Tave less than 3°F above Tref. b. Place Rod Control in MANUAL	Perform the following: 1. Monitor rod motion AND Tave. 2. Ensure Tave restored to programmed temperature. 3. Investigate cause of system upset. 4. IF NO instrument failure/malfunction is indicated, THEN return to procedure and step in effect.
☐ 2 Verify Reactor Power LESS THAN 75% rated thermal power (RTP).	Initiate actions to comply with Technical Specification SR 3.2.4.2.

Comments / Reference: ABN-703, Step 2.3.7	Revision: 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 8 OF 23

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION: QUADRANT POWER TILT alarms (u-ALB-6D, 4.10) should be considered inoperable when any Power range channel is inoperable.	
7 Check Quadrant Power Tilt Ratio within limits:	
☐ a. Check power range channels - ONE OR MORE INOPERABLE	a. Check QPTR alarm status. IF u-ALB-6D, 4.10 - LIT, THEN GO TO step 6b. IF u-ALB-6D, 4.10 - DARK, THEN GO TO step 8.
☐ b. Check Reactor Power - GREATER THAN 50%	b. Monitor Reactor Power. IF Reactor Power is raised above 50%, THEN perform step 7c.
☐ c. Refer to TS 3.2.4 AND Table 3.3.1-1, items 2, 3 (actions D and E).	

Comments / Reference: Technical Specification LCO 3.2.4 | Amendment: 161

QPTR
3.2.4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----NOTES----- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program.
SR 3.2.4.2	-----NOTE----- Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. ---------- Verify QPTR is within limit using the core power distribution measurement information.	In accordance with the Surveillance Frequency Control Program.

Comments / Reference: OPT-302, Step 5.2.3	Revision: 11

CPNPP OPERATIONS TESTING MANUAL	UNIT COMMON	PROCEDURE NO. OPT 302
CALCULATING POWER TILT RATIO	REVISION NO.11 MULTIPLE USE	PAGE 4 OF 8

5.0 PRECAUTIONS, LIMITATIONS AND NOTES

5.1 Precautions

None

5.2 Limitations

5.2.1 This procedure is performed when the reactor is operating in MODE 1 with THERMAL POWER > 50% of RTP:

- At least once per 7 days with the QUADRANT PWR TILT alarm (u-ALB-6D, 4.10) OPERABLE (SR 3.2.4.1),

  OR

- At least once per 12 hours with the QUADRANT PWR TILT alarm inoperable. (TRS 13.2.33.1).

5.2.2 With input from one Power Range Neutron Flux channel inoperable AND THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR in accordance with this procedure.

[C] 5.2.3 With input from one or more Power Range Neutron Flux channels inoperable AND THERMAL POWER > 75% RTP, QPTR is verified within limit using the movable incore detectors per NUC-208 (SR 3.2.4.2). SR 3.2.4.2 is not required until 12 hours after the inputs from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

[C] 5.2.4 IF measured QPTR exceeds 1.02, THEN the Shift Manager shall be promptly notified of the condition AND the ACTIONS of Technical Specification LCO 3.2.4 initiated. Core Performance Engineering shall also be informed if the limits are exceeded.

Original Question: CPNPP Exam Bank ILOT8203

Given the following conditions with Unit 1 operating at 100% power:

The following annunciators are in alarm:
1-ALB-6D-2.4, RX > 50% PWR LOW PR DET FLUX DEV HI
1-ALB-6D-3.4, PR CHAN DEV
1-ALB-6D-4.10, QUADRANT PWR TILT
Rod Control is in AUTOMATIC.
NO other alarms or automatic control actions occurred.
ABN-703, Power Range Instrumentation Malfunction is in progress.

Which of the following describes the cause of the alarms and what action should be taken to mitigate the situation?

A. A Power Range NI Lower Detector has failed low.
Direct a power reduction to < 75% RTP due to QPTR being greater than Technical Specification limit.

B. A Power Range NI Lower Detector has failed high.
Perform the required channel bypasses that will allow the remaining channels to calculate QPTR.

C. A Power Range NI Lower Detector has failed low.
Verify QPTR within limits using the Core Power Distribution Measurement every 12 hours.

D. A Power Range NI Lower Detector has failed high.
Place Rod Control in MANUAL until the channel is restored.

Answer: C

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		2
Change: 4	Group		2
	K/A	035 A2.01	
Level of Difficulty: 3	Importance Rating		4.6

Steam Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the SGS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or ruptured steam generators

Proposed Question: 92

Given the following conditions:

- A large steam break occurred inside Unit 2 Containment.
- During the performance of EOP-0.0B, Reactor Trip or Safety Injection, Containment pressure rose to 19 psig.
- Proper Containment Spray alignment was verified.
- EOP-0.0B, Attachment 2, Safety Injection Actuation Alignment has been completed.
- Transition to EOP-2.0B, Faulted Steam Generator Isolation from EOP-0.0B is complete.
- Containment pressure is 23 psig.

Applying the Emergency Response Guidelines “rules of usage” which of the following identifies the procedure in effect and why?

A. Remain in EOP-2.0B, Faulted Steam Generator Isolation even though proper response for Containment Spray actuation has NOT been verified.

B. Enter FRZ-0.1B, Response to High Containment Pressure because proper response for Containment Spray actuation has NOT been verified.

C. Remain in EOP-2.0B, Faulted Steam Generator Isolation because proper response for Containment Spray actuation has been verified.

D. Enter FRZ-0.1B, Response to High Containment Pressure even though proper response for Containment Spray actuation has been verified.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it may be thought that because containment spray actuation was verified in EOP-0.0B that entry into FRZ-0.1B is not required. Because ORANGE path on CNTMT CSFST still exists it may be thought that containment spray actuation was not verified.

B. Incorrect. Plausible because entry into FRZ-0.1B is required with an ORANGE CSFST however containment spray actuation was verified in EOP-0.0B.

C. Incorrect. Plausible because it may be thought that because containment spray actuation was verified in EOP-0.0B that entry into FRZ-0.1B is not required however because the ORANGE CNTMT CSFST exists FRZ-0.1B must be re-entered.

D. Correct. Even though containment spray actuation was verified in EOP-0.0B the CNTMT CSFST is ORANGE and FRZ-0.1B performance is required.

Technical Reference(s): ODA-407, Attachment 8.A, Page 8 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a set of plant conditions (or an actual or simulated Control Room status) and a set of Critical Safety Function Status Trees, correctly **DETERMINE** the status of the Critical Safety functions and **IDENTIFY** any applicable Functional Restoration Guidelines.

Question Source:
Bank
Modified Bank ILOT5978 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41
55.43 5

Comments / Reference: ODA-407, Attachment 8.A, Page 8	Revision: 15

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 15 INFORMATION USE	PAGE 26 OF 56

ATTACHMENT 8.A
PAGE 8 OF 22

ERG RULES OF USAGE

Scenarios Affecting FRZ-0.1A/B Status	Requirements for Implementing FRZ-0.1A/B
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition HAS CLEARED when FRG implementation is initiated.	IF FRZ ORANGE condition has CLEARED when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN performance of FRZ-0.1A/B is NOT required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions AND there is NOT currently a challenge to the Containment barrier.
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition STILL EXISTS when FRG implementation is initiated.	IF an FRZ ORANGE condition exists when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN FRZ-0.1A/B performance is required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions BUT a challenge to the Containment barrier may exist.
EOP-0.0A/B is performed without Containment Spray actuation (Step 7 of EOP-0.0A/B identifies Containment Spray NOT required/NOT verified). The appropriate recovery actions are initiated with transition out of EOP-0.0A/B. The FRZ ORANGE condition COMES IN AND remains in during implementation of recovery actions (after FRG implementation initiated).	All CSFSTs are monitored and FRGs are implemented per the rules of usage. FRG implementation is initiated based on the highest CSF priority. IF an FRZ ORANGE condition exists, THEN FRZ-0.1A/B performance is required. A challenge to the Containment barrier exists AND proper response for Containment Spray actuation is verified to minimize challenges to the Containment barrier.
EOP-0.0A/B is performed without Containment Spray actuation (Step 7 of EOP-0.0A/B identifies Containment Spray NOT required/NOT verified). The appropriate recovery actions are initiated with transition out of EOP-0.0A/B. The FRZ ORANGE condition COMES IN after FRG implementation has been initiated, THEN clears prior to entering FRZ-0.1A/B.	All CSFSTs are monitored and FRGs are implemented per the rules of usage. FRG implementation is initiated based on the highest CSF priority. IF an FRZ ORANGE condition has previously existed AND FRZ-0.1A/B has NOT been performed, THEN FRZ-0.1A/B performance is required. Proper response for Containment Spray actuation is verified to ensure challenges to the Containment barrier have been addressed.

Original Question: CPNPP Exam Bank ILOT5978
Given the following conditions on Unit 2: • A large steam break has occurred inside Containment. • During the performance of EOP-0.0B, Reactor Trip or Safety Injection, Containment pressure rose to 19 psig. • Proper operation of Containment Spray System was verified. • EOP-0.0B, Attachment 2, Safety Injection Actuation Alignment has been completed. • A transition has just been made to EOP-2.0B, Faulted Steam Generator Isolation. • Containment pressure is now 22 psig. Which of the following identifies the status of the Containment Critical Safety Function and what action should be taken to mitigate the situation? A. Critical Safety Function CONTAINMENT Status Tree is ORANGE. Continue to monitor Containment pressure and transition to FRZ-0.1B, Response to High Containment Pressure if it exceeds 50 psig. B. Critical Safety Function CONTAINMENT Status Tree is RED. Continue to monitor Containment pressure and transition to FRZ-0.1B, Response to High Containment Pressure if it remains above 18 psig for more than 1 hour. C. Critical Safety Function CONTAINMENT Status Tree is RED. Transition to FRZ-0.1B, Response to High Containment Pressure to allow verification of proper operation of the Containment Phase B Isolation valves. D. Critical Safety Function CONTAINMENT Status Tree is ORANGE. Transition to FRZ-0.1B, Response to High Containment Pressure and then transition back to EOP-2.0B, Faulted Steam Generator Isolation. Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		2
Change: 3	Group		2
	K/A	086 G 2.4.11	
Level of Difficulty: 3	Importance Rating		4.2

Fire Protection System: Emergency Procedures/Plan: Knowledge of abnormal condition procedures

Proposed Question: 93

Given the following conditions:

- Both Units are at 100% power.
- A fire is burning in the Unit 1 Cable Spreading Room.
- Both Units are shutting down.
- The Shift Manager is injured and CANNOT perform his duties.

Who will assume the duties of the Shift Manager and which Unit will control operation of systems and equipment common to both Units?

In accordance with ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room the duties of the Shift Manager will be assumed by the...

A. ...Unit 1 Unit Supervisor, Unit 1 will control common systems/equipment.

B. ...Unit 1 Unit Supervisor, Unit 2 will control common systems/equipment.

C. ...Unit 2 Unit Supervisor, Unit 1 will control common systems/equipment.

D. ...Unit 2 Unit Supervisor, Unit 2 will control common systems/equipment.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that the Unit 1 US would assume SM duties. Unit 1 will control common systems and equipment.

B. Incorrect. Plausible because it could be thought that the Unit 1 US would assume SM duties and then Unit 2 would control common systems and equipment.

C. Correct. ABN-803A directs the Unit 2 or CPC SRO to assume SM duties and also directs Unit 1 to control common systems and equipment.

D. Incorrect. Plausible because it could be thought that because the fire is in the Unit 1 Cable Spreading Room that the Unit 2 US would assume SM duties and Unit 2 would control common systems and equipment.

Technical Reference(s) ABN-803A, Step 2.3.1 NOTE; ABN-803A, Step 2.3.6 NOTE — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Fire in the Electrical or Control Building in accordance with ABN-803, Response to A Fire In the Control Room or Cable Spreading Room.

Question Source: Bank ILOT8073
Modified Bank ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: ABN-803A, Step 2.3.1 NOTE	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 11	PAGE 6 OF 62

2.3 Operator Actions

CAUTION: Use of this procedure may result in abnormal configuration. Management review of steps performed is necessary to ensure configuration tracking and restoration.

NOTE:

- The decision to evacuate the Control Room shall be made by the Shift Manager, based upon the ability to safely control the plant from the Control Room.
- Once the decision to leave the Control Room has been made, Emergency Response Guidelines (ERGs) DO NOT apply. ERGs may be referred to, but should not be used for Reactor Trip Response.
- A fire in this area will require simultaneous shutdown of both Unit 1 and Unit 2. In this event Unit 1 will control manipulation of system(s)/equipment common to both units unless otherwise directed by the Shift Manager.
- Evaluate the necessity of donning SCBAs, if not already worn, prior to leaving Control Room.
- The symbol [R] has been located throughout this procedure where real or potential radiation hazards are positively identified. This identification technique should not preclude the worker from following good radiation work practices throughout this procedure to ensure his/her occupational exposure is maintained As Low As Reasonably Achievable (ALARA).
- Three two-way radios are maintained at the Remote Shutdown Panel for performance of this procedure.
- This procedure is written assuming minimum staffing requirements. However, should additional personnel be available, consideration should be given to supporting timely completion of Attachments 2, 3, and 4 followed by shutting down secondary plant equipment when conditions permit. IPO-009A may be referred to for general guidance on securing the secondary plant.

☐ 1. Refer to appropriate Fire Preplan Instruction.

Comments / Reference: ABN-803A, Step 2.3.6 NOTE	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 11	PAGE 7 OF 62

2.3 Operator Actions

☐ 5. Consult with Shift Manager to determine if performance of this procedure is necessary based upon the fire assessment and current plant conditions. IF the decision is made to evacuate Control Room, THEN continue with this procedure.

NOTE:

- The following steps outline the duties of various shift personnel when Control Room is evacuated. In the absence of the Shift Manager, the CPC Supervisor will assume his duties. |
- Operator notification may be by verbal, Gaitronics, OR two way radio. In the event Gaitronics is unavailable, use alternate methods such as phone, radio, personnel, as necessary to communicate.
- EPP-201 will be reviewed at the Remote Shutdown Panel to select the appropriate emergency classification.

6. Shift Manager/Unit Supervisor evacuation actions:

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		3
Change: 3	Category		1
	K/A	G 2.1.34	
Level of Difficulty: 4	Importance Rating		3.5

Conduct of Operations: Knowledge of primary and secondary plant chemistry limits

Proposed Question: 94

Given the following conditions:

- Unit 2 is at 100% power.
- At 1000 on June 16th, Chemistry reports RCS DOSE EQUIVALENT I-131 sample results for the past 4 hours:
  - 0600 – 1.85 µCi/gm
  - 0700 – 12.6 µCi/gm
  - 0800 – 53.2 µCi/gm
  - 0900 – 75.7 µCi/gm

Which of the following states the REQUIRED ACTION based on the Chemistry report?

A. Restore RCS DOSE EQUIVALENT I-131 to within limit by 0900 on June 18th or be in HOT STANDBY by 1500 on June 18th and in COLD SHUTDOWN by 2100 on June 19th.

B. Restore RCS DOSE EQUIVALENT I-131 to within limit by 1000 on June 18th or be in HOT STANDBY by 1600 on June 18th and in COLD SHUTDOWN by 2200 on June 19th.

C. Be in HOT STANDBY by 1600 on June 16th and in COLD SHUTDOWN by 2200 on June 17th.

D. Be in HOT STANDBY by 1500 on June 16th and in COLD SHUTDOWN by 2100 on June 17th.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that 48 hours are available to restore DEI-131 to within limits before required entry into MODE 3, however because at 0900 DEI-131 is greater than 60µCi/gm MODE 3 must be entered in 6 hours and MODE 5 in 36 hours.
B. Incorrect. Plausible if thought that 48 hours are available to restore DEI-131 to within limits before required entry into MODE 3, however because at 0900 DEI-131 is greater than 60µCi/gm MODE 3 must be entered in 6 hours. Also the start time for exceeding 60µCi/gm is 0900 not 1000 when reported.
C. Incorrect. Plausible because it could be thought that the clock for action starts at 1000 vice 0900.
D. Correct. Based on sample results at 0900 LCO 3.4.16 Condition C is entered and the unit must be in MODE 3 by 1500 on June 16th and in MODE 5 by 2100 on June 17th.

Technical Reference(s) Technical Specification LCO 3.4.16 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: Technical Specification LCO 3.4.16

Learning Objective: **APPLY** the administrative requirements of the Reactor Coolant System including Technical Specifications, TRM and ODCM.

Question Source: Bank ______
Modified Bank ILOT6157 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments / Reference: Technical Specification LCO 3.4.16	Amendment: 161

RCS Specific Activity
3.4.16

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------------------NOTE----------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 $\leq$ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

Comments / Reference: Technical Specification LCO 3.4.16	Amendment: 161

RCS Specific Activity
3.4.16

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

Original Question: CPNPP Exam Bank ILOT6157

Unit 1 is operating at 100% power with steady state conditions.

At 0800 on January 5th, Chemistry reports the following RCS DOSE EQUIVALENT I-131 sample results for the past 4 hours:

- 0400 0.75 microcuries/gram
- 0500 2.15 microcuries/gram
- 0600 45.6 microcuries/gram
- 0700 80.0 microcuries/gram

WHICH of the following is the action required based on the Chemistry reports?

A. Be in at least HOT STANDBY by 1100.

B. Be in at least HOT STANDBY by 1300.

C. Restore the Dose Equivalent I-131 within the limits by 0500 January 7th, or be in HOT STANDBY by 1100 on January 7th.

D. Restore the Dose Equivalent I-131 within the limits by 0700 January 7th, or be in HOT STANDBY by 1300 on January 7th.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 2/27/2014	Tier		3
Change: 2	Category		2
	K/A	G 2.2.42	
Level of Difficulty: 4	Importance Rating		4.6

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Proposed Question: 95

Given the following conditions:

- Unit 1 is at 100% power with the following indications:
  - Containment narrow range pressure is -0.25 psig.
  - Containment temperature is 112°F.
- Emergency Core Cooling System Accumulator indications are as follows:

	1-01	1-02	1-03	1-04
• Pressure (psig)	635	636	640	645
• Level (%)	60	59	60	62
• Boron (ppm)	2399	2426	2431	2416

- Refueling Water Storage Tank (RWST) indications are as follows:
  - Level is 95%.
  - Temperature is 121°F.
  - Boron concentration is 2432 ppm.

Based on the above indications what Technical Specification Limiting Condition(s) for Operation must be entered and what action must be taken?

A. LCO 3.6.4, Containment Pressure; restore within limits in 8 hours.
LCO 3.6.5, Containment Air Temperature; restore within limits in 8 hours.

B. LCO 3.5.1, ECCS Accumulators; restore Accumulator 1-01 boron concentration within limits in 72 hours.
LCO 3.6.5, Containment Air Temperature; restore within limits in 8 hours.

C. LCO 3.5.4, Refueling Water Storage Tank; restore Refueling Water Storage Tank temperature within limits in 8 hours.
LCO 3.5.4, Refueling Water Storage Tank; restore Refueling Water Storage Tank level within limits in 1 hour.

D. LCO 3.5.1, ECCS Accumulators; restore Accumulator 1-04 pressure and level within limits in 24 hours.
LCO 3.5.4, Refueling Water Storage Tank; restore Refueling Water Storage Tank temperature within limits in 8 hours.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that containment pressure of -.0.25 (alarm would be annunciated) psig would require entry into LCO 3.6.4, however pressure must be less than -0.3 psig. Also, it could be thought that containment temperature of 112°F (alarm setpoint) would require entry into LCO 3.6.5; however temperature must be greater than 120°F.
B. Incorrect. Plausible because it could be thought that boron concentration must be greater than 2400 ppm as is the case with the RWST; however boron must be greater than 2300 ppm for the accumulators. Also, it could be thought that containment temperature of 112°F (alarm setpoint) would require entry into LCO 3.6.5; however temperature must be greater than 120°F.
C. Incorrect. Plausible because in accordance with LCO 3.5.4 RWST temperature must be restored in 8 hours. Also, it could be thought that RWST level must be greater than 95% to not enter LCO 3.5.4 on level.
D. Correct. In accordance with LCO 3.5.1; 1-04 accumulator pressure and level must be restored in 24 hours. Also, in accordance with LCO 3.5.4 RWST temperature must be restored in 8 hours.

Technical Reference(s): Technical Specification LCO 3.5.1; Technical Specification LCO 3.5.4; Technical Specification LCO 3.6.4; Technical Specification LCO 3.6.5; OPT-102A-1

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Emergency Core Cooling System including Technical Specifications, TRM and ODCM.

Question Source: Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments / Reference: Technical Specification LCO 3.5.1	Amendment: 161

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to 1000 ≤ psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

Comments / Reference: Technical Specification LCO 3.5.4	Amendment: 161

RWST
3.5.4

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: Technical Specification LCO 3.6.4	Amendment: 161

Containment Pressure
3.6.4

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ - 0.3 psig and ≤ + 1.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: Technical Specification LCO 3.6.5	Amendment: 161

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be $\leq$ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: OPT-102A-1	Revision: 37

MODE 1 AND 2 SHIFTLY SURVEILLANCES

PARAMETER DESCRIPTION	CHANNEL	READINGS		ACCEPTANCE CRITERIA
		DAY	MID	
SSI TEMPERATURE SR 3.7.9.2	1-TR-4260-1		°F	≤102 °F
	1-TR-4260-2		°F	≤102 °F
SSI LEVEL SR 3.7.9.1 ≥776 FT requires TRM 13.7.34 entry	X-LI-4288		FT	≥770 FT
	X-LI-4289		FT	≥770 FT
ECCS ACCUMULATOR ISOLATION VALVES SR 3.5.1.1 SR 3.5.1.5	1-MLB-1A2, 1.8	SAT UNSAT	SAT UNSAT	VALVE OPEN (MLB DARK) AND VALVE MOTOR BREAKER OFF (NO DEVIATIONS ON OWI-103-1202)
	1-MLB-1B2, 1.8	SAT UNSAT	SAT UNSAT	
	1-MLB-1A2, 2.8	SAT UNSAT	SAT UNSAT	
	1-MLB-1B2, 2.8	SAT UNSAT	SAT UNSAT	
ECCS VALVE ALIGNMENT SR 3.5.2.1	1/1-8802A 69/1-8802A	SAT UNSAT	SAT UNSAT	VALVE CLOSED AND POWER SWITCH OFF
	1/1-8802B 69/1-8802B	SAT UNSAT	SAT UNSAT	VALVE CLOSED AND POWER SWITCH OFF
	1/1-8835 69/1-8835	SAT UNSAT	SAT UNSAT	VALVE OPEN AND POWER SWITCH OFF
	1/1-8813 69/1-8813	SAT UNSAT	SAT UNSAT	VALVE OPEN AND POWER SWITCH OFF
	1/1-8806 69/1-8806	SAT UNSAT	SAT UNSAT	VALVE OPEN AND POWER SWITCH OFF
	1/1-8809A 69/1-8809A	SAT UNSAT	SAT UNSAT	VALVE OPEN AND POWER SWITCH OFF
	1/1-8809B 69/1-8809B	SAT UNSAT	SAT UNSAT	VALVE OPEN AND POWER SWITCH OFF
	1/1-8840 69/1-8840	SAT UNSAT	SAT UNSAT	VALVE CLOSED AND POWER SWITCH OFF
RWST LEVEL SR 3.3.2.1.7b	1-LI-930	%	%	LEVEL ≥ 95% AND MAXIMUM DEVIATION BETWEEN OPERABLE CHANNELS ≤4%
	1-LI-931	%	%	
	1-LI-932	%	%	
	1-LI-933	%	%	
RWST TEMPERATURE SR 3.5.4.1	1-TI-4793		°F	40°F TO 120 °F
CRDM SHROUD EXHAUST TEMPERATURE TRS 13.7.36.1 NOTE (1)	1-TI-5454	°F	°F	≤163 °F
	1-TI-5455	°F	°F	≤163 °F

Comments / Reference: OPT-102A-1	Revision: 37

MODE 1 AND 2 SHIFTLY SURVEILLANCES

PARAMETER DESCRIPTION	CHANNEL	READINGS DAY	READINGS MID	ACCEPTANCE CRITERIA
CONTAINMENT AVERAGE TEMPERATURE 3.6.5.1 TRS 13.7.36.1 NOTE (1)	1-TI-5400A	°F	°F	≤110 °F *Actual TS is 120 °F. Ref. 1-ALB-3A/4.1*
CONTAINMENT PRESSURE, IR SR 3.3.2.1.1c SR 3.3.2.1.2c SR 3.3.2.1.3b.3 SR 3.3.2.1.4c	1-PI-935	PSIG	PSIG	MAXIMUM DEVIATION BETWEEN OPERABLE CHANNELS ≤3.0 PSIG
	1-PI-937	PSIG	PSIG	
	1-PI-934	PSIG	PSIG	
	1-PI-936	PSIG	PSIG	
CONTAINMENT PRESSURE, NR SR 3.6.4.1	1-PI-5470A	PSIG	PSIG	-0.3 PSIG TO 1.3 PSIG
	1-PI-5470B	PSIG	PSIG	-0.3 PSIG TO 1.3 PSIG
REACTOR CAVITY EXHAUST TEMPERATURE (NEUTRON DETECTOR WELL EXHAUST TEMPERATURE) TRS 13.7.36.1 NOTE (1)	1-TE-5445	°F	°F	≤ 150 °F
	1-TE-5446	°F	°F	≤ 150 °F
	1-TE-5447	°F	°F	≤ 150 °F
	1-TE-5448	°F	°F	≤ 150 °F
	1-TE-5449	°F	°F	≤ 150 °F
	1-TE-5450	°F	°F	≤ 150 °F
	1-TE-5451	°F	°F	≤ 150 °F
	1-TE-5452	°F	°F	≤ 150 °F
ECCS ACCUMULATOR LEVEL SR 3.5.1.2 * Primary channel except when declared inoperable. If inoperable, circle other channel to indicate its use for the surveillance. Ref. TS BASES Accumulator level shall meet acceptance criteria, using new channel, prior to shifting to new channel. ** Computer points should be used to validate deviation prior to initiating any report and/or work document. (EVAL-2009-003699-01)	1-LI-950 * (L6950A)**	%	%	PRIMARY CHANNEL LEVEL 39% TO 61 % AND MAXIMUM DEVIATION BETWEEN OPERABLE CHANNELS ≤ 5%** IF LEVEL OUTSIDE THIS CRITERIA, VERIFY ALB-4C, 1.4, 2.4, 3.4 & 4.4 NOT LIT AND RESTORE LEVEL. A level increase of ≥12% above the reference level may require sampling within 6 hours. Ref. SR 3.5.1.4
	1-LI-951 (L6951A)**	%	%	
	1-LI-952 * (L6952A)**	%	%	
	1-LI-953 (L6953A)**	%	%	
	1-LI-954 * (L6954A)**	%	%	
	1-LI-955 (L6955A)**	%	%	
	1-LI-956 * (L6956A)**	%	%	
	1-LI-957 (L6957A)**	%	%	
ECCS ACCUMULATOR PRESSURE SR 3.5.1.3 * Primary channel except when declared inoperable. If inoperable, circle other channel to indicate its use for the surveillance. Ref. TS BASES Accumulator pressure shall meet acceptance criteria, using new channel, prior to shifting to new channel.	1-PI-960 *	PSIG	PSIG	PRIMARY CHANNEL PRESSURE 623 PSIG TO 644 PSIG AND MAXIMUM DEVIATION BETWEEN OPERABLE CHANNELS ≤ 30 PSIG
	1-PI-961	PSIG	PSIG	
	1-PI-962 *	PSIG	PSIG	
	1-PI-963	PSIG	PSIG	
	1-PI-964 *	PSIG	PSIG	
	1-PI-965	PSIG	PSIG	
	1-PI-966 *	PSIG	PSIG	
	1-PI-967	PSIG	PSIG	

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		3
Change: 3	Category		2
	K/A	G 2.2.20	
Level of Difficulty: 3	Importance Rating		3.8

Equipment Control: Knowledge of the process for managing troubleshooting activities

Proposed Question: 96

Given the following condition:

- Troubleshooting is in progress under a troubleshooting Work Order for IV1EC1/3, TRN A 118 VAC RPS/SFGD BOP INSTALLED SPARE INVERTER IV1EC1/3.

Which of the following describes the process to complete troubleshooting and repair activities on IV1EC1/3, TRN A 118 VAC RPS/SFGD BOP INSTALLED SPARE INVERTER IV1EC1/3 once it is determined that several inverter internal components must be replaced?

A. The Responsible Work Organization Supervisor should initial and date the Work Order changes and provide verbal authorization to continue the work.

B. The Responsible Work Organization Supervisor should perform a technical review on the Work Order and provide written authorization to continue the work.

C. The existing Work Order must be revised, Operations must re-impact the Work Order and the Shift Manager must authorize continuing the work.

D. The existing Work Order must be edited, Operations re-impact is NOT required and the Shift Manager need NOT authorize continuing the work.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the RWO supervisor could authorize an editorial change by initial and date of the change and verbally authorizing work to continue.
B. Incorrect. Plausible because the RWO supervisor normally performs a technical review of work orders but cannot authorize continuing work IAW requirements of STI-606.01 because revision is required.
C. Correct. STI-606-0.1 requires the work order be revised when work order intent changes and the work order must be re-impacted when troubleshooting becomes corrective. The SM must review and authorize work when a troubleshooting work order scope is expanded to include corrective actions.
D. Incorrect. Plausible because an editorial change would not require re-impact or SM authorization.

Technical Reference(s) STI-606.01, Step 6.11.18 Attached w/ Revision: See Comments / Reference
STA-202, Step 6.10

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source: Bank ILOT7240
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 3

Comments / Reference: STI-606.01, Step 6.11.18	Revision: 0

CPNPP STATION INSTRUCTION MANUAL		PROCEDURE NO. STI-606.01
WORK CONTROL PROCESS	REVISION NO. 0 INFORMATION USE	PAGE 75 OF 125

NOTE: By definition, any non-editorial change constitutes a "revision". See definition 4.45a.

[C] 6.11.18 IF it is necessary to make non-editorial changes (i.e., a revision) to the work instructions OR make additions to the ROUTE of a Work Order, THEN the RWO Supervisor should determine the necessity for revision and ensure the applicable requirements of Section 6.8 (Planning) are incorporated. [05967; 4349667]

6.11.18.1 Revisions to troubleshooting WOs shall retain the initial troubleshooting instructions and indicate the revised description, additional work instructions and testing requirements.

[C] 6.11.18.2 The revised WO shall receive at least the same level of interdiscipline and supervisory review as the original and others as affected. [04444]

A. Operations Re-Impact is required if:

- The plant MODE at work start is other than as specified on the Impact Statement.
- The current prerequisites, compensatory actions or special instructions are no longer applicable.
- Troubleshooting WO is revised to perform corrective action.
- The WO is covered by a clearance.
- Scope is added to the WO.
- A WO Route is modified.
- A WO Child is modified.

B. Re-impact is NOT required if the activity is Shop Work only.

6.11.18.3 The RWO Supervisor shall ensure that appropriate reviews and signatures are complete and initial and date page one of the WO on the discipline line of the appropriate revision column for both technical and safety review.

[C] 6.11.18.4 The Shift Manager shall review and authorize the revision when: [05967; 24975; 25081; 25182; 4949667]

Comments / Reference: STI-606-01, Step 6.11.18	Revision: 0

CPNPP STATION INSTRUCTION MANUAL		PROCEDURE NO. STI-606.01
WORK CONTROL PROCESS	REVISION NO. 0 INFORMATION USE	PAGE 76 OF 125

(6.11.18.4) E. A troubleshooting Work Order is revised to specify corrective action OR;

F. A Preventive (PM) or Surveillance (SV) Work Order is revised to perform corrective maintenance.

Comments / Reference: STA-202, Step 6.10	Revision: 36

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-202
NUCLEAR GENERATION PROCEDURE CHANGE PROCESS	REVISION NO. 36 INFORMATION USE	PAGE 35 OF 172

6.10 Editorial Changes

Editorial changes may be approved without performing an Administrative Review, Technical Review, Nuclear Safety Review, Applicability Determination, 10CFR50.59 review or 10CFR 72.48 review since any change which could impact the reviews cannot be considered as an editorial change. Editorial changes shall be limited to those changes allowed by Attachment 8.F. Administrative changes and Typographical errors are a subset of editorial changes.

6.10.1 Editorial PCN

A. Editorial PCNs are initiated per Section 6.7.

B. Editorial PCNs may be approved by:

- the individual performing the original technical review of the most recent procedure revision, OR
- the group supervisor of the individual normally responsible for the procedures' maintenance, OR
- the Approval Authority.

C. IF the working group supervisor determines that the in-process work document should be changed immediately AND confirms that the change is editorial as allowed by Attachment 8.F, THEN the working group supervisor may initiate Steps 6.10.1A and 6.10.1B (e.g., telecon with designated procedure writer, improvement suggestion submittal, ODA-207 change request, TOPS) and change the in-process procedure as follows:

1. Make a pen and ink correction to the in-process procedure.
2. Note that the change is an editorial change,
3. Initial and date the correction,
4. Allow the work to continue.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/27/2014	Tier		3
Change: 3	Category		3
	K/A	G 2.3.11	
Level of Difficulty: 4	Importance Rating		4.3

Radiation Control: Ability to control radiation releases

Proposed Question: 97

Given the following conditions:

- Unit 2 is in MODE 1 at 100% power.
- ABN-106, High Secondary Activity is in progress.
- The unit has been operating with a 50 gpd Steam Generator Tube Leak for the past 8 days.
- Chemistry notifies the control room that the Specific Activity of the Condensate system is 0.12 µCi/gm Dose Equivalent I-131.

Which of the following action is required to limit the potential radiation release in response to the Chemistry report?

A. Reduce power to 20% and trip the reactor in accordance with IPO-003B, Power Operations.
Be in MODE 4 in 12 hours in accordance with LCO 3.7.18, Secondary Specific Activity.

B. Reduce power to 20% and trip the reactor in accordance with IPO-003B, Power Operations.
Be in MODE 5 in 36 hours in accordance with LCO 3.7.18, Secondary Specific Activity.

C. Reduce power to 20% and trip the reactor in accordance with ABN-106, High Secondary Activity.
Be in MODE 4 in 12 hours in accordance with LCO 3.7.18, Secondary Specific Activity.

D. Reduce power to 20% and trip the reactor in accordance with ABN-106, High Secondary Activity.
Be in MODE 5 in 36 hours in accordance with LCO 3.7.18, Secondary Specific Activity.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because shutdown of the plant using IPO-003B will be followed such that a power ramp to 20% is performed and the reactor tripped in order to be in MODE 3 within 6 hours. It could be thought that LCO 3.7.18 only is applicable in MODES 1, 2 & 3 which makes the MODE 4 in 12 hours plausible. Also the 50 gpd tube leak does not require a plant shutdown.

B. Correct. ABN-106, Section 2 does not require a plant shutdown based on the 50 gpd tube leak; however TS LCO 3.7.18 does require the plant to be in MODE 3 in 6 hours and MODE 5 in 36 hours. To shutdown the plant IPO-003B will be followed such that a power ramp to 20% is performed and the reactor tripped in order to be in MODE 3 within 6 hours. Then the unit would be placed in MODE 5 within the next 36 hours.

C. Incorrect. Plausible because it could be thought that ABN-106 would require a plant shutdown based on the 50 gpd tube leak however the plant shutdown would be accomplished with IPO-003B and is required by LCO 3.7.18 not ABN-106. It could be thought that LCO 3.7.18 only is applicable in MODES 1, 2 & 3 which makes the MODE 4 in 12 hours plausible.

D. Incorrect. Plausible because it could be thought that ABN-106 would require a plant shutdown based on the 50 gpd tube leak however the plant shutdown would be accomplished with IPO-003B and is required by LCO 3.7.18 not ABN-106. LCO 3.7.18 requires the unit to be in MODE 3 within 6 hours and then in MODE 5 within the next 36 hours.

Technical Reference(s): ABN-106, Steps 2.3.12 & 2.3.14
Technical Specification LCO 3.7.18 & Bases

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Liquid Waste systems including Technical Specifications, TRM and ODCM.

Question Source:
Bank ______
Modified Bank ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 ______
55.43 4

Comments / Reference: ABN-106, Step 2.3.12	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 10 OF 31

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11 Verify no unplanned radioactive release to environment - IN PROGRESS: ☐ a. Monitor condenser off-gas radiation and plant vent stack radiation levels AND trend: • u-RE-2959, (COG-u82) CONDENSER OFF GAS • X-RE-5567A (B), [PVG-384 (385)] S(N) VENT STACK NOBLE GAS • X-RE-5570A (B), [PVG-084 (085)] S(N) WRGM LOW RANGE	Perform the following: 1) Refer to STA-603, to document any unplanned release if vent stack monitors show an increase. 2) Notify Radiation Protection of any off-gas or vent stack activity. 3) Notify Duty Manager [C] 4) Refer to EPP-201.
☐ 12 Notify Chemistry to sample and analyze Secondary System activity: a. Verify Specific Activity of the Secondary - WITHIN TS 3.7.18 LIMITS	a. Notify Shift Manager -AND- implement the ACTION statement of the specification. GO TO Step 14.

Comments / Reference: ABN-106, Step 2.3.14	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 11 OF 31

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 13 Verify Steam Generator Blowdown - IN OPERATION	Place Blowdown System in Operation as follows: a. Raise alarm setpoint on blowdown sampling detector u-RE-4200, (SGS-u64) BLOWDOWN SMPL as necessary to clear alarm. b. Place Blowdown in service per SOP-305A/B.
14 Monitor Unit for increased RCS leakage: ☐ a. Commence trending RCS leakage rates AND Secondary Coolant Radiation levels at 10 minute intervals (correlate reading to leak rate). WHEN stable for 1 hour (≤10% increase), THEN monitor every 2 hours. WHEN stable for 24 hours, THEN return to increased awareness frequency (shiftly) AND contact Chemistry to reset RMS alarm setpoints as necessary (no higher than correlated 75 gpd (.052 gpm) leak rate). ☐ b. Verify RCS leakage remains - STABLE	Perform the following: 1) Notify Shift Manager. 2) Notify Duty Manager. 3) Refer to TS For Limiting Conditions For Operation per Section 4.1, this procedure.

Comments / Reference: Technical Specification LCO 3.7.18	Amendment: 161

Secondary Specific Activity
3.7.18

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: Technical Specification LCO 3.7.18 Bases	Revision: 68

B 3.7 PLANT SYSTEMS

B 3.7.18 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 03/27/2014	Tier		3
Change: 3	Category		3
	K/A	G 2.3.15	
Level of Difficulty: 3	Importance Rating		3.7

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: 98

Given the following conditions:

- Both Units are in MODE 1.
- The following process radiation monitor status is observed;
  - X-RE-5895A, CR VENT N. INTK (CRV053) status is BLUE.
  - X-RE-5895B, CR VENT N. INTK (CRV054) status is GREEN.
  - X-RE-5896A, CR VENT S. INTK (CRV091) status is GREEN.
  - X-RE-5896B, CR VENT S. INTK (CRV092) status is GREEN.

Which of the following identifies the required Technical Specification actions in accordance with LCO 3.3.7, Control Room Emergency Filtration/Pressurization System (CREFS) Actuation Instrumentation?

Place affected CREFS train in emergency recirculation within...

A. ...7 days AND secure the Control Room makeup air supply fan from the North Air Intake within 7 days.

B. ...7 days OR secure the Control Room makeup air supply fan from the North Air Intake within 7 days.

C. ...30 days AND secure the Control Room makeup air supply fan from the North Air Intake within 30 days.

D. ...30 days OR secure the Control Room makeup air supply fan from the North Air Intake within 30 days.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that the OR statement from LCO 3.3.7 action A is an AND statement not an OR statement.

B. Correct. With one air intake radiation monitor inoperable, TS LCO 3.3.7, CREFS Actuation Instrumentation Condition A requires either placing the affected train of CREFS in emergency recirculation within 7 days or securing the CR makeup supply fan from the affected air intake within 7 days.

C. Incorrect. Plausible because it could be thought that the OR statement from LCO 3.3.7 action A is an AND statement not an OR statement and that the action time is 30 days like LCO 3.3.3 and LCO 3.3.4.

D. Incorrect. Plausible because it could be thought that the action time is 30 days like LCO 3.3.3 and LCO 3.3.4. The OR statement is correct the 30 days is not correct.

Technical Reference(s): Technical Specification LCO 3.3.7; Technical Specification Table 3.3.7-1; Technical Specification LCO 3.3.3; Technical Specification LCO 3.3.4

Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e., LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Control Room Ventilation System.

Question Source:
- Bank ______
- Modified Bank ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 4

Comments / Reference: Technical Specification LCO 3.3.7	Amendment: 161

CREFS Actuation Instrumentation
3.3.7

3.3 INSTRUMENTATION

3.3.7 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

LCO 3.3.7 The CREFS actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7-1

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Place the affected CREFS train(s) in emergency recirculation mode.	7 days
	<u>OR</u>	
	A.2 -----NOTE----- Applicable only to Functions 3a and 3b. ----- Secure the Control Room makeup air supply fan from the affected air intake.	7 days

Comments / Reference: Technical Specification Table 3.3.7-1	Amendment: 161

CREFS Actuation Instrumentation
3.3.7

Table 3.3.7-1 (page 1 of 1)
CREFS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1, 2, 3, 4, 5, and 6, (a)	2 trains	SR 3.3.7.6	NA
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 5, and 6, (a)	2 trains	SR 3.3.7.2	NA
3. Control Room Radiation				
a. Control Room Air North Intake	1, 2, 3, 4, 5, and 6, (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	1.4×10^{-4} µCi/ml
b. Control Room Air South Intake	1, 2, 3, 4, 5, and 6, (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	1.4×10^{-4} µCi/ml
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

Comments / Reference: Technical Specification LCO 3.3.3	Amendment: 161

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.8.	Immediately

Comments / Reference: Technical Specification LCO 3.3.4	Amendment: 161

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Functions in Table 3.3.4-1 and the required hot shutdown panel (HSP) controls shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each Function and required HSP control.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable. OR One or more required HSP controls inoperable.	A.1 Restore required Function and required HSP controls to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 03/03/2014	Tier		3
Change: 2	Category		4
	K/A	G 2.4.26	
Level of Difficulty: 3	Importance Rating		3.6

Emergency Procedures/Plan: Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage

Proposed Question: 99

Given the following conditions:

- At 1500 hours, a Nuclear Equipment Operator (NEO) designated as Nozzleman 1 on the watchbill has been notified that his wife has gone to the hospital in labor.
- The Shift Manager has decided to release the NEO to go to the hospital.

Which of the following lists the action required with regard to the Fire Brigade in accordance with ODA-102, Conduct of Operations and STA-727, Fire Brigade?

Release the NEO from the Fire Brigade and immediately...

A. ...take action such that the Nozzleman 1 position will be filled by 1700 with a qualified Nozzleman.

B. ...assign the Safe Shutdown 1 NEO as Nozzleman 1 and take action to ensure relief by 1700 with a qualified Nozzleman.

C. ...assign an NEO to proceed to the scene of the fire if one was to occur in lieu of the unavailable Nozzleman 1.

D. ...assign Hoseman 1 as Nozzleman 1 and fill the Hoseman 1 position with another Prompt Team Hoseman.

Proposed Answer: A

Explanation:

A. Correct. Fire Brigade manning may be reduced by 1 for up to 2 hours excluding shift turnover if immediate action is taken to fill the position and the position is filled within 2 hours.

B. Incorrect. Plausible because it could be thought that another shift NEO could fill the position as long as the position is filled within 2 hours, however, the safe shutdown NEOs are specifically excluded from filling Fire Brigade positions.

C. Incorrect. Plausible because in accordance with STA-727 Step 6.3.2.1 Note the task of one nozzleman to proceed to the scene of the fire to perform a preliminary estimate of the size and type of fire and report the status to the Fire Brigade Leader may be designated. However, this does not release the Shift Manager from restoring the minimum Fire Brigade composition within 2 hours.

D. Incorrect. Plausible because it could be thought the plant knowledge requirement can be relaxed to allow filling the position; however, an Operations advisor would have to be assigned to allow the use of Hoseman as a Nozzleman.

Technical Reference(s): ODA-102, Attachment 8.A; STA-727, Steps 6.1.2.1, 6.2.2 & 6.3.2.1 — Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source:
- Bank ____
- Modified Bank ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ____

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 1

Comments / Reference: ODA-102, Attachment 8.A	Revision: 26

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 26 INFORMATION USE	PAGE 37 OF 37

ATTACHMENT 8.A
PAGE 3 OF 3

[C] MINIMUM SHIFT CREW COMPOSITION
[00021, 00044, 00136, 01078, 05131, 05699, 06025, 06029, 06771, 07194, 08649, 22609, 22739, 23344]

[C] (12) A site Fire Brigade of at least five members shall be maintained onsite at all times. The Fire Brigade shall not include the SM and the four other members of the minimum Operations shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. The Fire Brigade may be less than the minimum requirements for a period of time not to exceed 2 hours to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

Comments / Reference: STA-727, Step 6.1.2.1	Revision: 5

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-727
FIRE BRIGADE	REVISION NO. 5 INFORMATION USE	PAGE 8 OF 19

6.0 INSTRUCTIONS

6.1 General: Operations Fire Brigade

[C] 6.1.1 The Comanche Peak Nuclear Power Plant (CPNPP) shall have a five person Fire Brigade available 24 hours per day. The Operations Shift Manager/Unit Supervisor shall not be a member of the Fire Brigade. The Fire Brigade shall be comprised of personnel in Operations and Plant personnel whose removal from their normal functions will not impair safe operation of the plant. [01275, 01280]

[C] 6.1.2 A sufficient number of Operations and Plant personnel shall receive Fire Brigade training to insure CPNPP has the immediate availability of a fully staffed Fire Brigade at all times. [01274]

[C] 6.1.2.1 The Fire Brigade may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty crew members provided immediate action is taken to restore the brigade composition to within the minimum requirements. This provision does not permit the brigade to be unmanned below the minimum upon shift change due to an oncoming member being late or absent. [01275]

Comments / Reference: STA-727, Step 6.2.2.2	Revision: 5

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-727
FIRE BRIGADE	REVISION NO. 5 INFORMATION USE	PAGE 9 OF 19

[C] 6.2 Organization of the Fire Brigade [01275]

6.2.1 The Operations Fire Brigade shall consist of:

6.2.1.1 One Fire Brigade Leader and Four Fire Brigade members.

[C] 6.2.2 Qualification of Fire Brigade members shall be the following:

[C] 6.2.2.1 The Fire Brigade Leader in addition to Fire Brigade Training shall have extensive knowledge of plant safety-related systems by virtue of training and experience as either a licensed operator (SRO, RO) or as a non-licensed operator (NEO) qualified in Safeguards, Auxiliary, Turbine and Perimeter Operational Activities/Watch Stations, in accordance with TRA-202, “Nuclear Equipment Operator Training.” [04778]

[C] 6.2.2.2 Nozzlemen in addition to the Fire Brigade Training shall have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressant on safe shutdown capability. Nozzlemen shall meet these requirements by completing, or by being currently enrolled in the Nuclear Equipment Operator Training Program, or hold a USNRC license (active or inactive) on CPNPP. Individuals who hold an SRO Certification may also be approved by the Shift Operations Manager on a case by case basis. [04778]

Comments / Reference: STA-727, Step 6.2.2.5	Revision: 5

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-727
FIRE BRIGADE	REVISION NO. 5 INFORMATION USE	PAGE 10 OF 19

6.2.2.3.1 No one who has had known heart disease, epilepsy, or emphysema shall be qualified for the Fire Brigade unless a physician's certificate of the employees fitness to participate in such activities is provided.

6.2.2.4 Fire Brigade Members shall complete an extensive initial Fire Brigade training program and shall also participate in recurring training. These programs include classroom instruction, hands on exercises, and drills. Specifics of this training program are discussed in depth in procedure TRA-104, "Fire Protection Training."

6.2.2.5 The requirements for Plant safety-related systems knowledge, for Fire Brigade Leader or Nozzleman, may be relaxed to allow Plant Personnel to staff these positions. The use of an Operations advisor, who is a qualified NEO or holds a license, will meet this requirement.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 03/03/2014	Tier		3
Change: 2	Category		4
	K/A	G 2.4.16	
Level of Difficulty: 2	Importance Rating		4.4

Emergency Procedures/Plan: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as operating procedures, abnormal operating procedures, and severe accident management guidelines

Proposed Question: 100

Given the following conditions:

- A Station Blackout has been in progress for several hours.
- Unit 1 is responding to the Station Blackout in accordance with ECA-0.0A, Loss of All AC Power.
- While checking Core Exit Thermocouple (CET) temperatures they are found to be 1220°F and rising.

Which of the following is the required action?

The Unit Supervisor should...

A. ...remain in ECA-0.0A, Loss of All AC Power and ensure actions required to restore power to any AC Safeguards bus are in progress.

B. ...transition to SACRG-1, Severe Accident Control Room Guideline Initial Response and verify a GENERAL EMERGENCY has been declared.

C. ...enter ABN-601, Response to a 138/345 KV System Malfunction concurrent with ECA-0.0A and restore power to any AC Safeguards bus.

D. ...transition to ECA-0.2A, Loss of All AC Power Recovery With SI Required and manually align SI valves in preparation for power restoration.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this action would continuously be performed, however, once CET temperatures exceed 1200°F entry into SACRG-1 is required.
B. Correct. IAW Step 24 of ECA-0.0A when CET temperatures exceed 1200°F entry into SACRG-1 is required.
C. Incorrect. Plausible because this action would be in progress, however, once CET temperatures exceed 1200°F entry into SACRG-1 is required.
D. Incorrect. Plausible because entry into ECA-0.2A would be required because a SI would be needed with CET temperatures greater than 1200°F, however, at least one safeguards bus is needed to enter ECA-0.2A.

Technical Reference(s) ECA-0.0A, Flowchart
ECA-0.0A, Step 24
Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the proper transitions out of ECA-0.0, Loss of All AC Power.

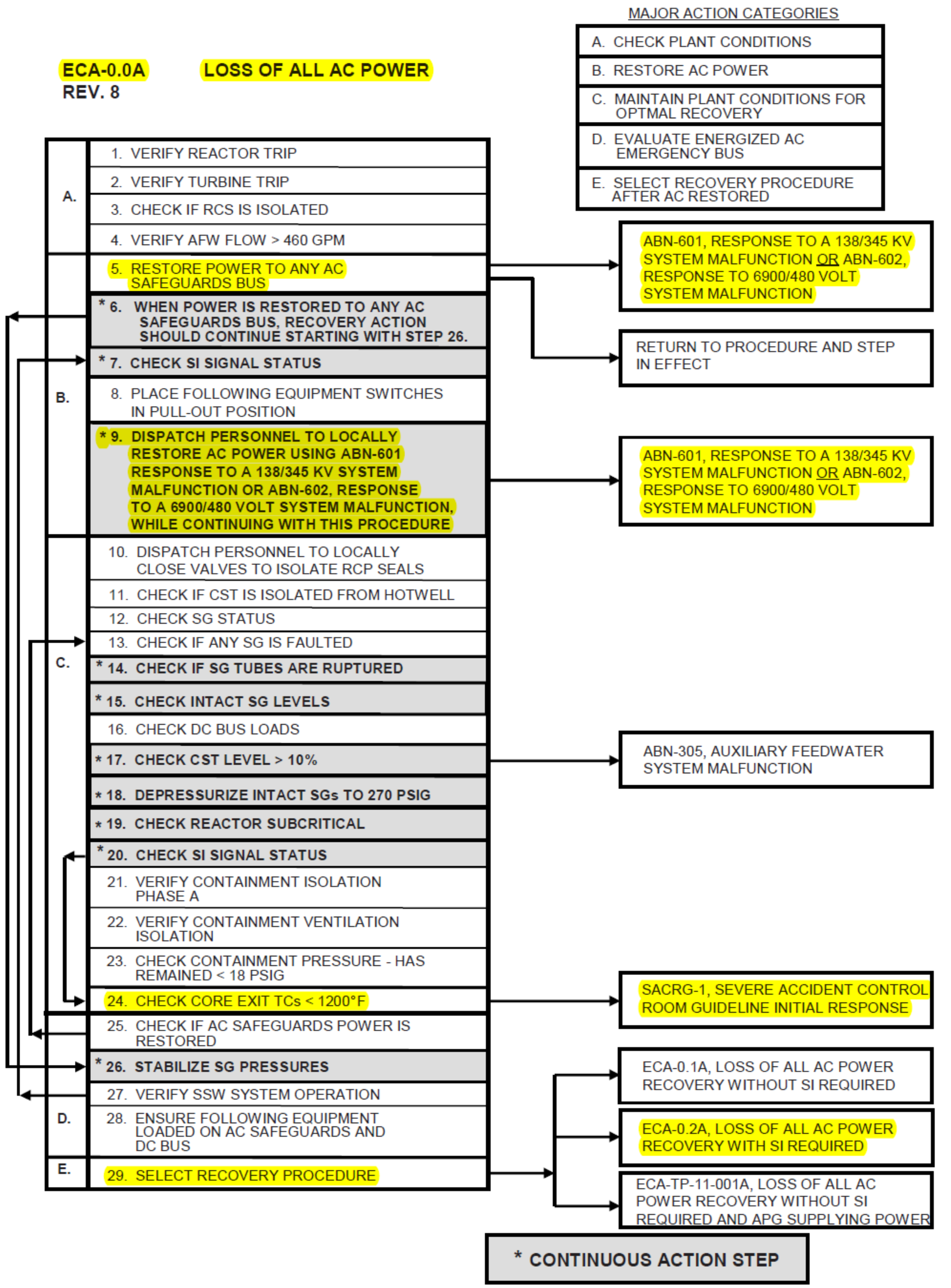
Question Source: Bank ILOT8323
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 5

Comments / Reference: ECA-0.0A, Flowchart	Revision: 8



Comments / Reference: ECA-0.0A, Step 24	Revision: 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 20 OF 88

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24	Check Core Exit TCs - LESS THAN 1200°F	IF core exit temperatures greater than 1200°F and increasing, THEN go to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE, Step 1.