76. 005AG2.1.07 001/1/2/STUCK CRDM/C/A - 4.4/4.7/BANK-HL-17/HL-18 NRC/SRO/AML

Initial conditions:

- Time = 0900.
- Unit 1 is at 60% power following a refueling outage.
- The OATC is withdrawing rods when one DRPI is seen not moving with its group.
- The OATC immediately stops withdrawing rods, and all rod motion stops.
- CBD, Group 2, Rod H-8 DRPI indicates 198 steps.
- CBD, Group 2, step counters indicate 209 steps.

Current conditions:

- Time = 0945.
- No rod motion has occurred since 0900.
- I&C has verified no faults on the DRPI system.
- I&C has verified that the rod lift coil for Control Rod H-8 is failed.

Which one of the following completes the below statements?

Based on the initial conditions, at 0900 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

Based on the current conditions, at 0945 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

	Rod H-8 status <u>at 0900</u>	Rod H-8 Status <u>at 0945</u>
A.	OPERABLE	inoperable
B.	inoperable	inoperable
C.	OPERABLE	OPERABLE
D✓	inoperable	OPERABLE

005AG2.1.07 Inoperable/Stuck Control Rod

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
(CFR 41.5. / 43.5 / 45.12 / 45.13)

<u>K/A MATCH ANALYSIS:</u>

determine whether the rod is OPERABLE when the rod is initially discovered to be not moving. I & C will then report back a problem with lift coils has been discovered and the candidate must determine with the new information whether rod is OPERABLE.

10 CFR 55.43(b)(2)

DISTRACTOR ANALYSIS

A. Incorrect. First part is incorrect. Per Tech Spec Bases, rods that will NOT move are considered untrippable unless there is verification that a rod control system failure is preventing rod motion.
Second part is incorrect, once I & C determined a lift coil problem existed, the lift coil problem will not prevent the rod from tripping, the rod is operable.

B. Incorrect. The first part is correct. Per Tech Spec Bases, rods that will NOT move are considered untrippable unless there is verification that a rod control system failure is preventing rod motion.
Second part is incorrect, once I & C determined a lift coil problem existed, the lift coil problem will not prevent the rod from tripping, the rod is operable.

C. Incorrect. First part is incorrect. Per Tech Spec Bases, rods that will NOT move are considered untrippable unless there is verification that a rod control system failure is preventing rod motion.
Second part is correct. Once I & C determined a lift coil problem existed, the lift coil problem will not prevent the rod from tripping, the rod is operable.

D. Correct. The first part is correct. Per Tech Spec Bases, rods that will NOT move are considered untrippable unless there is verification that a rod control system failure is preventing rod motion.
Second part is correct. Once I & C determined a lift coil problem existed, the lift coil problem will not prevent the rod from tripping, the rod is operable.

REFERENCES

Tech Spec and Bases for 3.1.4, Rod Group Alignment Limits.
HL-17 NRC Q #79 Re-use (014A2.04)

VEGP learning objectives

LO-LP-39205-02 Given a set of Tech Specs and the Bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCO's of section 3.1 are exceeded.

b. The required actions for all section 3.1. LCO's.

LO-LP-39205-08 Describe the bases for any given Tech Spec in section 3.1.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions 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) untrippable.	A.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	<u>OR</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.3	Verify SDM is ≥ the limit specified in the COLR.	Once per 12 hours
	AND		
	B.4	Perform SR 3.2.1.1.	72 hours
	AND		
	B.5	Perform SR 3.2.2.1.	72 hours
	AND		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

BASES (continued)

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The rod OPERABILITY (i.e., trippability) requirement is satisfied provided that the rod will fully insert in the required rod drop time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable).

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. When required, movable incore detectors may be used to determine rod position and verify the rod alignment requirement of this LCO is met.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which a self-sustaining chain reaction ($K_{eff} \geq 1$) occurs, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down, with no self-sustaining chain reaction. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

1. 014A2.04 001/2/2/NIS/H 3.4/3.9/NEW/HL-17 NRC/SRO/EMT/GCW

Initial Conditions:

- Time=0900
- Unit 1 is at 60% power following a refueling outage.
- The OATC is withdrawing rods when one DRPI is seen not moving with its group.
- The OATC immediately stops withdrawing rods, and all rod motion stops.
- CBD Group 2 Rod H-8 DRPI indicates 198 steps.
- CBD Group 2 step counters indicate 212 steps.

Current Conditions:

- Time=0945
- No rod motion has occurred since 0900.
- I & C has verified no faults on the DRPI system.
- I & C has verified that the rod lift coil for Control Rod H-8 is failed.

Which one of the following correctly completes the below statements?

Based on the initial conditions, at 0900 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

Based on the current conditions, at 0945 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

	Rod H-8 status at 0900	Rod H-8 Status at 0945
A.	OPERABLE	INOPERABLE
B.	INOPERABLE	INOPERABLE
C.	OPERABLE	OPERABLE
D✓	INOPERABLE	OPERABLE

77. 005G2.1.32 001/2/1/RHR - COND OPS/MEM - 3.8/4.0/NEW/HL-18 NRC/SRO/TNT

Given the following plant conditions:

- Unit 1 is in Mode 5.
- 'A' RHR in service for Shutdown Cooling.

Which ONE of the following completes the following statement?

To comply with Tech Spec 3.4.7, "RCS Loops - Mode 5, Loops Filled," an RCP can be started if the MAXIMUM Secondary side water temperature of each SG is __(1)__ above each of the RCS Cold Leg temperatures,

and

the bases for this temperature is to __(2)__.

A. (1) < 50 °F

(2) prevent a vapor bubble forming and possibly causing a natural circulation flow obstruction

B✓ (1) < 50 °F

(2) prevent a low temperature overpressure event due to a thermal transient when an RCP is started

C. (1) < 25 °F

(2) prevent a vapor bubble forming and possibly causing a natural circulation flow obstruction

D. (1) < 25 °F

(2) prevent a low temperature overpressure event due to a thermal transient when an RCP is started

005G2.1.32 Residual Heat Removal System (RHRS)

Ability to explain and apply system limits and precautions: (CFR 41.10 / 43.2 / 45.12):

K/A MATCH ANALYSIS:

The question presents a plausible scenario where an RCP is being started in Mode 5. The candidate has to determine the maximum temperature at which the RCP is allowed to be started and the Tech Spec Bases. This ensures with the RHR suction valves

overpressure protection system (for RHR design pressure considerations) can handle the event.

DISTRACTOR ANALYSIS:

A. Incorrect. The temperature limit is correct, however the bases is from the Safety Analysis of Tech Spec 3.4.7 for Mode 5 RCP starting from a different note (10 degree limit) so it is plausible the candidate may confuse this bases.This is an incorrect temperature as stated in the notes of Tech Spec 3.4.7 Note # 1.b, it is plausible the candidate may confuse this temperature for the different mode spec. The bases is for NOTES 1.b.

B. Correct. This is the correct temperature limit per Tech Spec 3.4.7 and the correct bases from the Safety Analysis of 3.4.7 and 3.4.12 (COPS) which insures that the COPS overpressure protection system can handle the event.

C. Incorrect. This is an incorrect temperature as stated in the notes of Tech Spec 3.4.7 Note # 1.b, it is plausible the candidate may confuse this temperature for the different mode spec. The bases is for NOTES 1.b.

D. Incorrect. This is an incorrect temperature as stated in the notes of Tech Spec 3.4.7 Note # 1.b. This is the correct bases from Tech Spec 3.4.7 safety analysis and 3.4.12 (COPS) which insures that the COPS overpressure protection system can handle the event.

REFERENCES:

Tech Spec 3.4.7, RCS Loops - Mode 5, Loops Filled and Bases
Tech Specs 3.4.12, Cold Overpressure Protection Systems (COPS) and Bases
13011-1, "Residual Heat Removal System", P&L 2.2.4 and 2.2.8

VEGP learning objectives:

LO-LP-39208-01 For any given item in section 3.4 of Tech Specs, be able to:

a. State the LCO.

b. State any one hour or less required actions.

LO-LP-39208-04 Describe the bases for any given Tech Spec in section 3.4.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes SRO only

No Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops — MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be above the highest point of the steam generator U-tubes.

---NOTES---

1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

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

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started unless the secondary side water temperature of each SG is < 50°F above each of the RCS cold leg temperatures.

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

BASES

LCO (continued)

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be < 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) during MODE 5 with the RCS loops filled. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it has an adequate water level and is OPERABLE. Additional requirements for an SG to be available as a heat sink are:

a. RCS loops and reactor pressure vessel filling and venting complete; and

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Protection Systems (COPS)

LCO 3.4.12 A COPS shall be OPERABLE with all safety injection pumps incapable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 440 psig and ≤ 460 psig, or

3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint within specified limits.

b. The RCS depressurized and an RCS vent of ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe).

APPLICABILITY: MODE 4 with any RCS cold leg temperature ≤ the COPS arming temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

--NOTE--
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The following are required during the COPS MODES or other specified condition in the COPS Applicability to ensure that mass and heat input transients do not occur, which either of the COPS overpressure protection means cannot handle:

a. Rendering both safety injection pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing the start of an RCP if the secondary temperature is more than 50°F above the RCS cold leg temperature in any one loop. With the RHR suction isolation valves open, this value is reduced to 25°F at an RCS temperature of 350°F and varies linearly to 50°F at an RCS temperature of 200°F for RHR design pressure considerations. LCO 3.4.6, "RCS Loops — MODE 4," and LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," contain notes on this limitation that provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when both centrifugal charging pumps are actuated. Thus, the LCO requires both safety injection pumps to be incapable of injecting into the RCS during the COPS MODES or other specified condition in the COPS Applicability.

Since neither one RCS relief valve nor the RCS vent can handle the pressure transient caused by accumulator injection when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits shown in the PTLR. The setpoints are derived by analyses that model the performance of the COPS, assuming

(continued)

2.2 LIMITATIONS

2.2.1 In Modes 1, 2 & 3 the RHR is required to have an operable RHR pump and HX in each train per Technical Specification LCO 3.5.2.

2.2.2 In Mode 4 the RHR is required to be operable or in operation per Technical Specification LCO 3.4.6.

2.2.3 In Mode 4 one train of RHR is required to be operable per Technical Specification LCO 3.5.3.

2.2.4 In Mode 5 the RHR is required to be operable and in operation per Technical Specification LCO 3.4.7 and LCO 3.4.8.

2.2.5 During refueling operations, the RHR is required to be operable and in operation per Technical Specification LCO 3.9.5 and LCO 3.9.6.

2.2.6 Minimum RHR Flow of ≥3000 gpm is only required in Mode 6.

2.2.7 In Mode 6 only, one train of RHR may be used as an OPERABLE boron injection flow path provided a flow path is established from the OPERABLE RWST via an RHR pump through the cold legs with water level ≥23 feet above the reactor vessel flange. The RHR pump may not be the pump that is being applied to meet LCO 3.9.5. (TR 13.1.2, 13.1.4) (LDCR TM 97-004)

2.2.8 If required for Cold Overpressure Protection, two RHR Suction Relief Valves are required OPERABLE per Technical Specification LCO 3.4.12.

2.2.9 When in Mode 1, 2, or 3, 1-HV-8716A/B must remain open to ensure injection flow into all cold legs consistent with the Westinghouse Design Bases and FSAR Accident Analysis. However, one valve at a time may be closed for short periods during surveillance testing. 14825-1, "Quarterly In service Valve Test" governs these manipulations.

2.2.10 When in Mode 1, 2, or 3, 1-HV-8809A/B must remain open to ensure injection flow into all cold legs consistent with the Westinghouse Design Bases and FSAR Accident Analysis. However, while in Mode 3, one valve at a time may be closed during surveillance testing per Technical Specification SR 3.4.14.1 as described in RER 87-1067. Procedure 14450-1, "RCS Pressure Isolation Valve Leak Test" governs these manipulations.

2.2.11 The RHR Suctions From Hot Legs Loops 1 and 4 (1-HV-8701A, 1-HV-8701B, 1-HV-8702A, 1-HV-8702B) are separately interlocked to prevent from being opened with RCS pressure greater than 365 psig.

78. 008A2.02 001/2/1/CCW - TANK LVL/C/A - 3.2/3.5/NEW/HL-18 NRC/SRO/KAJ

Initial conditions:

- Unit 1 is at 100% power.
- A leak develops on CCW Train 'A' to the Spent Fuel Pool Heat Exchanger.
- The CCW makeup valve has opened and surge tank level is slowly rising.

Current conditions:

- CCW to the Spent Fuel Pool Heat Exchanger has been isolated.
- CCW Train "A" surge tank level has stabilized.
- CCW Train 'A' remains in service with 2 pumps running.
- "B" Train Spent Fuel Pool Cooling has been placed in service.

Based on the existing conditions, which ONE of the following completes the following statement?

Per Tech Spec 3.7.7, "Component Cooling Water (CCW)," and Bases, the SS will declare CCW Train 'A' to be ___(1)___,

and

per 13715A-1, "Component Cooling Water System Train A," CCW Train 'A' is required to ___(2)___

A. (1) OPERABLE

(2) remain in its present configuration

B✓ (1) OPERABLE

(2) be placed in single pump operation

C. (1) inoperable

(2) remain in its present configuration

D. (1) inoperable

(2) be placed in single pump operation

008A2.02 Component Cooling Water System (CCWS)

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:
(CFR 41.5 / 43.5 / 45.3 / 45.13)

High / Low surge tank level

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where a CCW leak to the SFP Hx has occurred and has been isolated. 2 CCW pumps remain in operation and the candidate has to determine if the system is operable with the SFP Hx isolated and whether or not CCW has to be placed in single pump operations. The SR for Tech Spec 3.7.7 states isolation of a CCW load does not render CCW inoperable.

DISTRACTOR ANALYSIS:

A. Incorrect. CCW is operable per SR 3.7.7.1 with the SFP Hx isolated, however, 13715A-1 requires CCW train A to be placed in single pump operations.

B. Correct. CCW Train A is operable per SR 3.7.7.1 and has to be placed in single pump operations per 13715A-1.

C. Incorrect. CCW Train A is operable per SR 3.7.7.1 and is required to be placed in single pump operations per 13715A-1.

D. Incorrect. CCW Train A is operable per SR 3.7.7.1 and is required to be placed in single pump operations per 13715A-1.

REFERENCES:

13715A-1, Component Cooling Water System Train A, P & L 2.1.4
Tech Spec 3.7.7 Component Cooling Water, SR 3.7.7.1 NOTE
Tech Spec 3.7.7 Bases for Component Cooling Water System

VEGP learning objectives:

LO-PP-10101-01 From memory, state the following for the CCW System:

a. Heat Loads
c. System configuration for single pump operations

LO-LP-39211-04 Describe the bases for any given Tech Spec in section 3.7.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 Thoroughly fill and vent all applicable CCW components prior to returning them to service after maintenance. This minimizes system performance degradation due to gas entrainment.

2.1.2 To prevent overheating of CCW when the Residual Heat Removal (RHR) System is placed in service, CCW flow through the RHR Heat Exchangers should not be throttled for temperature control.

2.1.3 CCW Surge Tank level should be monitored closely following CCW pump starts to verify system leakage caused by relief valve failure to reseat is detected promptly.

2.1.4 Both the Spent Fuel Pool Cooling and RHR Heat Exchangers must be in service to run two CCW Pumps. If one of these heat exchangers is isolated (CCW side), then the CCW System shall be operated in single pump operation in accordance with Section 4.4.4 if continued system operation is required.

2.1.5 If the system is aligned for single pump operation, the SS should verify CAUTION TAGS are in place to reflect system configuration. Shift management may also consider TCPs to EOPs/AOPs and/or a Standing Order during the time that system is in single pump operation, (C00036391). TS 3.7.7, 3.5.2, 3.5.3, 3.4.6, 3.4.7, 3.4.8, 3.9.5, 3.9.6, and TRM 13.1.4 shall be reviewed prior to Single Pump Operation.

2.1.6 When operating the CCW System in single pump operation, system pressure should remain above 85 psi.

2.1.7 If a CCW Pump has tripped from an overcurrent condition, its 186 lockout device (located on the pump's respective 4160 volt switchgear breaker) will have to be reset prior to pump start. The System Status Monitor Panel will indicate "BYPASSED" until the lockout relay is reset.

2.1.8 Chemistry shall be contacted prior to draining system to remove radiation monitor sample pump from service. Engineering recommends isolating sample pump prior to system drain down to help prevent air binding. Multiple venting of the CCW Heat Exchanger may be necessary to verify no air binding of the radiation monitor sample pump.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-------------------------------NOTE------------------------------ Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

BASES

LCO (continued)	A CCW train is considered OPERABLE when: a. Two pumps and associated surge tank are OPERABLE; and b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE. The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not necessarily make the CCW System inoperable. Consideration should be given to the size of the load isolated and the impact it will have on the rest of the CCW system before determining OPERABILITY.
APPLICABILITY	In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger. In Modes 5 or 6, there are no TS OPERABILITY requirements for the CCW System. However, the functional requirements of the CCW System are determined by the systems it supports.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

(continued)

79. 008AG2.4.30 001/1/1/PZR VAPOR - EP/MEM - 2.7/4.1/NEW/HL-18 NRC/SRO/TNT

Which ONE of the following events will require the EARLIEST notification to the NRC in accordance with 00152-C, "Federal and State Reporting Requirements"?

A. The presence of a loose part in the RCS is confirmed.

B. Initiation of a plant shutdown in accordance with Tech Spec 3.0.3.

C.✓ A Pressurizer Safety Valve sticks open resulting in a Safety Injection actuation.

D. A confirmed violation of Fitness for Duty requirements by a licensed operator.

008AG2.4.30 Pressurizer Vapor Space Accident

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator: (CFR 41.10 / 43.5 / 45.11)

<u>K/A MATCH ANALYSIS:</u>

The question present a plausible scenario where a PRZR Safety Valve fails open resulting in an SI actuation. The candidate will have to realize that a LOCA is in progress requiring an early notification to the NRC (1 hour notification).

<u>ANSWER / DISTRACTOR ANALYSIS:</u>

A. Incorrect. Confirmation of a loose part in the RCS is required to be reported promptly within 24 hours per item # 39 of 00152-C. In addition, this is required by Reg Guide 1.1.33 Section C.6, NOT 10CFR50.72 also.

B. Incorrect. Initiation of any nuclear plant shutdown required by Technical Specification is required to be reported with 4 hours per item # 33.1 of 00152-C. In addition, this is required per 10CFR50.72 but is NOT required within 1 hour.

C. Correct. A safety valve failure resulting in an SI actuation will require an emergency declaration per the Emergency Plan. This is an indication an RCS leak has occurred that is greater than the capacity of the charging system. NRC is required to be notified immediately after state and local authorities and no later than 1 hour. This is item 2.1 of 00152-C and is a 10CFR50.72 requirement.

D. Incorrect. Violation of the FFD policy by a licensed operator is a 24 hour notification per item # 44 (1) (ii) of 00152-C. In addition, this is a 10CFR26.719(b) requirement NOT a 10CFR50.72 requirement.

<u>REFERENCES:</u>

00152-C, Federal and State Reporting Requirements. (Item # 2.1, 33.1, 37.6, 39, 44)

VEGP learning objectives:

LO-LP-63152-C Describe what is meant by the following terms: (SRO only)
a. Immediate notification
b. follow-up report
c. material access area
d. reportable events

LO-LP-63152-C State the responsibilities of personnel for Federal and State reporting requirements as described by Vogtle Procedure 00152-C, "Federal and State Reporting Requirements". (SRO only)

This question is SRO only because 00152-C, Federal and State Reporting Requirements is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

TABLE 2
FEDERAL AND STATE REPORTING MATRIX
PLANT ORIGINATED

Item No.	Requirement	Report Title Or Condition	Frequency	Method	Prepared By Reviewed By	Approved By	Submitted By	Submitted To
1	10CFR 50 Appendix E Sect IV.D.3	Emergency Notification	Within 15 minutes of declaring an emergency	ENN [4]Commercial Telephone	ED N/A	ED	ED	State & local govt.
2	10CFR50.72	Immediate Notification						
2.1	10CFR50.72 (a)(1)(i)	The declaration of any of the emergency classes specified in the licensees approved Emergency Plan NOTE: See 2.3 for follow-up notification required	Immediately after notification of state or local agencies and not later than one hour after the declaration of the Emergency Classes	ENS [2]Commercial Telephone	ED N/A	ED	ED	NRC-OC NRC-RI
2.2	10CFR50.72 (a)(1)(ii) & (b)(1)	For Non-Emergency events that occurred within 3 years of discovery. If not reported as a declaration of an Emergency Class under 2.1 above, report any deviation from the plant's Technical Specifications authorized pursuant to 10CFR 50.54(x) NOTE: See 2.3 for follow-up notification required	Within 1 hour from occurrence of deviation from Technical Specifications.	ENS [2]Commercial Telephone	SM N/A	SM	SM	NRC-OC NRC-RI
2.3	10CFR50.72 (c)	For items 2.1, 2.2, 33.1 thru 33.4, 37.1 thru 37.5 immediately report the following: 1. Any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the emergency classes, if such a declaration has not been previously made. 2. Any change from one emergency class to another, or a termination of the emergency class. 3. The results of ensuing evaluations or assessments of plant conditions. 4. The effectiveness of response or protective measures taken. 5. Information related to plant behavior that is not understood.	Immediately and maintain an open, continuous communication channel with NRC Operations Center upon request of the NRC.	ENS [2] Commercial Telephone	ED or SM N/A	ED or SM	ED or SM	NRC-OC NRC-RI
3	10CFR 50.91 (b)	Notice to state of request amendment of operating license	At the time of filing amendment request with NRC	Written	NLM PRB	VP/ NLM	VP	NRC to provide name of State official

Footnotes at end of Table 2

TABLE 2
FEDERAL AND STATE REPORTING MATRIX
PLANT ORIGINATED

Item No.	Requirement	Report Title Or Condition	Frequency	Method	Prepared By / Reviewed By	Approved By	Submitted By	Submitted To
28	10CFR 140.15(e)	Any material change in proof of financial protection or in any other financial information filed with the NRC under Part 140	Promptly	Written	SCS / Risk Mgt.	NLM	NLM	NRC-NRR or NRC-NMSS
29	10CFR 50.72(a)(4)	Activation of the Emergency Response Data System.	Within 1 hour after declaring emergency class (alert, site area emergency or general emergency)	Activation of ERDS satisfies this requirement	N/A	N/A	N/A	N/A
30	40CFR 260-265	No Reporting Required			N/A	N/A	N/A	N/A
31	40CFR 302	When releases of radioactivity exceed Tech Spec limits (item 92) and when subsequent releases, within a 24 hour period, equal a reportable quantity the EPA National Response Center must be notified	Each release	Telephone	CHEM / CM	VP	MEACRS	EPA
32	49CFR 171.15 NMP-EN-901-002	Incidents involving Reportable Quantity (RQ) of hazardous materials Note: The following responsibility apply to the person in physical control of the hazardous material at the time of the release.	Earliest practicable moment and written report within 30 days of the incident	Telephone Written	EACRS or Transporter		SM	USCG DOT (if a RQ of a hazardous substance is released)
33	10CFR 50.72(b)(2)	Events requiring 4 hour notification						
33.1	10CFR50.72 (b)(2)(i)	Initiation of any nuclear plant shutdown required by Technical Specifications NOTE: See 2.3 for follow-up notification required	As soon as practical and in all cases within 4 hours	ENS [2] Commercial Telephone	SM / N/A	SM	SM	NRC-OC NRC-RI
33.2	10CFR50.72 (b)(2)(iv)(A)	Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. NOTE: See 2.3 for follow-up notification required	As soon as practical and in all cases within 4 hours	ENS [2] Commercial Telephone	SM / N/A	SM	SM	NRC-OC NRC-RI

Footnotes at end of Table 2

TABLE 2
FEDERAL AND STATE REPORTING MATRIX
PLANT ORIGINATED

Item No.	Requirement	Report Title Or Condition	Frequency	Method	Prepared By Reviewed By	Approved By	Submitted By	Submitted To
37.3	10CFR 50.72 (b)(3) (v)&(vi)	Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or system that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition, (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident Events covered above may include one or more procedural errors, equipment failures. and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function NOTE: See Item 2.3 for follow-up required.	Within 8 hours	ENS [2] Commercial Telephone, Other System	SM N/A	SM	SM	NRC-OC NRC-RI
37.4	10CFR 50.72(b)(3) (xii)	Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment. Reference; NUREG 1022 Section 3.2.11 NOTE: See item 2.3 for follow-up required.	Within 8 hours	ENS [2] Commercial Telephone	SM HPM	SM	SM	NRC-OC NRC-RI
37.5	10CFR 50.72(b)(3) (xiii)	Any event that results in a major loss of emergency assessment capability, offsite response capability or communications capability (e.g., significant portion of control room indication, Emergency Notification System or offsite notification system as delineated in EPIP 91204-C and 91706-C) (1991321693) NOTE: See item 2.3 for follow-up required.	Within 8 hours	ENS [2] Commercial Telephone	SM HPM	SM	SM	NRC-OC NRC-RI
37.6	FSAR Section 5.4.13 NUREG-0737	Relief and safety valve failures to close will promptly be reported to the nuclear regulatory commission. (1989316194)	Promptly	LER Process	OM PIS PRB	VPP	NLM	NRC-OC NRC-R

Footnotes at end of Table 2.

TABLE 2
FEDERAL AND STATE REPORTING MATRIX
PLANT ORIGINATED

Item No.	Requirement	Report Title Or Condition	Frequency	Method	Prepared By / Reviewed By	Approved By	Submitted By	Submitted To
38	NUREG 0654 Appendix 1	Unusual Event Summary Report	Within 24 hour of close out of Unusual Event	[4] Written	ED / N/A	ED	ED	NRC-RO NRC-RI State & Local Govt.
39	Reg Guide 1.133 Section C.6	The presence of a loose part is confirmed. Report condition in accordance with prompt notification requirements (1985304490)(1985307529) NOTE: See item 66 for written report	Promptly Within 24 hours	ENS [2] Commercial Telephone	SM / N/A	SM	SM	NRC-OC NRC-RI
40	Tech Specs 2.2.4	Safety limit violation Notify the PM and VP (1995330840) (See Item 4)	Within 24 hours	Commercial Telephone	SM / N/A	SM	SM	VP SRB PRB PM
41	10CFR26.719(c) (1)	Drug and alcohol testing errors or unsatisfactory performance discovered in performance testing at either a licensee testing facility or an (Health and Human Services) HHS-certified laboratory, in the testing of quality control or actual specimens, or through the processing of reviews under § 26.39 and MRO reviews under § 26.185, as well as any other errors or matters that could adversely reflect on the integrity of the random selection or testing process, the licensee or other entity shall submit to the NRC a report of the incident and corrective actions taken or planned.	Within 30 days of completion of the investigation	Written	SNSH / NL	NLM	NLM	NRC-DCD
41.1	10CFR26.719(c) (2)	False positive error on a blind performance test sample	Within 24 hours of discovery	Telephone	SM / N/A	SM	SM	NRC-OC NRC-RI
41.2	10CFR26.719(c) (3)	False negative error on a QA check of validity screening test.	Within 24 hours of discovery	Telephone	SM / N/A	SM	SM	NRC-OC NRC-RI

Footnotes at end of Table 2.

TABLE 2
FEDERAL AND STATE REPORTING MATRIX
PLANT ORIGINATED

Item No.	Requirement	Report Title Or Condition	Frequency	Method	Prepared By Reviewed By	Approved By	Submitted By	Submitted To
44	10CFR 26.719(b)	Notify the commission of *significant FFD policy violations or programmatic failures.* The following significant FFD policy violations and programmatic failures must be reported to the NRC Operations Center by telephone within 24 hours after the licensee or other entity discovers the violation: (1) The use, sale, distribution, possession, or presence of illegal drugs, or the consumption or presence of alcohol within a protected area. (2) Any acts by any person licensed under 10 CFR parts 52 and/or 55 to operate a power reactor, as well as any acts by SSNM transporters, FFD program personnel, or any supervisory personnel who are authorized under this part, if such acts— (i) Involve the use, sale, or possession of a controlled substance; (ii) Result in a determination that the individual has violated the licensee's or other entity's FFD policy (including subversion as defined in § 26.5); or (iii) Involve the consumption of alcohol within a protected area or while performing the duties that require the individual to be subject to the FFD program; (3) Any intentional act that casts doubt on the integrity of the FFD program; and (4) Any programmatic failure, degradation, or discovered vulnerability of the FFD program that may permit undetected drug or alcohol use or abuse by individuals within a protected area, or by individuals who are assigned to perform duties that require them to be subject to the FFD program.	Within 24 hours	Telephone	SM N/A	SM	SM	NRC-OC
45	NPDES Permit No. GA0026786	Non-Compliance with any daily maximum effluent limitation specified in the permit.	Within 24 hours	[8] Telephone	CHEM CM ENVS	VP	MEACRS	GEPD
46		Not Used						

80. 011EA2.07 001/1/1/LRGE LOCA - WTR SEAL/C/A - 3.2/3.4/NEW/HL-18 NRC/SRO/KAJ

Initial conditions on Unit 2:

- A Large Break LOCA and Safety Injection occurred.
- All RCPs are stopped.
- 2AA02 is deenergized due to a fault.

Current conditions / events:

- The crew is implementing 19221-C, "Response to Inadequate Core Cooling."
- ACCW Pump #2 trips due to a locked rotor.
- Containment pressure is 12 psig.
- Core exit thermocouples are 1220°F and rising.
- SG NR levels are as follows:

SG #1 = 30% SG #2 = 21% SG #3 = 34% SG #4 = 26%

Based on the given conditions, which ONE of the following completes the following statement?

Per 19221-C, the Shift Supervisor ___(1)___ direct the start of at least one RCP,

and

will transition to 19010-C, "Loss of Reactor or Secondary Coolant," if at least two RCS WR Hot Leg temperatures lower to less than ___(2)___.

A. (1) will NOT

(2) 350°F

B. (1) will NOT

(2) 711°F

C.✓ (1) will

(2) 350°F

D. (1) will

(2) 711°F

011EA2.07 Large Break LOCA

Ability to determine or interpret the following as they apply to a Large Break LOCA: (CFR 43.5 / 45.13)

That equipment necessary for functioning of critical pump water seals is operable

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario during performance of 19221-C and has to determine whether or not to start RCPs with presented conditions (no ACCW available for seal cooling). The candidate also has to determine the wide range Th temperatures that will allow transition to 19010-C.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Per 19221-C, RCP will be started to attempt to obtain core cooling. Thot temperature of 350°F is where exit to 19010-C is allowed.

B. Incorrect. Per 19221-C, RCP will be started to attempt to obtain core cooling. 711°F is used as a plausible choice since it is mentioned earlier in the procedure to allow transition to 19010-C, however this is based on core exits and not Thot.

C. Correct. RCPs will be started in the present conditions (no ACCW cooling) and NR SG level >10% (32%). 350°F is the Thot which will allow transition to 19010-C.

D. Incorrect. RCPs will be started in the present conditions (no ACCW cooling). 711°F is used as a plausible choice since it is mentioned earlier in the procedure to allow transition to 19010-C, however this is based on core exits and not Thot.

REFERENCES:

19221-C, FR-C.1, Response to Inadequate Core Cooling
WOG Background Document for FR-C.1, Response to Inadequate Core Cooling

VEGP learning objectives:

LO-LP-37061-02 Using EOP 19221-C as a guide, briefly describe how each step is accomplished.

LO-LP-37061-07 State operator actions that can prevent or control core voiding.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. Check Core cooling:	
__a. Core Exit TCs - LESS THAN 1200°F.	__a. Go to Step 23.
__b. RVLIS full range indications - GREATER THAN 63%.	__b. Return to Step 19.
__c. At least two RCS WR Hot Leg temperatures - LESS THAN 350°F.	__c. Return to Step 19.
__22. Go to 19010-C, E-1 LOSS OF REACTOR OR SECONDARY COOLANT, Step 17.	

NOTE

Normal conditions are desired but NOT required for starting the RCPs.

23. Check if RCPs should be started:	
__a. Core Exit TCs - GREATER THAN 1200°F.	__a. Go to Step 26.
b. Check if an idle RCS cooling loop is available:	__b. Go to Step 24.
__• NR SG level - GREATER THAN 10% [32% ADVERSE].	
__• RCP in associated loop - AVAILABLE AND NOT OPERATING.	

Step 23 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__c. Start RCP in one idle RCS cooling loop.	
__d. Return to Step 23.a.	
24. Establish RCS vent path:	
__a. Arm both trains of COPS.	
__b. Open all PRZR PORVs and Block Valves.	
c. Check Core Exit TCs - REMAIN GREATER THAN 1200°F.	c. Go to Step 26.
25. Open all other RCS vent paths to Containment:	
a. Open RX HEAD VENT TO LETDOWN ISOLATION VLVs: __• HV-8095A __• HV-8096A __• HV-8095B __• HV-8096B __• HV-0442A __• HV-0442B	
__b. Reset SI.	__b. IF SI will NOT reset, THEN initiate ATTACHMENT E.
__c. Reset CI-A.	
__d. Open INSTR AIR CNMT ISO VLV HV-9378.	

Step 25 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
30. Check if RCPs should be stopped:	
__a. At least two RCS WR Hot Leg temperatures - LESS THAN 350°F.	__a. Go to Step 31.
__b. Stop all RCPs.	
31. Check ECCS flow:	31. Continue efforts to establish ECCS flow:
__CCP flow indicators - CHECK FOR BIT FLOW.	__• CCPs through BIT
-OR-	__• SIPs
	__• RHR Pumps
__SI flow indicators - CHECK FOR FLOW.	<u>IF</u> CCP flow <u>NOT</u> verified, <u>THEN</u> perform the following:
-OR-	__a. Reset SI if necessary.
__RHR flow indicators - CHECK FOR FLOW.	__ <u>IF</u> SI will <u>NOT</u> reset, <u>THEN</u> initiate ATTACHMENT E.
	__b. Start the NCP.
	__Return to Step 23.
32. Check Core cooling:	
__a. At least two RCS WR Hot Leg temperatures - LESS THAN 350°F.	__a. Return to Step 23.
__b. RCPs - NONE RUNNING.	__b. Stop all RCPs.

Step 32 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__c. RVLIS full range indication - GREATER THAN 63%.	__c. Return to Step 23.
33. Check RCS vent paths - CLOSED:	
__a. PRZR PORVs.	__a. Close PRZR PORVs.
	__ IF any PRZR PORVs can NOT be closed, THEN close its Block Valve.
b. Other RCS vent paths - CLOSED:	__b. Close any open RCS vent paths.
1) RX HEAD VENT TO LETDOWN ISOLATION VLVs:	
__ • HV-8095A __ • HV-8095B __ • HV-8096A __ • HV-8096B	

Step 33 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2) CVCS letdown isolation valves:	
___ • HV-8149A - LETDOWN ORIFICE 45 GPM	
___ • HV-8149B - LETDOWN ORIFICE 75 GPM	
___ • HV-8149C - LETDOWN ORIFICE 75 GPM	
___ • LV-0459 - LETDOWN ISOLATION VLV DOWNSTREAM	
___ • LV-0460 - LETDOWN ISOLATION VLV UPSTREAM	
3) EXCESS LETDOWN LINE ISO VLVs:	
___ • HV-8153	
___ • HV-8154	
___ 34. Go to 19010-C, E-1 LOSS OF REACTOR OR SECONDARY COOLANT, Step 17.	

END OF PROCEDURE TEXT

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

PURPOSE: To inform the operator to start RCPs when required even if all normal startup conditions have not been met

BASIS:

The RCPs could be required to temporarily cool the core under highly voided RCS conditions. The RCPs should be started when required even if all normal startup conditions have not been met. Failure to start the RCPs when required could result in core damage.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

STEP: Check If RCPs Should Be Started

PURPOSE: To ensure core exit TC temperatures are greater than 1200°F before restarting RCPs

BASIS:

The operator will enter this step if:

a. He is unable to depressurize the SGs; or
b. SG depressurization was not effective in restoring adequate core cooling; or
c. Secondary heat sink is lost

The actions of Step 18 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.

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 low-head safety 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.

The pressurizer PORVs require instrument air for long-term operation, however,

instrument air may not be available to the pressurizer PORVs if the event sequence included: a) initiation of a Phase A containment isolation signal, or b) a coincident loss of instrument air. For example, small LOCA event sequences may result in initiation of safety injection, initiation of containment isolation Phase A, subsequent repressurization of the RCS to the pressurizer PORV setpoint and cycling of the PORVs. If the instrument air supply was lost to the pressurizer PORVs, a large volume air receiver located inside containment can provide for limited operation (i.e., number of cycles) of the pressurizer PORVs. Should FR-C.1 subsequently be implemented, by the time that the operator would perform Step 18, the pressurizer PORVs may have lost their ability to open. Hence the operators may not be able to open the PORVs and maintain them open to rapidly depressurize the RCS. To address this possibility, the following actions are performed in the RNO column:

o Reset SI signal - The action to reset automatic actuation logic is taken so that safeguards equipment that receive the SI signal may be realigned or reset.

o Reset containment isolation Phase A - The action to reset automatic actuation logic is taken so that equipment (e.g., isolation valves) that receive a Phase A signal can be realigned. No valve will reposition upon actuation of the reset, but subsequent control actions will open the valves. Until the cause of the automatic actuation is determined or corrected, Phase A containment isolation valves should remain closed unless required to be opened to establish necessary process streams such as instrument air.

o Start one air compressor and establish instrument air to containment - The actions to provide a sustained source of instrument air to containment is taken to support operation of air-operated equipment inside containment such as the pressurizer PORVs. The instrument air system for the ERG Reference Plant includes an air receiver inside containment to allow limited equipment operation, however, the line from the air compressor (located outside containment) to the air receiver is isolated with Phase A isolation. In addition to

opening the containment isolation valves, a compressor may also have to be started (with attendant electrical considerations) to establish a sustained source of instrument air to equipment inside containment.

81. 015A2.04 001/2/2/NI - EFFECTS/MEM - 3.3/3.8/NEW/HL-18 NRC/SRO/TNT

Initial conditions:

- Unit 1 is at 100% power.

- **1600 on 5-13-2013,** the 7 day surveillance for Tech Spec 3.2.3, Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology) was discovered missed.

Current conditions:

- A Xenon oscillation is in progress causing AFD to rise toward the target value.

- The crew is performing 12004DF-1, "Power Operation (Mode 1)," Section 4.3.2 for AFD Control.

Which one of the following completes the following statement?

To delay declaring the LCO NOT met, the surveillance is required to be performed satisfactorily no later than ___(1)___

and

to properly dampen the xenon oscillation, the crew will insert CBD ___(2)___,

A. (1) 1600 on 5-20-2013

(2) when Delta I reaches the right side of the doghouse to force Delta I down to target

B✓ (1) 1600 on 5-20-2013

(2) insert CBD over time as Delta I is rising to keep Delta I on target

C. (1) 1600 on 5-14-2013

(2) when Delta I reaches the right side of the doghouse to force Delta I down to target

D. (1) 1600 on 5-14-20

(2) insert CBD over time as Delta I is rising to keep Delta I on target

015A2.04 Nuclear Instrumentation System (NIS)

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:
(CFR: 41.5 / 43.5 / 45.3 / 45.5)

Effects on axial flux density of control rod alignment and sequencing, xenon production and decay, boron vs control rod reactivity changes.

K/A MATCH ANALYSIS:

Question requires the candidate to determine or predict the correct method (operations) for Delta I control during a Xenon oscillation when Delta I is rising above target as required per 12004DF-1, Power Operation Mode 1, section 4.3.2, AFD Control.

The candidate also has to determine the correct completion time from time of discovery if the AFD Surveillance for Tech Spec 3.2.3 is missed (7 day surveillance requirement).

This question is SRO per Figure 1 of the NRC Clarification Guidance for SRO-only questions (Tech Specs) due to knowledge of TS bases that is required to analyze TS required actions and terminology.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. This is an incorrect method to dampen a Delta I oscillation, waiting for the peak and forcing Delta I down to target will initiate a diverging xenon oscillation and is NOT the method prescribed in 12004DF-1, section 4.3.2 for AFD Control. 2nd half of this choice is correct.

B. Correct. This is the correct method for dampening a Delta I oscillation in accordance with 12004DF-1, section 4.3.2 AFD Control.

C. Incorrect. This is an incorrect method to dampen a Delta I oscillation, waiting for the peak and forcing Delta I down to target will initiate a diverging xenon oscillation and is NOT the method prescribed in 12004DF-1, section 4.3.2 for AFD Control. Per SR 3.0.3, from time of discovery of the missed surveillance, you have 24 hours or up to the limit of the specified frequency, whichever is greater. It is plausible the candidates may confuse this with Completion Time 1.3 which is the most restrictive of the two times.

D. Incorrect. This is the correct method for dampening a Delta I oscillation in accordance with 12004DF-1, section 4.3.2 AFD Control. The first half of this choice is correct. Per SR 3.0.3, from time of discovery of the missed surveillance, you have 24 hours or up to the limit of the specified frequency, whichever is greater. It is plausible the candidates may confuse this with Completion Time 1.3 which is the most restrictive of the two times.

REFERENCES:

REFERENCES:

Data Sheets for AFD Target from core cycle 17.
12004DF-1, Power Operation (Mode 1), section 4.3.2 for AFD Control
SR 3.0.3, Surveillance Requirement Applicability
1.3 Completion Times

VEGP learning objectives:

LO-LP-39206-05 Define AFD target band. State when it is used.
LO-LP-39204-04 State the allowable time intervals for extension of surveillances. State the result of failure to perform surveillances within this period.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

INITIALS

4.3.2 AFD Control (1984301405, 1985303129, 1985303130, 1985304319)

> NOTE
>
> Control Rod movement resulting from this section should be documented on Attachment 1. ☐

4.3.2.1 AFD Control Guidelines

a. Individual ex-core AFD indications should be monitored periodically. ______

b. AFD is considered outside of its limits when two or more operable ex-core channels are indicating AFD outside the limits. Individual channel AFD indications can be substantially different from the average ex-core channel AFD. ______

c. ABOVE 15% reactor power, AFD should be controlled at or NEAR the target value (PTDB Tab 6.0). ______

d. WHEN AT or ABOVE 50% reactor power, if AFD moves outside the limits of PTDB TAB 6, **comply** with TS LCO 3.2.3. ______

4.3.2.2 AFD Control Strategy

a. WHILE operating in the POWER OPERATION section, IF it is projected that AFD will exceed ±3 AFD units from the Target Value, or the swing of AFD is affecting the ability of the operating crew to maintain Tavg within ±0.5°F of Tref, **notify** Reactor Engineering to **perform** a detailed analysis of prior operating history to **determine** the appropriate time to **initiate** control and to **determine** the timing of subsequent control actions. ______

INITIALS

b. IF it is necessary to take action prior to receiving guidance from Reactor Engineering, requested in substep a. above, **perform** the following to dampen the oscillation while controlling Tavg within ±0.5°F:

(1) As Delta I rises up to approach the Target Delta I, **insert** CBD over time, using the guidance of Attachment 1, (typically around 10 steps), keeping Delta I on target. Use Attachment 1 to document control rods in AUTO and MANUAL. ________

(2) Slowly **withdraw** control rods as flux allows (typically, several days). Use Attachment 1 to document control rods in AUTO and MANUAL. ________

c. WHILE operating in the END-OF-LIFE COASTDOWN section, AFD is expected to move positive as power and temperature are REDUCED. Rod motion may be required to control AFD. AFD should be controlled to maintain 3% margin from the right side of the "doghouse" (PTDB Tab 6.0). IF AFD cannot be controlled inside this margin, **notify** Reactor Engineering. ________

d. WHILE performing a plant shutdown for maintenance activities or refueling, at greater than or equal to 50% power AFD should be controlled within the limits of PTDB Tab 6, or per the AFD control strategy provided by Reactor Engineering. ________

e. Rods should be **operated** in MANUAL control if automatic rod insertion to restore Tave to program following a transient is projected to drive AFD outside the limits of PTDB Tab 6. ________

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

1.3 Completion Times

DESCRIPTION (continued)

However, when a <u>subsequent</u> train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The <u>total</u> Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

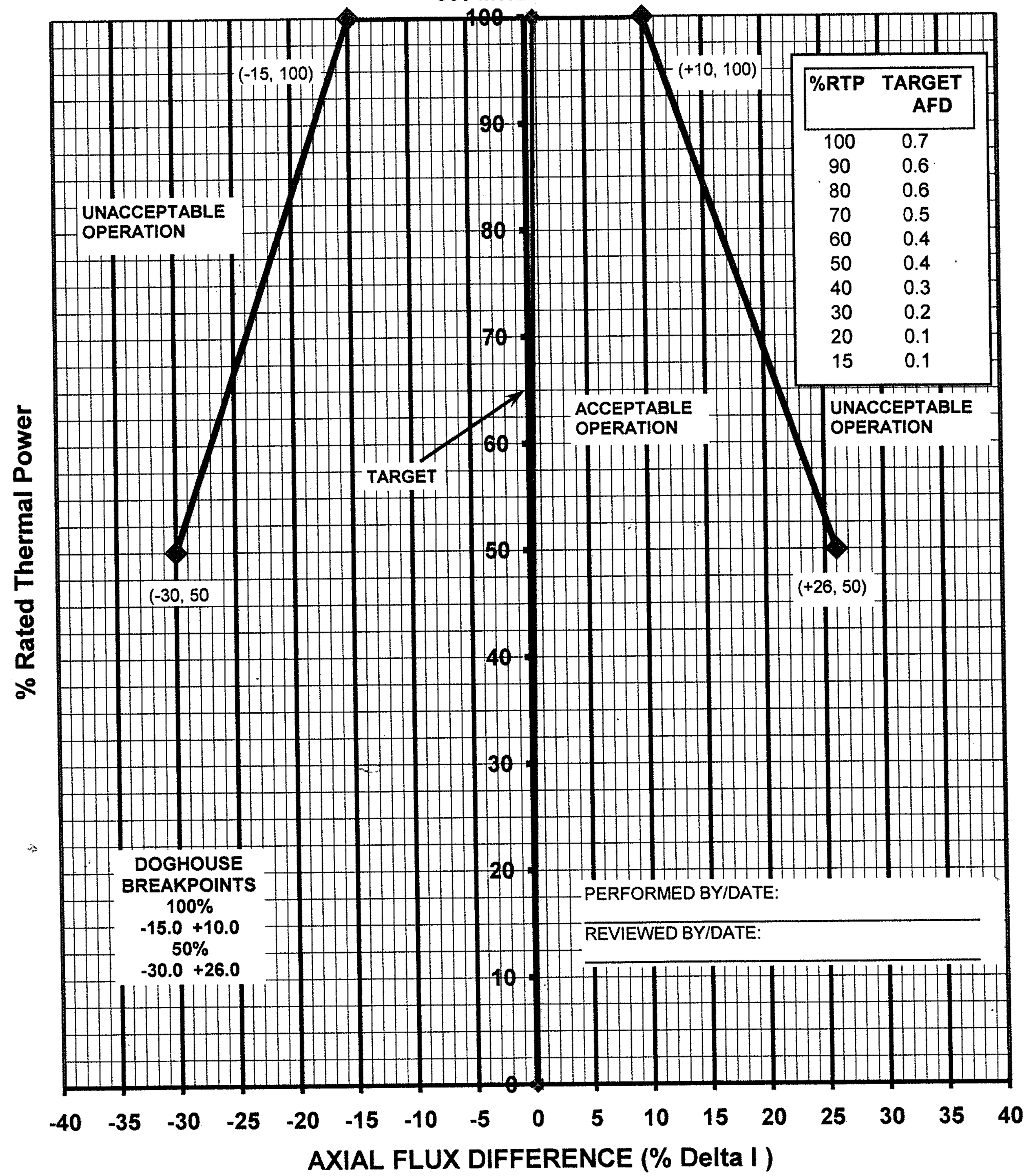
The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

Axial Flux Difference Limits As A Function Of Rated Thermal Power

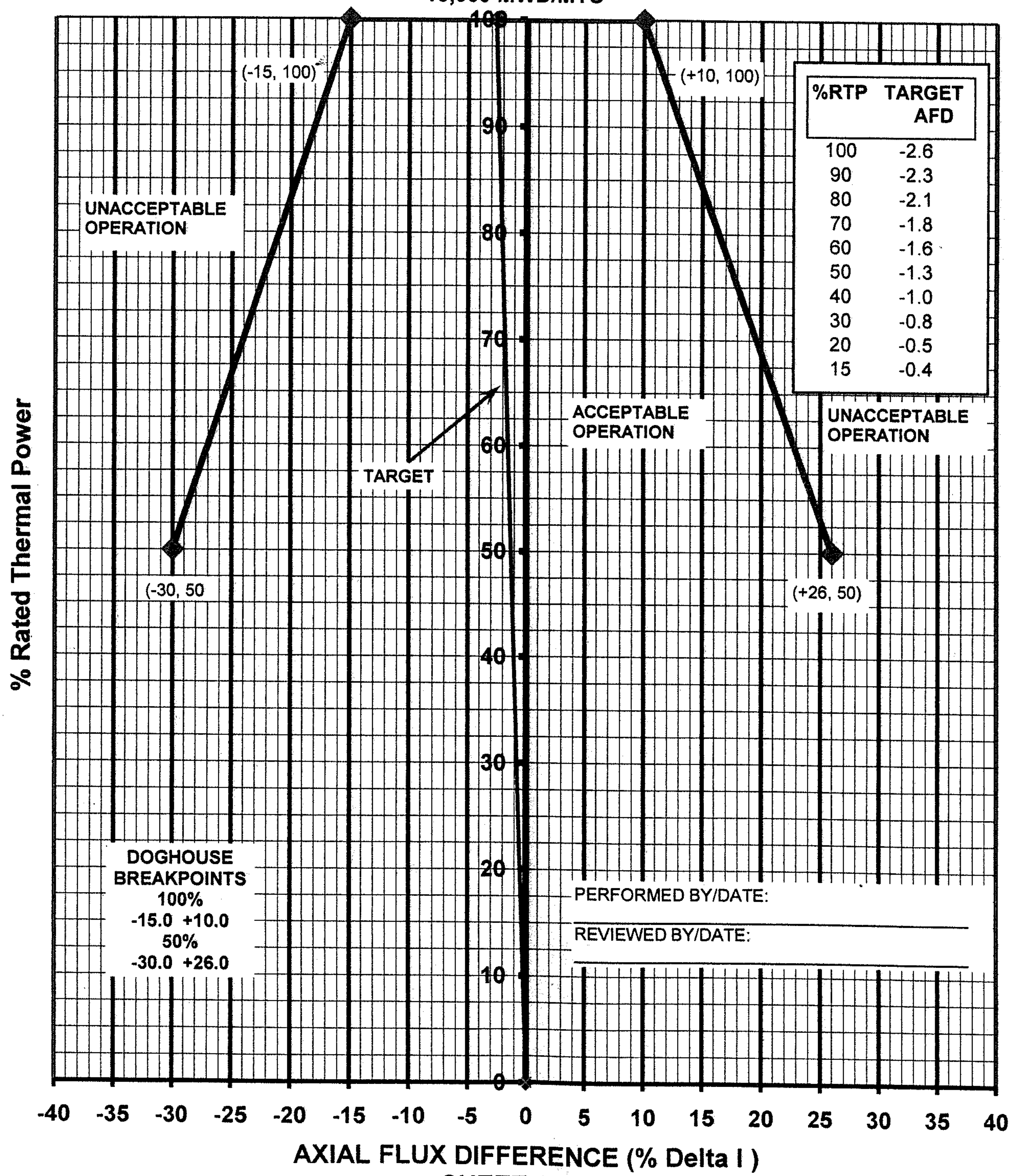
UNIT 1 CYCLE 17
500 MWD/MTU



SHEET 1 OF 1

Axial Flux Difference Limits As A Function Of Rated Thermal Power

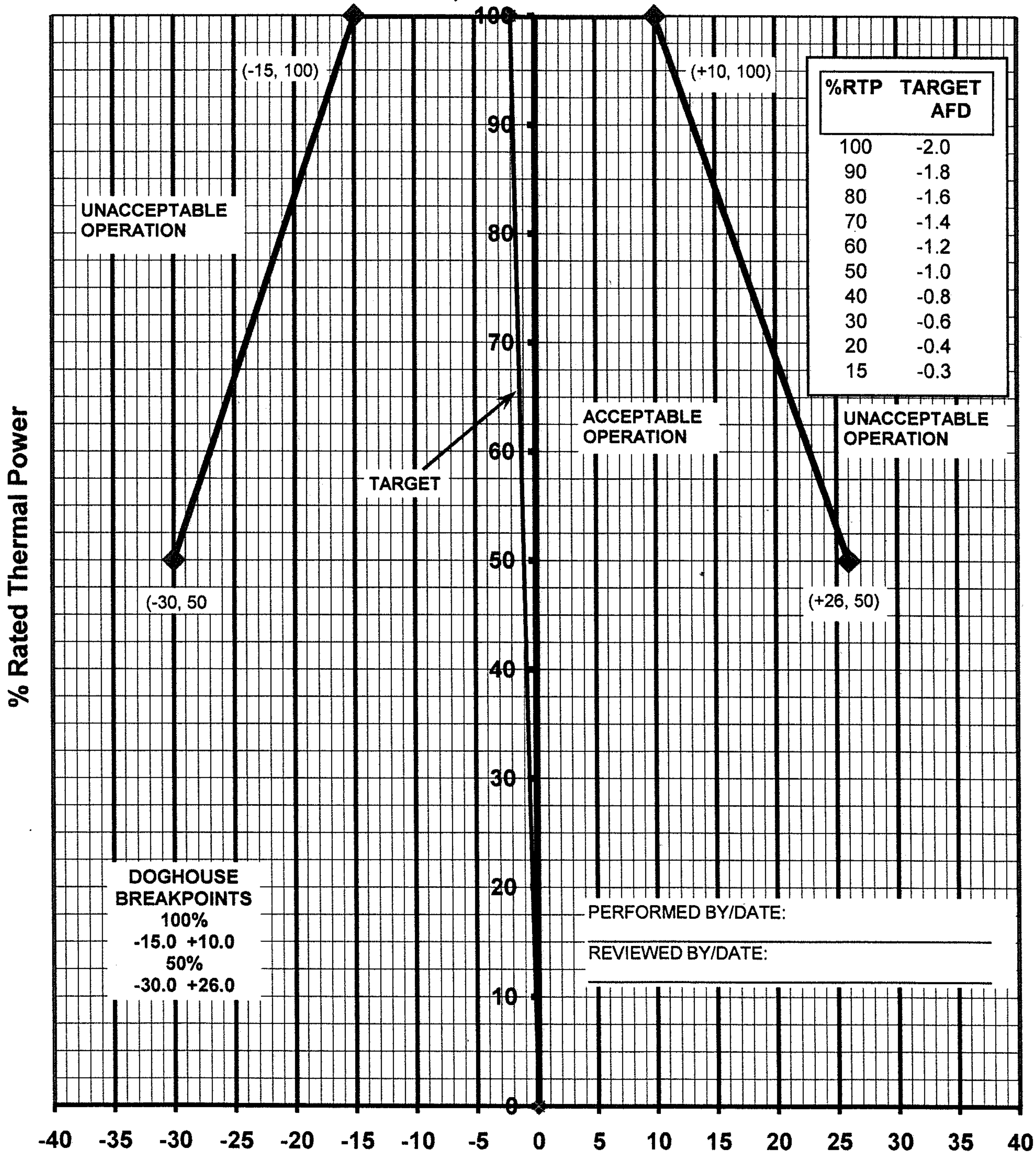
UNIT 1 CYCLE 17
10,000 MWD/MTU



SHEET 1 OF 1

Axial Flux Difference Limits As A Function Of Rated Thermal Power

UNIT 1 CYCLE 17
20,000 MWD/MTU



%RTP	TARGET AFD
100	-2.0
90	-1.8
80	-1.6
70	-1.4
60	-1.2
50	-1.0
40	-0.8
30	-0.6
20	-0.4
15	-0.3

SHEET 1 OF 1

82. 025AA2.04 001/1/1/RHR - LEAK LOC/C/A - 3.3/3.6/NEW/HL-18 NRC/SRO/KAJ

Initial conditions:

- Unit 1 RCS is in Mode 5.

Current conditions:

- The RHR Test Recirculation Valves to the RWST were left open following cavity drain down operations (1-1205-U6-027 and 1-1205-U4-226).
- ALB11-B06 PRZR LO LEVEL HTR CNTL OFF LTDN SECURED illuminates.
- PRZR level is lowering rapidly.

Which ONE of the following completes the following statement?

Closing HV-8716A/B, RHR Train A(B) Hot Leg Crossover Iso Vlvs, on the QMCB ____(1)____ isolate the leak.

and

based on the given conditions, an Emergency Action Level (EAL) threshold ____(2)____ been exceeded.

REFERENCE PROVIDED

A✓ (1) will
(2) has

B. (1) will
(2) has NOT

C. (1) will NOT
(2) has

D. (1) will NOT
(2) has NOT

025AA2.04 Loss of Residual Heat Removal System (RHRS)

Ability to determine or interpret the following as they apply to the Loss of Residual Heat Removal System: (CFR 43.5 / 45.13)

Location and isolability of leaks.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where an RCS leak is in progress while in Mode 5. The leak is due to the RHR to RWST recirculation line valves 1-1205-U6-027 and 1-1204-U4-226 being left open following a reactor cavity drain down evolution causing a loss of RCS inventory to the RWST. The candidate knows the leak is to the RWST and has to determine if closing HV-8716A/B RHR TRAIN A(B) HOT LEG CROSSOVER ISO VLVs will or will NOT isolate the leak. Closing these valves will isolate the leak.

The candidate also has to determine whether an EAL classification is required using NMP-110, GL03, Cold Initiating Condition Emergency Action Level Matrix - Modes 5, 6, and Defueled Only. The candidate has to determine from given plant conditions whether to use the Hot or the Cold Matrix.

The question is SRO only as the Vogtle specific LOIT objective for classification of emergency events is designated as an SRO Only objective.

DISTRACTOR ANALYSIS:

A. Correct. The RHR to RWST leak via the recirculation line can be isolated by closing the HV-8716A and 8716B valves.
NOUE (CU1) classification is required using the Cold Matrix.

B. Incorrect. 1st half of this choice is correct as the RHR to RWST leak via the recirculation line can be isolated by closing the HV-8716A and 8716B valves. However, with the given conditions NOUE (CU1) classification is required using the Cold Matrix, the 2nd half is wrong as an EAL classification is required.

C. Incorrect. 1st half of this choice is wrong as the RHR to RWST leak via the recirculation line can be isolated by closing the HV-8716A and 8716B valves. NOUE (CU1) classification is required using the Cold Matrix. The 2nd half of this choice is correct.

D. Incorrect. 1st half is wrong as the RHR to RWST leak via the recirculation line can be isolated by closing the HV-8716A and 8716B valves. With the given conditions NOUE (CU1) classification is required using the Cold Matrix, the 2nd half is wrong as an EAL classification is required.

REFERENCES:

1X4DB122, Residual Heat Removal System
13005-1, Reactor Coolant System and Refueling Cavity Draining, section 4.6.

The following reference will be provided to the operators for this exam.

NMP-EP-110, GL03, Cold Initiating Condition Emergency Action Level Matrix - Modes 5, 6, and Defueled Only

VEGP learning objectives:

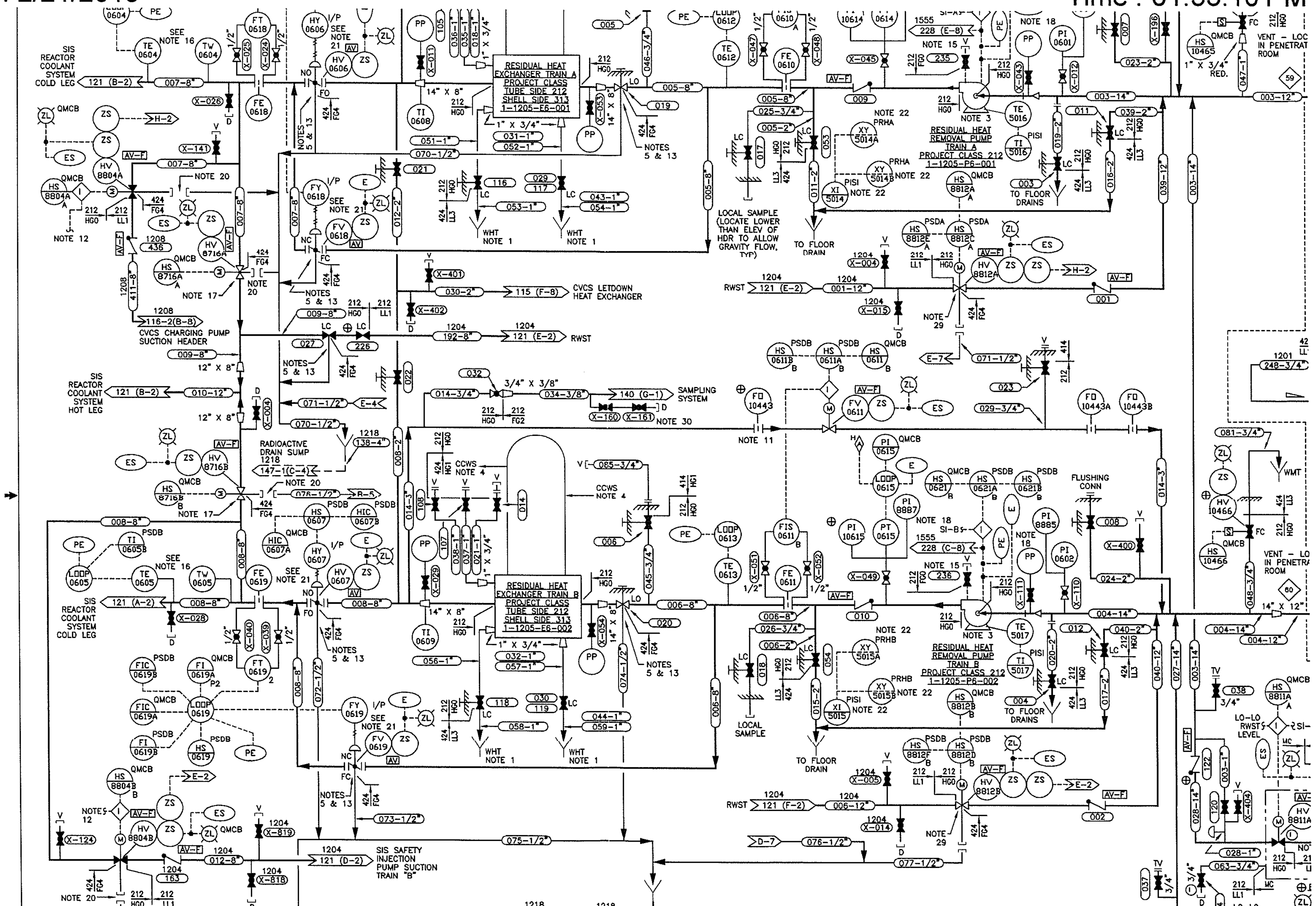
LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).

LO-LP-12101-13 Briefly describe the method and RHR system alignments for draining and filling of the reactor cavity

This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

Some examples of SRO only actions per section G, Fuel Handling Facilities and Procedures of 10CFR55.43(b)(7) Emergency Classifications.

Date: 2/21/2013 Time : 01:55:10 PM



NOTES:

1. FOR DRAIN HEADER INFORMATION SEE DWGS 1X4DB143, 144-1, 1X4DB145-5 AND 1X4DB183.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT: (1) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER, AND (2) EQUIPMENT MARKED ⊕.
3. FOR PUMP AND MOTOR COOLING WATER CONNECTIONS, SEE DWGS

10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON DWG CX4DG001. 51.0
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION

21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.

29. A SECTION OF THE LEAKOFF LINE HAS BEEN REMOVED AND BOTH ENDS CAPPED.
30. THESE ARE KEROTEST VALVES AND MAY NOT ALLOW REVERSE FLOW.

Title: C:\Documents and Settings\Administrator\Desktop\HL-18\Reference Material\P&ID\1X4DB122.cal

'ower to Essential Busses
Iinutes. (pg. 78)

:om transformers
IXRB resulting in loss of
wer to **BOTH** 1(2)AA02
ater than 15 minutes

generators supplying power
OR 1(2)BA03.

: of Required DC Power for
82)

Vital DC power to 125
1, CD1, AND DD1
ge indications less than 105

r to at least one DC bus
n the time of loss.

CU8 - Inadvertent Criticality
Modes 5 and 6 Only. (pg. 83)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

CU1 - RCS Leakage **Mode 5 Only.** (pg. 76)

1. Unable to establish or maintain pressurizer level greater than 17%.

CU2 - Unplanned Loss of RCS Inventory with Irradiated Fuel in the RPV **Mode 6 Only** (pg. 77)

1. UNPLANNED RCS level lowering below 194' (RPV flange) for greater than 15 minutes

OR

2. a. RPV level **CANNOT** be monitored

AND

b. A possible loss of RPV inventory may be occurring as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

CU4 - Unplanned Loss of
Capability with Irradiated
Modes 5 and 6 Only. (pg

NOTE The Emergency L
15 minutes has elapsed,
as soon as it is determinea
likely excee

1. An UNPLANNED even
exceeding 200°F.

OR

2. Loss of all RCS tempera
indication for greater tha

CU1

Initiating Condition

RCS Leakage.

Operating Mode Applicability: Cold Shutdown Only (Mode 5)

Threshold Values:

1. Unable to establish or maintain pressurizer level greater than 17%.

Basis:

This IC is included as a NOUE because it is considered to be a potential degradation of the level of safety of the plant. The inability to establish and maintain level is indicative of loss of RCS inventory. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

INITIALS

NOTE

This section is intended to be used to adjust or drain-down Refueling Cavity water level during refuelings with the Reactor Vessel Head removed and the Refueling Cavity flooded.

4.6 DRAINING THE REFUELING CAVITY TO RWST USING THE RHR SYSTEM FUELED OR DE FUELED AND UPPER INTERNALS REMOVED

NOTES

- With fuel in the vessel and water level in the refueling cavity less than 217 feet 0 inches elevation (23 feet above the vessel flange), both trains of the RHRS are required to be operable with one train operating.
- With fuel in the vessel, performance of this section assumes use of the RHR train that is operable, but not operating.
- Approximate Reactor Cavity level versus gallons:

  Cavity only approximately 11,272 gallons/foot
  Cavity and transfer canal approximately 12,600 gallons/foot.

CAUTIONS

- If Fuel Reconstitution is in progress or planned, Reactor Engineering should be contacted to assure that fuel transfer tube isolation is adequate for reactor cavity drain-down.
- If non-borated water is used during the cavity decontamination process, operators should be aware of the potential for a positive reactivity insertion.

4.6.1 IF this is to be a drain-down (to less than 207 feet elevation), **verify** adequate RCS level monitoring controls of either 12007-C, "Refueling Operations" or 12000-C, "Post Refueling Operations" have been completed and are in effect PRIOR to lowering level below 207 feet elevation. ________

INITIALS

4.6.2 **Verify** underwater lights in Reactor Cavity and Transfer Canal are off prior to commencing draindown below the level of the lights. ________

4.6.3 **Verify** Plant Computer is selected to the current MODE and **trend** RHR Pump parameters (Pts. J9623, 9624) for early detection of possible RHR Pump degradation due to vortexing. ________

4.6.4 IF Transfer Tube Gate Valve 1-1213-U6-086 is open, perform the following:

a. **Verify** the Transfer Canal Gate is closed and the Transfer Canal Gate Seal is inflated. ________

b. **Inspect** Spent Fuel Pool area for possible Siphon Pathways (hoses) which might inadvertently lower Spent Fuel Pool level during draining. ________

> NOTE
>
> Closure of transfer tube gate valve, 1-1213-U6-086 takes approximately 12 minutes.

c. IF draining to less than 217 feet, **post** an operator at 1-1213-U6-086 to **close** the valve IF indications are observed that the Transfer Canal Gate Seal is leaking. ________

4.6.5 **Align** the selected RHR train for startup on miniflow with the pump suction from the hot leg and **discharge** aligned to the RWST and cold legs as follows:

4.6.5.1 **Verify** open RHR PMP-A(B) TO COLD LEG 1&2(3&4) ISO VLV 1-HV-8809A(8809B) with 1HS-8809A(B), ________

INITIALS

Critical

4.6.5.2 **Close** both RHR TRAIN A and B TO HOT LEG CROSSOVER ISO valves:

- 1-HV-8716A 1HS-8716A ________

________ CV

- 1-HV-8716B 1HS-8716B ________

________ CV

4.6.5.3 **Close** RHR TO HL ISO VLV 1-HV-8840 with 1HS-8840. ________

4.6.5.4 **Verify** RHR HX TRAIN-A(B) BYPASS 1FIC-0618A(0619A) is in MAN and at 0% demand (closed). ________

4.6.5.5 **Close** RHR HX TRAIN-A(B) OUTLET 1HIC-0606A(0607A). ________

4.6.5.6 **Unlock** and **open** RHR TEST RECIRCULATION TO RWST:

- 1-1205-U6-027 ________
- 1-1205-U4-226 ________

4.6.5.7 **Place** RHR PUMP A(B) in PULL TO LOCK with 1HS-0620(0621). ________

4.6.5.8 **Close** RWST TO RHR PMP-A(B) SUCTION 1-HV-8812A(8812B) with 1HS-8812A(B). ________

83. 029EG2.4.18 001/1/1/ATWS - EP/C/A - 3.3/4.0/NEW/HL-18 NRC/SRO/AML

Initial conditions:

- Unit 1 is at 100% power at EOL.
- A total loss of feedwater occurs.
- The reactor can NOT be tripped.
- The crew enters 19211-C, "Response to Nuclear Power Generation/ATWT."
- The UO manually trips the Turbine.

Current conditions:

- The Shift Supervisor is performing the step to "Check Core Exit TCs - LESS THAN 1200°F."
- Core Exit TCs are 1208°F and lowering.

Which ONE of the following completes the following statement?

The EOP bases for tripping the main turbine during the event is to ___(1)___,

and

the Shift Supervisor is required to ___(2)___.

A.✓ (1) maintain Steam Generator inventory

(2) continue performing the actions of 19211-C

B. (1) maintain Steam Generator inventory

(2) Go to SACRG-1, "Severe Accident Control Room Guideline Initial Response"

C. (1) allow the RCS to heatup adding negative reactivity from MTC

(2) continue performing the actions of 19211-C

D. (1) allow the RCS to heatup adding negative reactivity from MTC

(2) Go to SACRG-1, "Severe Accident Control Room Guideline Initial Response"

029EG2.4.18 Anticipated Transient Without Scram

Knowledge of the specific bases for EOPs:
(CFR: 41.10 / 43.1 / 45.13)

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario where an ATWT exists due to a loss of feedwater flow. The candidate has to determine the bases for tripping the Turbine during the performance of 19211-C, Response to Nuclear Power Generation, ATWT per the Westinghouse Owners Group (WOG) guidance.

The candidate also has to determine whether continued performance of 19211-C or a transition to the SAMGs procedure SACRG-1 is appropriate.

This question is SRO only per the guidance of the NRC Clarification Guidance for SRO Only Questions Figure 2 Assessment and Selection of Procedures.

DISTRACTOR ANALYSIS:

A. Correct. The bases for tripping the Turbine during an ATWT due to a loss of feedwater is to maintain SG inventory. With CETCs over 1200°F and lowering, continued performance of 19211-C is required.

B. Incorrect. The 1st half of this choice is correct as the bases for tripping the Turbine during an ATWT due to a loss of feedwater is to maintain SG inventory. The 2nd half of this question is incorrect since with CETCs over 1200°F and lowering, continued performance of 19211-C is required. It is plausible that the candidate may think a transition to SACRG-1 is required with core exits > 1200°F and is the proper course of action if CETCs are rising.

C. Incorrect. The 1st half of this choice is incorrect. The bases for tripping the Turbine during an ATWT due to a loss of feedwater is to maintain SG inventory. It is plausible the candidate may think tripping the turbine is to allow RCS heatup to add negative reactivity as specified in step 23 RNO. Tripping the turbine will heat up the RCS and add negative reactivity but it is not the bases during a loss of FW ATWT. With CETCs over 1200°F and lowering, continued performance of 19211-C is required.

D. The 1st half of this choice is incorrect. The bases for tripping the Turbine during an ATWT due to a loss of feedwater is to maintain SG inventory. It is plausible the candidate may think tripping the turbine is to allow RCS heatup to add negative reactivity as specified in step 23 RNO. Tripping the turbine will heat up the RCS and add negative reactivity but it is not the bases during a loss of FW ATWT. The 2nd half of this question is incorrect since with CETCs over 1200°F and lowering, continued performance of 19211-C is required. It is plausible that the candidate may think a transition to SACRG-1 is required with core exits > 1200°F and is the proper course of action if CETCs are rising.

REFERENCES:

19211-C, Response to Nuclear Power Generation / ATWT
FR-S.1 WOG Background for Response to Nuclear Power Generation / ATWT

LO-LP-37041-08 State the major action categories of EOP 19211-C.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	

NOTE

This Functional Restoration Procedure should **NOT** be implemented if both 4160V AC emergency busses are de-energized; 19100-C should be entered.

CAUTION

RCPs should not be tripped with Reactor power greater than 5%.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Verify Reactor trip:	__1. Trip Reactor using both Reactor trip handswitches.
__• Rod Bottom Lights – LIT.	__IF Reactor NOT tripped, THEN insert Control Rods.
__• Reactor Trip and Bypass Breakers – OPEN.	
__• Neutron Flux – LOWERING.	
2. Verify Turbine trip:	
__a. All Turbine Stop Valves – CLOSED.	__a. Trip Turbine.
	__ IF Turbine will NOT trip, THEN run back Turbine.
	__ IF Turbine can NOT be run back, THEN close Main Steamline Isolation and Bypass Valves.

Step 2 continued on next page

STEP: Verify Turbine Trip

PURPOSE: To ensure that the turbine is tripped

BASIS:

The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require. For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory.

If the turbine will not trip, a turbine runback (manual decrease in load) at maximum rate will also reduce steam flow in a delayed manner. If the turbine stop valves cannot be closed by either trip or runback, the MSIVs should be closed.

ACTIONS:

- o Determine if all turbine stop valves are closed
- o Determine if turbine will not trip
- o Determine if turbine cannot be run back
- o Trip the turbine
- o Manually run back turbine
- o Close main steamline isolation and bypass valves

INSTRUMENTATION:

- o Turbine stop valve position indication
- o MSIVs and bypass valves position indication

CONTROL/EQUIPMENT:

- o Switches for turbine trip (e.g. manual trip buttons, overspeed test switch, EH control oil pump switches)
- o Controls to manually run back turbine
- o Switches to close MSIVs and bypass valves

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. Verify affected SG ARV(s) - CLOSED: __ PV-3000 (SG 1) __ PV-3010 (SG 2) __ PV-3020 (SG 3) __ PV-3030 (SG 4)	21. IF SG ARV(s) can NOT be closed, THEN locally unlock and close associated SG ARV INLET isolation valve: __ 1301-U4-136 (SG 1) __ 1301-U4-137 (SG 2) __ 1301-U4-138 (SG 3) __ 1301-U4-139 (SG 4)
__ 22. Check Core Exit TCs - LESS THAN 1200°F.	__ 22. IF Core Exit TCs greater than 1200°F and rising, THEN go to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

STEP: Check Core Exit TCs - LESS THAN 1200°F

PURPOSE: To ensure severe conditions do not exist that require a transition to the SAMGs

BASIS:

The Severe Accident Management Guidelines (SAMGs) are entered from the ERGs by the control room operators when core damage occurs. The ERG to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the ERGs to the SAMGs. The 1200°F criteria for transition from the ERGs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

If the operator enters this step and core exit TC temperatures are greater than 1200°F and increasing, the operator should transition to the SAMGs. This condition indicates that all attempts to restore core cooling have failed and core damage can not be prevented and the operator should go to the SAMGs.

If the operator enters this step and core exit TC temperatures are less than 1200°F or core exit TC temperatures are greater than 1200°F and decreasing, the operator will stay in the loop between Steps 4 and 15 in guideline FR-S.1 to continue efforts to emergency borate the RCS and check for sources of positive reactivity.

ACTIONS:

- o Determine if core exit TCs are less than 1200°F
- o Determine if core exit TCs are greater than 1200°F
- o Determine if core exit TCs are increasing

INSTRUMENTATION:

Core exit TC temperature indication

84. 034A1.02 001/1/2/FUEL HAND - CANAL LV/C/A - 2.9/3.7/NEW/HL-18 NRC/SRO/KAJ

Unit 1 is in a Refueling Outage:

- Core off-load is in progress.
- ALB05-E02 SPENT FUEL PIT LO LEVEL alarms in the Control Room.
- SFP level lowers to 22' 10" above the fuel due to a leak.

For the given conditions, which ONE of the following completes the following statement?

Per Tech Spec 3.9.7, "Refueling Cavity Water Level," immediate suspension of core off-load ___(1)___ required,

and

the Safety Analysis for the Bases of Tech Spec 3.9.7 limit is to ensure ___(2)___.

A. (1) is

(2) during all phases of spent fuel transfer the gamma dose rate at the surface of the water is 2.5 mrem or less.

B.✓ (1) is

(2) the radiological consequences of a fuel handling accident are within acceptable limits of 10 CFR 100

C. (1) is NOT

(2) during all phases of spent fuel transfer the gamma dose rate at the surface of the water is 2.5 mrem or less.

D. (1) is NOT

(2) the radiological consequences of a fuel handling accident are within acceptable limits of 10 CFR 100

034A1.02 Fuel Handling Equipment System (FHES)

Ability to predict and / or monitor changes in parameters to prevent exceeding design limits associated with the Fuel Handling System controls including:
(CFR 41.5 / 45.5)

Water level in the refueling canal.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where spent fuel pool low level alarm is received and the candidate has to determine whether suspension of core off-load is required. Off-load is required to be suspended as the water level is < 23 feet above the fuel.

The candidate also must determine the bases for the Refueling Cavity Water level in accordance with Tech Spec 3.9.7 "Refueling Cavity Water Level".

This question is SRO per the NRC Clarification Guidance for SRO-only Question Figure 1, (Tech Specs). Knowledge of TS Bases that is required to analyze TS required actions and terminology.

DISTRACTOR ANALYSIS:

A. Incorrect. The 1st half is correct as suspension of core off-load is required per the actions of Tech Spec 3.9.7. The 2nd half is plausible as the candidate may think the water level is to maintain this minimum dose rate, however, this is accomplished by the long spent fuel tool and limits on the FH Building and reactor cavity hoists.

B. Correct. Suspension of core off-load is required and the bases is to ensure the radiological consequences of a fuel handling event are within acceptable limits of 10CFR100.

C. Incorrect. The 1st half is incorrect as suspension of core off-load is required per the actions of Tech Spec 3.9.7. The 2nd half is plausible as the candidate may think the water level is to maintain this minimum dose rate, however, this is accomplished by the long spent fuel tool and limits on the FH Building and reactor cavity hoists.

D. Incorrect. The 1st half is incorrect as suspension of core off-load is required per the actions of Tech Spec 3.9.7. The 2nd half is correct as the bases is to ensure the radiological consequences of a fuel handling event are within acceptable limits of 10CFR100.

REFERENCES:

Tech Spec 3.9.7, Refueling Cavity Water Level
Tech Spec 3.9.7 Bases for Refueling Cavity Water Level
Tech Spec 3.7.15 Fuel Storage Pool Water Level
PTDB Tab # 8.3 Midloop Level Instrumentation
VEGP-FSAR page 9.1-38 (9.1.4.3.4 for Radiation Shielding)

VEGP learning objectives:

a. State the LCO.

b. State any one hour or less required actions.

LO-LP-39213-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.9 are exceeded.

b. The required actions for all section 3.9 LCOs.

LO-LP-39213-04 Describe the bases for any given Tech Spec in section 3.9.

LO-LP-39213-05 State the reason for the water level requirements over the vessel flange and the spent fuel pool.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 133 to be used in the accident analysis for iodine. This relates to the assumption that 99.25% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory, other than iodine 127 and 129, which are 30%, and iodine 131, which is 12% (Ref. 2).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 4 and 5).

(continued)

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Water Level

LCO 3.7.15 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.

ACTIONS

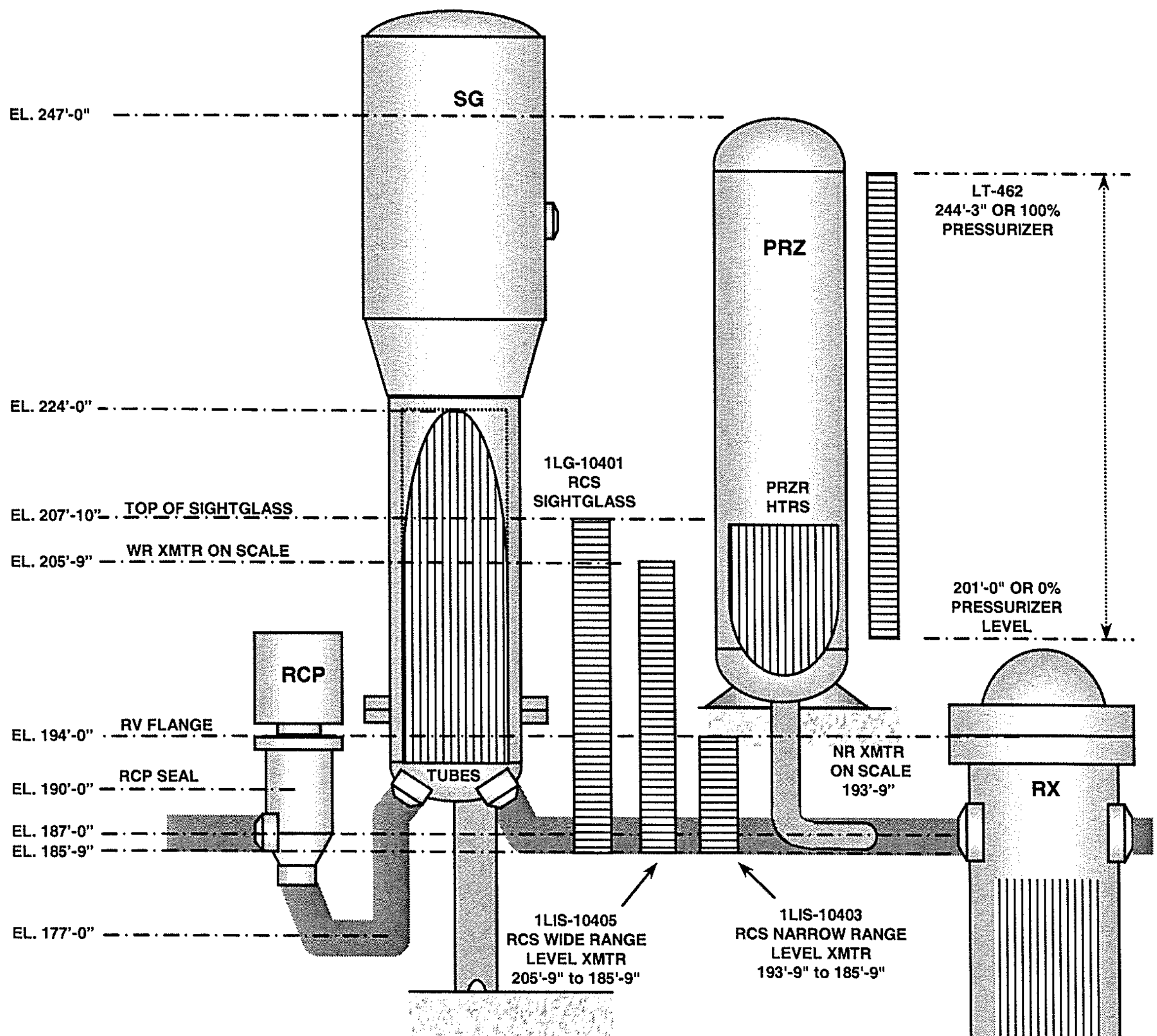
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies in the fuel storage pool.	Immediately

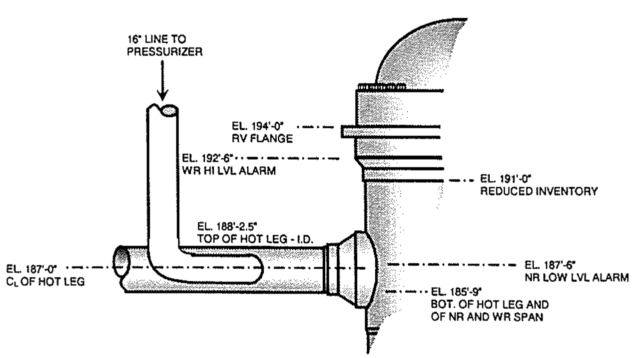
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

PICTORIAL AIDS

TAB 8.3
MID LOOP LEVEL INSTRUMENTATION: CRITICAL ELEVATIONS





Robert Rowland / 1-7-2003
Reviewed By Date

#84

tool as the WABA is raised and lowered. The gripper shaft interacts with the four comb assemblies by means of cables which raise and lower in opposing motion to the gripper shaft. The gripper shaft locks into the outer upper shaft of the tool. This prevents the WABA from being accidentally released while it is being transferred.

9.1.4.3.2 Seismic Considerations

The safety classifications for all fuel handling and storage equipment are listed in section 3.2. These safety classes provide criteria for the seismic design of the various components. Safety-related equipment is designed to withstand the forces of the operating basis earthquake (OBE) and SSE. For normal conditions plus OBE loadings, the resulting stresses are limited to allowable working stresses as defined in the ASME B&PV Code, Section III, Appendix XVII, for normal and upset conditions. For normal conditions plus SSE loadings, the stresses are limited to within the allowable values given by Subarticle NA 2110 for critical parts of the equipment which are required to maintain the capability of the equipment to perform its safety function. Permanent deformation is allowed for the loading combination, which includes the SSE to the extent that there is no loss of safety function.

For Safety Class 3 fuel handling and storage equipment, consideration is given to the OBE only insofar as failure of the Safety Class 3 equipment might adversely affect other safety-related equipment.

For nonnuclear safety equipment, design for the SSE is considered if failure might adversely affect safety-related equipment. Design for the OBE is considered if failure of the nonnuclear safety component might adversely affect safety-related equipment.

9.1.4.3.3 Containment Pressure Boundary Integrity

The fuel transfer tube which connects the refueling canal (inside the reactor containment) and the fuel storage area (outside the containment) is closed on the refueling canal side by a blind flange at all times except during refueling operations. Two seals are located around the periphery of the blind flange with leak-check provisions between them.

9.1.4.3.4 Radiation Shielding

During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 2.5 mrem/h or less. This is accomplished by maintaining a nominal 10 ft of water above the top of the active fuel height during all handling operations.

The two fuel handling devices used to lift spent fuel assemblies are the refueling machine and the fuel handling machine. The refueling machine contains positive stops which prevent the fuel assembly from being raised above a safe shielding height. The hoist on the fuel handling machine and the containment fuel storage area crane moves spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to within this safe shielding height. In the event that fuel is damaged and fuel fragments or pellets are collected on filters or other devices, the shielding requirements for the movement of the filters govern rather than the requirements on the submergence of spent fuel.

85. 039G2.2.25 001/2/1/MRSS - EQUIP CTRL/MEM - 3.2/4.2/MOD-LOIT/HL-18 NRC/SRO/TNT

Unit 1 is at 100% power.

Maintenance testing of the Main Steam Safety valves is in progress.

The following test results are obtained for the valve lift settings:

1PSV-3002	1205 psig
1PSV-3011	1210 psig
1PSV-3022	1160 psig

NO valve lift setting adjustments have been made.

Which one of the following completes the following statement?

The MAXIMUM Allowable Power Range Neutron Flux High Trip Setpoint per Tech Spec 3.7.1, Main Steam Safety Valves (MSSVs) is __(1)___

and

per the Tech Spec bases, the Main Steam Safety valves are designed to limit ___(2)___ during a full power Turbine trip without steam dump?

REFERENCE PROVIDED

A. (1) 71%

(2) reactor coolant pressure boundary Safety Limit to 110% of design pressure

B. (1) 51%

(2) reactor coolant pressure boundary Safety Limit to 110% of design pressure

C✓ (1) 71%

(2) secondary pressure to ≤ 110% design pressure

D. (1) 51%

(2) secondary pressure to ≤ 110% design pressure

039G2.2.25 Main and Reheat Steam System (MRSS)

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits:

(CFR 41.5 / 41.7 / 43.2)

K/A MATCH ANALYSIS:

The candidate must determine the correct bases per Technical Specification 3.7.1, Main Steam Safety Valves and the bases from the Safety Analysis for having the Main Steam Safety Valves.

The question is SRO only per the NRC Clarification Guidelines for SRO-only Questions, Figure 1 (Tech Specs), Knowledge of TS Bases that is required to analyze TS required actions and terminology.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. The first half of this choice is correct. The 2nd half of this choice is incorrect and is the bases for the PRZR Code Safeties, not the Main Steam Safety Valves.

B. Incorrect. First part incorrect. PSV 3011 is out of tolerance high at 1210 psig (setpoint 1185 psig with a band of 1149.5 psig to 1208.7 psig) and PSV 3022 is out of tolerance low at 1160 psig (1200 psig with a band of 1164 psig to 1224 psig) Candidate may not apply the proper setpoint or not apply the low tolerance thinking early is conservative. A tolerance low is showing degradation (drift) and may open late the next demand.

The 2nd half of this choice is incorrect and is the bases for the PRZR Code Safeties, not the Main Steam Safety Valves.

C. Correct. The first half of this choice is correct. The 2nd half of the choice is correct to limit secondary design pressure to ≤110 design pressure during a full power Turbine trip without steam dump.

D. Incorrect. First part incorrect. PSV 3011 is out of tolerance high at 1210 psig (setpoint 1185 psig with a band of 1149.5 psig to 1208.7 psig) and PSV 3022 is out of tolerance low at 1160 psig (1200 psig with a band of 1164 psig to 1224 psig) Candidate may not apply the proper setpoint or not apply the low tolerance thinking early is conservative. A tolerance low is showing degradation (drift) and may open late the next demand.

The 2nd half of the choice is correct to limit secondary design pressure to ≤110 design pressure during a full power Turbine trip without steam dump.

REFERENCES:

Technical Specification 3.7.1 and Bases, Main Steam Safety Valves and Table 3.7.1-1 and 3.7.1-2

Technical Specification 3.4.10 and Bases, Pressurizer Safety Valves

VEGP learning objectives:

LO-LP-39211-01 For any given item in section 3.7 of Tech Specs, be able to:

a. State the LCO.

b. State any one hour or less required actions.

LO-LP-39211-04 Describe the bases for any given Tech Spec in section 3.7.

LO-LP-39211-05 List the five relief setpoints of the steam generator safety valves.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3 (Ref. 1). The actual MSSV capacity is 114% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis requirement is that secondary system pressure is limited to 110% of design pressure which is specified in Reference 2. The actual design basis applied for the MSSVs comes from Reference 6 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 105% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

(continued)

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Range Neutron Flux High Trip Setpoint with Inoperable Main Steam Safety Valves

NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH TRIP SETPOINT (% RTP)
1	71
2	51
3	31

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

	VALVE NUMBER				LIFT SETTING (psig + 2%, -3%)
	STEAM GENERATOR				
	#1	#2	#3	#4	
1.	PSV3001	PSV3011	PSV3021	PSV3031	1185 psig
2.	PSV3002	PSV3012	PSV3022	PSV3032	1200 psig
3.	PSV3003	PSV3013	PSV3023	PSV3033	1210 psig
4.	PSV3004	PSV3014	PSV3024	PSV3034	1220 psig
5.	PSV3005	PSV3015	PSV3025	PSV3035	1235 psig

BASES

BACKGROUND (continued)

MODES 1, 2, 3, and MODE 4 with all RCS cold leg temperatures > the COPS arming temperature specified in the PTLR. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries;

f. Locked rotor; and

g. Feedwater line break.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, e, and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

(continued)

86. 056AG2.4.45 001/1/1/LOSP- EP/C/A-4.1/4.3/NEW/HL-18 NRC/SRO/

At 10:00:

- Unit 1 is at 220°F

At 10:05 the following alarms illuminate:

- ALB32-D02, RESV AUX XFMR 1NXRA HI SIDE PHOC LOR TRIP
- ALB32-E02, RESV AUX XFMR 1NXRB HI SIDE PHOC LOR TRIP

- ALB35-A10, DG1A TRIP OVERSPEED
- ALB35-F10, DG1A EMERGENCY START
- ALB36-A01, 4160V SWGR 1AA02 TROUBLE

- ALB37-A01, 4160V SWGR 1BA03 TROUBLE alarms, then subsequently clears.
- ALB38-F10, DG1B EMERGENCY START

Current time is 10:25:

Based on the current time, which ONE of the following is the correct Emergency Classification required to be declared?

REFERENCE PROVIDED

A. Alert Emergency (CA3)

B✓ Alert Emergency (SA5)

C. Notification of Unusual Event (SU1)

D. Notification of Unusual Event (CU3)

056AG2.4.45 Loss of Offsite Power

Ability to prioritize and interpret the significance of each annunciator or alarm:
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

<u>K/A MATCH ANALYSIS:</u>

The candidate is given various alarms and indications associated with the electrical distribution system. The candidate has to analyze the alarms to determine the plant status and determine the correct emergency classification, there is a time given when the event occurred that will also play into the classification.

The question is SRO only due to the Vogtle specific objective for Classification of an Emergency is an SRO only objective.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. CA3 is a Cold Matrix classification, the plant is in Mode 4, not Mode 5 or 6. The mode was NOT stated in the question but just an RCS temperature given to increase the plausibility the candidate may select the wrong classification matrix. If the candidate selects the wrong matrix with the given alarms, it is plausible he could misinterpret the event and classify wrong. With the multiple alarms and indications, this can easily occur.

B. Correct. SA5 is the correct classification using the Hot Matrix, this is still a difficult determination with the multiple annunciator windows illuminated. The plant is only one failure away from a total plant blackout in this condition but the candidate has to determine this and correlate the event has been ongoing for > 15 minutes.

C. Incorrect. SU1 is a Hot Matrix classification. The plant is only one failure away from a total plant blackout in this condition but the candidate has to determine this and correlate the event has been ongoing for > 15 minutes. This choice is very plausible as the only difference between this and SA5 is that both diesels have to be carrying the buses to classify as SU1 versus 1 DG as in the correct choice. This is a difficult determination with the mulitiple annuciators illuminated.

D. Incorrect. CA3 is a Cold Matrix classification, the plant is in Mode 4, not Mode 5 or 6. The mode was NOT stated in the question but just an RCS temperature given to increase the plausiblity the candidate may select the wrong classification matrix. IF, the plant were in Mode 5 and the Cold Matrix required to be used, this choice would then be correct.

REFERENCES:

The following references will be provided to the candidates during the exam.

NMP-EP-110, GL03, Cold Initiating Condition Emergency Action Level Matrix - Modes 5, 6, and Defueled Only

NMP-EP-110, GL03, Hot Initiating Condition Emergency Action Level Matrix - Modes 1, 2, 3, and 4 Only

VEGP learning objectives:

LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).

This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

	SS3 - Loss of All Vital DC Power (pg. 67)	**SS4** - Complete Loss of Heat Re 68)
	1. Loss of Vital DC power to 125 VDC Buses AD1, BD1, CD1, **AND** DD1 indicated by bus voltage indications less than 105 VDC for greater than 15 minutes.	***NOTE:*** *Heat Sink CSF should NO total AFW flow is less than 570 gp*
		1. Complete Loss of Heat F indicated by: a. Core Cooling CSF - ORAN **AND** b. Heat Sink CSF - RED
otential Loss of **EITHER** Barrier (pg. 39)	**SA5** - AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout. (pg. 64)	**SA2** - Failure of RPS Instrument Initiate an Automatic Reactor Tri Setpoint Has Been Exceeded AN Successful. Modes 1, 2, 3 Only (
Matrix		***NOTE*** *A successful manual trip declaration is any action taken fr Board (MCB) that rapidly inserts can be accomplished by tripping Reactor Trip switches on the MC*
	1. a. AC power capability to 1(2)AA02 **AND** 1(2)BA03 reduced to a single power source for greater than 15 minutes **AND** b. ANY additional single failure will result in station blackout.	1. Automatic trip was **NOT** suc was successful as indicated by a. An automatic Reactor I setpoint exceeded **AND** b. An automatic reactor trip di **AND** c. A successful manual trip oc Room (NOTE)

SA5

Initiating Condition

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value:

1. a. AC power capability to 1(2)AA02 **AND** 1(2)BA03 reduced to a single power source for greater than 15 minutes

   AND

   b. ANY additional single failure will result in station blackout.

Basis:

This IC and the associated Threshold Values are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the SAT, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power.

87. 062G2.2.44 001/2/1/AC ELEC - EQUIP/C/A - 4.2/4.4/NEW/HL-18 NRC/SRO/TNT

Given the following plant alarms and indications:

- All Channel 1 Trip Status lights are illuminated.
- ALB34-E02, INVERTERS 1AD1I1 1AD1I11 TROUBLE
- ALB34-E03, 120V AC PANELS 1AY1A 1AY2A TROUBLE

Which ONE of the following answers the following questions?

(1) The given indications were caused by the loss of which electrical equipment,

and

(2) when restored to service, when would the associated 120V AC Panel be considered OPERABLE per Tech Spec 3.8.9, "Distribution Systems - Operating"?

A. (1) 1AD1I11 and 1AY2A

(2) 1AY2A - The bus is energized at its proper voltage from the associated inverter only.

B. (1) 1AD1I11 and 1AY2A

(2) 1AY2A - The bus is energized at its proper voltage from the associated inverter or regulating transformer.

C. (1) 1AD1I1 and 1AY1A

(2) 1AY1A - The bus is energized at its proper voltage from the associated inverter only.

D✓ (1) 1AD1I1 and 1AY1A

(2) 1AY1A - The bus is energized at its proper voltage from the associated inverter or regulating transformer.

062G2.2.44 A.C. Electrical Distribution

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions:
(CFR: 41.5 / 43.5 / 45.12)

<u>K/A MATCH ANALYSIS:</u>

120V AC Distribution system with given alarms and indications. The candidate also has to determine the components which as a minimum are required to be returned to service to call the 120V Vital Bus OPERABLE.

This question is SRO per the NRC Clarification Guidance for SRO-only Questions. Figure 1, (Tech Specs). Knowledge of TS Bases that is required to analyze TS required actions and terminology.

DISTRACTOR ANALYSIS:

A. Incorrect. The1st half is wrong as this is the incorrect inverter and 120V AC distribution panel. This can be determined due to the Channel 1 trip status lights but is the only indication that differentiates which bus has failed. The 2nd half is incorrect, the bus is operable from either the inverter or the 120V regulated transformer, not just from the inverter per the bases of Tech Spec 3.8.9.

B. Incorrect. The1st half is wrong as this is the incorrect inverter and 120V AC distribution panel. This can be determined due to the Channel 1 trip status lights but is the only indication that differentiates which bus has failed. The 2nd half of the choice is correct, the bus is operable from either the inverter or the 120V regulated transformer per the bases of Tech Spec 3.8.9.

C. Incorrect. The first half is correct, this is the correct inverter and 120V AC distribution panel. This can be determined due to the Channel 1 trip status lights but is the only indication that differentiates which bus has failed. The 2nd half is incorrect, the bus is operable from either the inverter or the 120V regulated transformer, not just from the inverter per the bases of Tech Spec 3.8.9.

D. Correct. The first half is correct, this is the correct inverter and 120V AC distribution panel. This can be determined due to the Channel 1 trip status lights but is the only indication that differentiates which bus has failed. The 2nd half of the choice is correct, the bus is operable from either the inverter or the 120V regulated transformer per the bases of Tech Spec 3.8.9.

REFERENCES:

17034-1, Window E02, INVERTERS 1AD1I1 1AD1I11 TROUBLE
17034-1, Window E03, 120V AC PANELS 1AY1A 1AY2A TROUBLE
18032-1, Loss of 120V AC Instrument Power, Section A Symptoms for Loss of Vital Instrument Panel 1AY1A (CB-B52)
Tech Spec 3.8.9 Bases, Distribution Systems - Operating

VEGP learning objectives:

LO-LP-39212-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.8 are exceeded.

a. Whether any Tech Spec LCOs of section 3.8 are exceeded.

b. The required actions for all section 3.8 LCOs.

LO-LP-39212-04 Describe the bases for any given Tech Spec in section 3.8.

LO-LP-60324-01 Given the appropriate plant drawings, logics, and/or procedures, describe how the plant will respond to a loss of the following 120VAC instrument panels:

a. 1AY1A
b. 1AY2A

HL-18 NRC Exam 2013-301 Examination KEY

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

WINDOW E02

ORIGIN	SETPOINT	INVERTERS 1AD1I1 1AD1I11 TROUBLE
1AD1I1 Prot Rly 1AD1I11 Prot Rly	Not Applicable	

1.0 **PROBABLE CAUSE**

1. 7.5 kVA Inverter 1AD1I1:
   a. Input DC loss.
   b. High DC voltage.
   c. Loss of AC output.
2. 10 kVA Inverter 1AD1I11:
   a. Low DC Voltage.
   b. High DC Voltage.
   c. Low AC Output Voltage.
   d. Inverter Failure.
   e. High AC Output Voltage.
   f. Loss of 125V DC power from switchgear.

2.0 **AUTOMATIC ACTIONS**

1. NONE.
2. An overload condition will trip the Output Breaker for 1AD1I1.

3.0 **INITIAL OPERATOR ACTIONS**

IF an inverter has tripped, **initiate** 18032-1, "Loss Of 120V AC Instrument Power."

WINDOW E02
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF an inverter has NOT tripped, **dispatch** an operator to Inverter 1AD1I1 (Control Building) and 1AD1I11 (Auxiliary Building) to determine alarm cause.

   a. IF affected inverter AC output voltage is greater than 132V AC as determined by observing the voltmeter on the affected 120V AC Distribution Panel (1AY1A or 1AY2A), **perform** the following:

      (1) **Open** the INVERTER OUTPUT Breaker.

      (2) **Open** the DC INPUT Breaker,.

      (3) **Initiate** 18032-1, "Loss Of 120V AC Instrument Power."

   b. **Take** actions as necessary to clear the alarm condition.

   c. IF alarm continues, **consider** placing the affected bus on the maintenance feed and removing the inverter from service per 13431-1, "120V AC 1E Vital Instrument Distribution System."

   d. IF a fault is indicated, **initiate** maintenance as required.

2. **Refer to** Technical Specification LCO 3.8.7 and 3.8.8.

5.0 COMPENSATORY OPERATOR ACTIONS

1. **Initiate** maintenance to correct problem (i.e., restore alarm).

2. IF after three days the alarm has NOT been restored, **initiate** a Temporary Modification per NMP-ES-054, "Temporary Modifications" to clear the bad input(s). **Record** this action required on Figure 5 of 10018-C, "Annunciator Control."

END OF SUB-PROCEDURE

REFERENCES: 1X6AR07-2, 1X6AR07-24, 1X3AQ03-58

WINDOW E03

ORIGIN	SETPOINT	
1AY1A Prot Rly 1AY2A Prot Rly	Not Applicable	120V AC PANELS 1AY1A 1AY2A TROUBLE

1.0 **PROBABLE CAUSE**

1. One of the breakers on Panel 1AY1A or 1AY2A tripped.
2. Bus ground fault.
3. Panel undervoltage.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Check for associated alarms and indications AND IF loss of voltage is indicated, **initiate** 18032-1, "Loss Of 120V AC Instrument Power."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. If necessary, **dispatch** an operator to Panel IAYIA/1AY2A to check for:
   a. Ground fault indication.
   b. Tripped breakers.
   c. Other abnormal conditions.
2. IF alarm is due to 1AY2A breaker tripping or loss of voltage:
   a. **Determine** what loads are affected.
   b. IF panel voltage is lost or breaker 1AY2A-09 is tripped, **check** at QESF to determine if a manual bypass exists for any Channel 2 detector.

WINDOW E03
(Continued)

c. **Refer to** Technical Specification LCO 3.3.7 and Technical Requirement Manual TR 13.3.6 for operability requirements for Control Room and Fuel Handling Building radiation detection instrumentation.

d. **Notify** Maintenance and **return** affected equipment to service once the cause has been corrected.

3. IF alarm is due to 1AY1A breaker tripping:

a. **Determine** what loads are affected,

b. **Notify** Maintenance and **return** affected equipment to service once the cause has been corrected.

4. IF alarm is due to a ground fault:

a. Coordinating with Electrical Department, and using "120V AC/125V DC Panel Load Data Base" 1X3D-AA-M01C, **determine** which circuits can be momentarily de-energized by evaluating effect on plant systems.

b. Momentarily **de-energize** selected circuits to locate the source of the ground.

c. **Initiate** maintenance to clear the ground.

5. **Refer to** Technical Specification LCO 3.8.9 and 3.8.10.

5.0 COMPENSATORY OPERATOR ACTIONS

1. **Initiate** maintenance to correct problem (i.e., restore alarm).

2. IF after three days the alarm has NOT been restored, **initiate** a Temporary Modification per NMP-ES-054, "Temporary Modifications" to clear the bad input(s). **Record** this action required on Figure 5 of 10018-C, "Annunciator Control."

END OF SUB-PROCEDURE

REFERENCES: IX3D-AA-G02A, 1X3D-AA-G02C, 1X3AF01-3, 1X3AF01-162

BASES

LCO (continued)

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. OPERABLE DC electrical subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage or Class 1E regulating transformer.

In addition, tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable with the exceptions stated below. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

The LCO is modified by a Note that allows an exception to the OPERABILITY requirement that all tie breakers must be open. This exception is provided for the sole purpose of facilitating the transfer of preferred offsite power sources independent of DG operation. The 4160 V ESF buses may be manually connected within the unit by tie breakers and fed from one offsite power source provided the following precautions and limitations are followed:

1. Either one of the RATs, but not the SAT, may be utilized as the single offsite power source for both 4160 V ESF buses during the transfer evolution;

2. No additional nonsafety related 4160 V loads, other than those normally fed from 4160 V ESF buses 1/2AA02 and 1/2BA03, shall be manually connected to the RAT while the buses are interconnected;

(continued)

BASES

ACTIONS

A.1 (continued)

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one or more AC vital buses inoperable and the remaining OPERABLE AC vital buses capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition, overall reliability is reduced. An additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter with DC power available to the inverter or the Class 1E regulating transformer.

Condition B represents one or more AC vital buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptable power.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to

(continued)

SECTION A. LOSS OF VITAL INSTRUMENT PANEL 1AY1A (CB-B52)

- All Channel I trip status lights (except IR P-6, CNMT HI-3 PRESS, and RWST LO-LO LEVEL) lit.
- Simultaneous loss of Source and Intermediate Range Channel N-31/35 and Power Range Channel N-41.
- ALB34-E03 120V AC PANELS 1AY1A 1AY2A TROUBLE
- ALB34-E02 INVERTERS 1AD1I1 1AD1I11 TROUBLE

SECTION B. LOSS OF VITAL INSTRUMENT PANEL 1AY2A (AB-118)

- ALB36-A04 SEQ A TROUBLE
- ALB34-E02 INVERTERS 1AD1I1 1AD1I11 TROUBLE
- ALB34-E03 120V AC PANELS 1AY1A 1AY2A TROUBLE
- Loss of DRMS safety related display.
- Loss of Train A system status monitoring panel.
- CVI actuation.

SECTION C. LOSS OF VITAL INSTRUMENT PANEL 1BY1B (CB-B47)

- All Channel II trip status lights (except IR P-6, CNMT HI-3 PRESS, and RWST LO-LO LEVEL) lit.
- Simultaneous loss of Source and Intermediate Range Channel N-32/36 and Power Range Channel N-42.
- ALB34-C01 120V AC PANELS 1BY1B 1BY2B TROUBLE
- ALB34-C02 INVERTERS 1BD1I2 1BD1I12 TROUBLE

88. 063A2.02 001/2/1/DC - LOSS VENT/C/A - 2.3/3.1/MOD-LOIT/HL-18 NRC/SRO/TNT

Given the following plant conditions:

- Battery 1AD1B has been placed on Equalize Charge per 13405-1, "125 VDC 1E Electrical Distribution System."
- ALB50-A07 BAT RM TRN A SPLY FAN LO AIR FLOW illuminates.
- No additional operator action has been performed.

Based on the given conditions, which ONE of the following is the primary impact on the Battery charging operation, and the correct action(s) to perform to mitigate the consequences of the condition?

A.✓ Explosive Hydrogen gas could accumulate in the Battery Room.

Battery room doors must be propped open. A "Repositioned Door Form" must be posted per 00310-C, "Standard For Use of Doors."

B. Explosive Hydrogen gas could accumulate in the Battery Room.

Control Building Temperature Monitoring must be established per 14915-1, "Special Conditions Surveillance Logs."

C. Battery Room temperature exceeding Tech Spec limits.

Battery room doors must be propped open. A "Repositioned Door Form" must be posted per 00310-C, "Standard For Use of Doors."

D. Battery Room temperature exceeding Tech Spec limits.

Control Building Temperature Monitoring must be established per 14915-1, "Special Conditions Surveillance Logs."

063A2.02 D.C. Electrical Distribution System

Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.13):

Loss of ventilation during battery charging

<u>K/A MATCH ANALYSIS:</u>

The question is presented a plausible scenario where a 1E battery room supply fan trips while the batteries are on equalize charge. The candidate has to determine the primary impact on battery charging operation and corrective actions to mitigate the consequences of the event.

ANSWER / DISTRACTOR ANALYSIS:

A. Correct. Both halves of the choice are correct. The primary concern without the battery room ventilation fans running is hydrogen buildup. The battery rooms are required to be propped open per 00310-C, Standard for Use of Doors.

B. Incorrect. 1st half of the question is correct. The primary concern without the battery room ventilation fans running is hydrogen buildup. 14915-1 would be a plausible choice as the rooms will heat up but is not required per procedure.

C. Incorrect. 1st half of the question is incorrect. The primary concern without the battery room ventilation fans running is hydrogen buildup. Battery rooms exceeding the Tech Spec limit will be a longer term concern. The battery rooms are required to be propped open per 00310-C, Standard for Use of Doors.

D. Incorrect. 1st half of the question is incorrect. The primary concern without the battery room ventilation fans running is hydrogen buildup. Temperature monitoring per OSP-14915-1 would be a plausible choice if the candidate feels battery room temperature is the primary concern.

REFERENCES:

00310-C, "Standard For Use of Doors"
14915-1, "Special Conditions Surveillance Logs"
13302-1, "Control Building ESF Ventilation Systems"
13405-1, "125 VDC 1E Electrical Distribution System"
ALB50-A07 BAT RM TRN A SPLY FAN LO AIR FLOW

VEGP learning objectives:

LO-LP-41201-01 Briefly describe the principle of operation of a lead acid wet cell. Include both charging and discharging.

LO-LP-63310-05 Relate the actions taken by C&T when a plant door is mispositioned.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 Do not smoke, use open flames, or operate space heaters in the vicinity of the Storage Batteries. ________

2.1.2 Battery Room Ventilation Systems should be in operation to limit the buildup of hydrogen in the Battery Rooms. ________

2.1.3 If Battery Room Ventilation System is not available, verify doors to Battery Room propped open per 00310-C. ________

2.1.4 In Mode 5 or 6 when a battery must be removed from service for long periods such as for testing, it is preferable to transfer the 120V Vital busses to their regulated power supply, provided the Inverter is not required per Technical Specification 3.8.8. Transferring the vital busses to the regulated source will reduce the potential for power losses due to 125V DC bus instabilities. (Tech Spec 3.8.7 and 3.8.8) ________

2.1.5 The Control Room should be notified that alarms associated with battery chargers should be expected when starting up, shutting down, or swapping battery chargers. ________

2.2 LIMITATIONS

2.2.1 The 125V DC 1E Electrical Busses shall be operable in Modes, 1, 2, 3 and 4 per Technical Specification LCO 3.8.4 and LCO 3.8.9. ________

2.2.2 The 125V DC 1E Electrical Busses shall be operable in Modes 5 and 6 per Technical Specification LCO 3.8.5 and LCO 3.8.10. ________

2.2.3 The DC Input Breaker to the 25kVA Inverters cannot be closed until the Inverter Internal Capacitor Bank has been charged. ________

2.2.4 The Battery Charger 480V AC input voltage shall be 480V AC ±10% (432V-528V). ________

2.2.5 If the electrical bus must be energized by the battery chargers alone (without battery breaker closed in), only one charger should be energized to supply the bus. ________

2.2.6 A 72-hour equalizing charge should be performed every six months on all 1E batteries. ________

1.0 PURPOSE

1.1 The purpose of this procedure is to ensure that locked doors and doors designed for a specific purpose, are properly used by all individuals and that these doors be maintained in good working condition. Specific purpose doors include doors such as those for fire protection, flood protection, tornado protection, physical security, high energy line break protection, and doors to ensure proper operation of HVAC systems.

1.2 Doors that are not described above are not controlled by this procedure

2.0 DEFINITIONS

None

3.0 RESPONSIBILITIES

3.1 INDIVIDUALS

It is the responsibility of all individuals (SNC personnel and contractors) to properly use doors, not to abuse them. Proper use involves ensuring the door is properly positioned after passing through it. **For most doors** this involves verifying the door is closed and latched. Report doors that are broken or not working properly to the dispatcher (Ext. 3333). By reporting these doors proper compensatory measures can be taken and appropriate repairs can be made to them as well. (2002201174)

3.2 RESPONSIBLE INDIVIDUAL

3.2.1 Verifying that all doors controlled by this procedure have a Repositioned Door Form (Figure 1) posted when a door is out of position for any reason other than normal ingress or egress.

3.2.2 Obtaining an engineering evaluation if required and annotated on Figure 1.

3.2.3 Verifying that repositioned doors are returned to normal configuration as soon a possible.

3.3 SHIFT SUPPORT SUPERVISOR (SSS)

Maintain awareness of the doors that are propped opened for work through the information provided by Figure 1 and assess any impact caused by multiple openings. The SSS will maintain copies of Figure 1 for doors that have been propped open and will discard forms when notified that the work is completed.

REPOSITIONED DOOR FORM

Door No.:________________________

Door Type: HELBA ☐ Tornado ☐ Flood ☐ HVAC pressure boundary ☐
Other: ☐ ______________________________

Date Door Repositioned/Blocked: ____________
Estimated time the door will remain repositioned (day, hours, minutes, etc): ____________

Reason for Repositioning/Blocking door (include WO, DCP, TM, CRF number etc.): ____________

- Will the door/area be unattended with the door propped open at any time? Yes ☐ No ☐
- Will the door be propped open in such a way that it can not be closed and secured quickly? Yes ☐ No ☐
- Is the door a Control Room pressure boundary door? This includes doors 12111L1154, 12111L1155, 12111L1173, 12111L1175, 12111L1176 and doors 22111L1129 and 22111L1130 Yes ☐ No ☐

If any question is answered YES an engineering evaluation form is required.

Engineering Evaluation RER # ______________ Date: ____________

Comments: (include compensatory actions, Mode restrains, etc.)

Responsible Individual: ______________________________ Phone: ____________
Print Name

SSS/SS Approval:______________________________ Date: ____________
Signature

- Post this form on the door or in a conspicuous place near the door.
- When work is complete, return the door to its proper position, **sign and date the form, and return the form to C&T**.

Door restored to correct position: ______________________________ **Date:** ____________
Responsible Individual Signature

Figure 1 (Example)

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 The Control Building Safety Features Battery Room Exhaust Fans should be running any time the batteries are energized to prevent the accumulation of Hydrogen in the Battery Rooms.

2.1.2 If power is lost to Safety Features Battery Room Exhaust Fans and cannot be restored in a timely manner, the Maintenance Department should be directed to provide portable ventilation within 48 hours to prevent hydrogen buildup.

2.1.3 Each ventilation system should be operated as necessary to maintain the room temperature of the areas being served below the temperature alarm setpoints.

2.1.4 Unless Emergency conditions exist, the Chemistry Foreman should be notified prior to initiating Smoke Purge Mode in the Control Building. This is necessary to determine the presence of any airborne activity. Based on the sample results, a Batch Release Permit may be required. If an emergency exists, the Chemistry Foreman shall be notified as soon as possible.

DATA SHEET 13 Sheet 1 of 1

CONTROL BUILDING SHIFT AREA TEMPERATURE LOG

METHOD OF VERIFICATION	ROOM NO.	INDICATION DAY	INDICATION NIGHT	LIMIT(S)
AREA TEMPERATURE SHALL BE MAINTAINED WITHIN LIMITS	A055			100
	B065			100
	B074			100
VERIFY TEMPERATURE	B078			100
COMPLETED BY (INITIALS)				
SS REVIEWED BY (INITIALS)				
RECORD INSTRUMENT INFORMATION BELOW:				
INSTRUMENT ID NO.				
CAL DUE DATE				

NOTES:

1. TEMPERATURE INDICATION IS OBTAINED FROM HAND-HELD TEST EQUIPMENT TAKEN WAIST-HIGH AT "TEMP RDS" SIGN ONCE PER 12 HOURS AS PRESCRIBED.
2. IF ANY TEMPERATURE LIMIT IS NOT SATISFIED, IMMEDIATELY NOTIFY THE SS. IF THE SS DETERMINES A COMPONENT' OR SYSTEM INOPERABLE, INITIATE ACTION AS REQUIRED.
3. IF ONE OR MORE AREA TEMPERATURE LIMITS ARE EXCEEDED FOR MORE THAN 8 HOURS, NOTIFY THE MANAGER OF ENGINEERING SYSTEMS TO PREPARE AN ANALYSIS WITHIN 30 DAYS TO DETERMINE THE EFFECTS ON EQUIPMENT OPERABILITY AND QUALIFIED LIFE.
4. IF ONE OR MORE AREAS EXCEED 150°F, WITHIN 12 HOURS EITHER:
(A) RESTORE TEMPERATURE(S) WITHIN THE LIMITS AND PREPARE EVALUATION DESCRIBED IN 3. ABOVE OR
(B) DECLARE THE EQUIPMENT IN THE AFFECTED AREA(S) INOPERABLE.
5. IF TEMPERATURES ARE LESS THAN THE LIMIT AND TRENDING DOWNWARD, DATA MAY BE RECORDED AS LESS THAN THE LIMIT.

REFERENCE: FSAR 16.3.

Day Shift Complete: ____________/____________/____________

Night Shift Complete: ____________/____________/____________

Shift Supervisor Review: ____________/____________/____________

Initial Date Time

WINDOW A07

ORIGIN	SETPOINT	
1-FSL-12739	Not Applicable	BAT RM TRN A SPLY FAN LO AIR FLOW

1.0 PROBABLE CAUSE

1. Control Building Safety Feature (CBSF) Electrical Equipment Air Conditioning Unit Fan tripped.
2. Clogged Particulate Filter.
3. Dampers closed.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Stop** CBSF Electrical Equipment Air Conditioning Unit Fan A-1532-A7-001 using 1-HS-12733A on Panel QHVC.
2. **Dispatch** an operator to check for:
   a. Clogged Particulate Filter.
   b. Closed dampers.
3. IF equipment failure is indicated, **initiate** maintenance as required.
4. IF cooling unit CANNOT be restarted, doors to rooms B52, B55, and B76 must be opened to limit room temperature rise and prevent equipment failure.

89. 065AA2.05 001/1/1/IA - SHUTDOWN/C/A - 3.4/4.1/NEW/HL-18 NRC/SRO/KAJ

Initial conditions:

- Unit 2 is at 100% power.

- TPCCW cooling to the air compressors is reduced due to a small pipe break.

- The Shift Supervisor is implementing 18023-C, "Loss of Turbine Plant Cooling and Closed Cooling Water Systems," Section B, for loss of TPCCW.

- Instrument air pressure is degrading.

Given the conditions above, which ONE of the following completes the following statement?

Per 18023-C, the Shift Supervisor is required to direct the use of 13710-2, "Service Air System," to establish __(1)__ water for emergency cooling of the air compressors,

and

if air compressor cooling is NOT adequate, the Shift Supervisor is required to __(2)__.

A.✓ (1) utility

(2) initiate 18028-C, "Loss of Instrument Air"

B. (1) utility

(2) trip the reactor and initiate 19000-C, "Reactor Trip or Safety Injection"

C. (1) demin

(2) initiate 18028-C, "Loss of Instrument Air"

D. (1) demin

(2) trip the reactor and initiate 19000-C, "Reactor Trip or Safety Injection"

065AA2.05 Loss of Instrument Air

Ability to determine and interpret the following as they apply to the Loss of Instrument Air:
(CFR: 43.5 / 45.13)

When to commence plant shutdown if instrument air pressure is decreasing.

K/A MATCH ANALYSIS:

The candidate is presented with a scenario where air compressor cooling has been degraded, the candidate has to determine the backup cooling water to the air compressors (utility water) and what actions 18023-C for Loss of TPCW will direct the SS to perform.

ANSWER / DISTRACTOR ANALYSIS:

A. Correct. 1st half is correct, utility water serves as backup cooling to the air compressors. 18023-C will direct the SS to 18028-C for Loss of Air. The loss of air procedure will direct a unit shutdown if the air compressors cannot be restored.

B. Incorrect. 1st half is correct, utility water serves as backup cooling to the air compressors. 18023-C will not direct a reactor trip but will send the SS to 18028-C, Loss of Instrument Air.

C. Incorrect. 1st half of question is incorrect but plausible. Demin water services many loads in the turbine building and is the makeup to the TPCCW Surge Tank which acts as cooling water to the compressors. However, utility water not demin water is the procedurally directed method for backup cooling water. 18023-C will direct the SS to 18028-C for Loss of Air. The loss of air procedure will direct a unit shutdown if the air compressors cannot be restored.

D. Incorrect. 1st half of question is incorrect but plausible. Demin water services many loads in the turbine building and is the makeup to the TPCCW Surge Tank which acts as cooling water to the compressors. However, utility water not demin water is the procedurally directed method for backup cooling water. However, utility water not demin water is the procedurally directed method for backup cooling water. 18023-C will direct the SS to 18028-C for Loss of Air. The loss of air procedure will direct a unit shutdown if the air compressors cannot be restored.

REFERENCES:

18023-C, "Loss of Turbine Plant Cooling and Closed Cooling Water Systems", Section B, for loss of TPCCW".
18028-C, "Loss of Instrument Air".
13710-2, "Service Air System"

VEGP learning objectives:

LO-LP-60319-04 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is being performed

LO-PP-02101-04 State the cooling water supply to the compressors for normal operation and outage operations.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

B. LOSS OF TURBINE PLANT CLOSED COOLING WATER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B8. Check components in TABLE B1 - TEMPERATURES LESS THAN MAXIMUM.	__B8. Shutdown affected equipment as soon as possible.
__B9. Check air compressor cooling - ADEQUATE.	B9. Perform one of the following: __ Establish an emergency source of cooling water using 13710, SERVICE AIR SYSTEM and start compressors as required. -OR- __ Initiate 18028-C, LOSS OF INSTRUMENT AIR.
__B10. Check TPCCW - RESTORED.	__B10. Return to Step B1.
__B11. Return to procedure and step in effect.	

END OF SUB-PROCEDURE TEXT

INITIALS

4.4.4 Rotary Air Compressor Operation With Temporary Cooling Water Supplied From The Utility Water System

NOTES

- To restart Rotary Air Compressor 2-2401-C4-501 after alignment to Utility Water is completed and compressor has been initially run, Step 4.4.4.1 b.(8) should be performed. ☐
- To restart Rotary Air Compressor 2-2401-C4-502 after alignment to Utility Water is completed and compressor has been initially run, Step 4.4.4.2 b.(8) should be performed. ☐

4.4.4.1 To place Rotary Air Compressor 2-2401-C4-501 on temporary cooling supplied from Utility Water System, perform the following:

a. **Obtain** SS permission to align Utility Water to Rotary Air Compressor 2-2401-C4-501. ________

b. To align Utility Water to Rotary Air Compressor 2-2401-C4-501, perform the following:

(1) **Shut down** Rotary Air Compressor 2-2401-C4-501 per Section 4.3.1. ________

NOTES

- Each rotary air compressor should be supplied from an independent Utility Water station through 1-inch interior diameter hoses and fittings to ensure adequate water flow for cooling. ☐
- Utility Water station specified is based on location and most efficient routing of hoses. An alternate station MAY be used. ☐

(2) **Establish** Utility Water supply flow path by connecting the following valves using approximately 75 feet of hose:

- UTILITY WTR TURB BLDG LEVEL A STA ISO 2-2419-U4-542 (Level A TB8) ________
- TPCCW AIR CMPSR 1 AFT CLR DRN 2-1404-X4-993 ________

INITIALS

4.4.5 Reciprocating Air Compressor Operation With Temporary Cooling Water Supplied From The Utility Water System

> NOTE
>
> After alignment with Utility Water has been completed and AC has been initially run, subsequent starting and stopping of AIR CMPSR 3 should be per Step 4.4.5.1.c. □

4.4.5.1 To place Reciprocating Air Compressor 2-2401-C4-503 on temporary cooling supplied from Utility Water System, perform the following:

a. **Obtain** SS permission to align Utility Water to Reciprocating Air Compressor 2-2401-C4-503. ________

b. To align Utility Water to Reciprocating Air Compressor 2-2401-C4-503, perform the following:

(1) **Shut down** Reciprocating Air Compressor 2-2401-C4-503 per Section 4.3.2. ________

(2) **Close** the following TPCCW supply and return valves to the aftercooler and **install** TM Tags:

(a) TPCCW AFT CLR 3 INL FCV-17346 OUT ISO 2-1404-U4-694. ________

(b) TPCCW AFT CLR 3 INL FCV-17346 BYP ISO 2-1404-U4-590. ________

(c) TPCCW AFT CLR 3 OUTLET ISOLATION 2-1404-U4-678. ________

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__*19. **Check Instrument Air header pressure - REMAINS GREATER THAN 70 PSIG.**	*19. Perform the following: __a. Trip the reactor. __b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. __c. Initiate ATTACHMENT A, LOSS OF INSTRUMENT AIR IN MODE 3.
__20. Check header pressure – STABLE <u>OR</u> RISING.	20. <u>IF</u> leakage source can <u>NOT</u> be isolated, <u>THEN</u> restore/isolate <u>UNAFFECTED</u> unit Instrument Air as follows: a. Perform one of the following: __ <u>IF</u> Unit 1 is selected for the swing compressor, <u>THEN</u> close 2-2401-U4-510. -OR- __ <u>IF</u> Unit 2 is selected for the swing compressor, <u>THEN</u> close 1-2401-U4-510. __b. Verify swing compressor is running (TB-A-TC11).
__21. Check Instrument Air header pressure on PI-9361 - GREATER THAN 100 PSIG.	__21. Go to Step 24.

90. 076AG2.2.22 001/1/2/HI RC ACTIVITY/C/A - 4.0/4.7/NEW/HL-18 NRC/SRO/TNT

Given the following conditions:

- Unit 2 is at 80% power and holding for NIS Calorimetric.
- Chemistry reports that RCS Dose Equivalent I-131 is reading 125 microCi/gram.

Which ONE of the following completes the following statement?

Per Tech Spec 3.4.16, "RCS Specific Activity," Dose Equivalent I-131 falls in the __(1)__ operation region,

and

the Bases for the Tech Spec 3.4.16 limit ensures that __(2)__.

REFERENCE PROVIDED

A.✓ (1) unacceptable

(2) offsite radioactivity dose consequences are minimized in the event of a SGTR accident

B. (1) unacceptable

(2) the activity will not interfere with the detection of leakage to the containment

C. (1) acceptable

(2) offsite radioactivity dose consequences are minimized in the event of a SGTR accident

D. (1) acceptable

(2) the activity will not interfere with the detection of leakage to the containment

076AG2.2.22 High Reactor Coolant Activity

Knowledge of limiting conditions for operations and safety limits. (CFR 41.5. / 43.3 / 45.2)

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario given an RCS Dose Equivalent I-131 reading. Using a Tech Spec figure he has to determine if operation falls in the acceptable or unacceptable region. In addition, the candidate has to determine the bases for the RCS Dose Equivalent I-131 limit.

DISTRACTOR ANALYSIS:

A. Correct. Per Tech Spec figure 3.4.16-1, operation is in the unacceptable region and the bases for this is correct.

B. Incorrect. Per Tech Spec figure 3.4.16-1, operation is in the unacceptable region. This bases is incorrect but is plausible, this is the bases for 3.4.13 (Identified Leakage)

C. Incorrect. Per Tech Spec figure 3.4.16-1, operation is in the unacceptable region. The bases for this is correct.

D. Incorrect. Per Tech Spec figure 3.4.16-1, operation is in the unacceptable region. This bases is incorrect but is plausible, this is the bases for 3.4.13 (Identified Leakage)

REFERENCES:

Tech Spec and Bases 3.4.16, RCS Specific Activity
Tech Spec Bases 3.4.13, RCS Operational Leakage
Student is handed out Figure 3.4.16-1 for first part of Q.

VEGP learning objectives:

LO-LP-39208-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.4 are exceeded.

b. The required actions for all section 3.4 LCOs.

LO-LP-39208-04 Describe the bases for any given Tech Spec in section 3.4.

LO-LP-39208-06 State the reason for limiting the RCS specific activity.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

------------------------------NOTE------------------------------
LCO 3.0.4c is applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Perform SR 3.4.16.2.	4 hours
	<u>AND</u>	
	B.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-----------------------------NOTE---------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

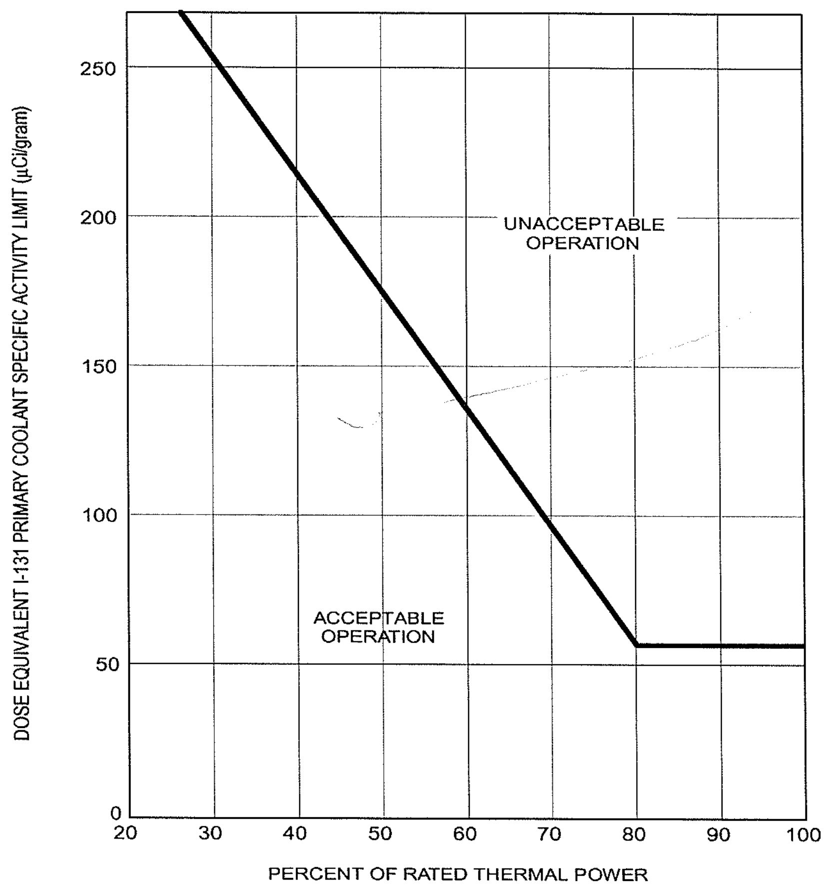


FIGURE 3.4.16-1
REACTOR COOLANT DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACITVITY >1 mCi/gram DOSE EQUIVALENT I-131

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the exclusion area boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO limits specific activity for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the exclusion area boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes that the reactor has been operating at the maximum allowable Technical Specification limit for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with concurrent large iodine spike that increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are well within the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the exclusion area boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the exclusion area boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to exclusion area boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can allow varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

(continued)

91. 086G2.2.37 001/2/2/FIRE PROT - EQUIP/C/A - 3.6/4.6/NEW/HL-18 NRC/SRO/TNT

Given the following:

- Unit 1 is operating at 100% power.

- A fire occurs in the Control Building level B.

- 17103A-C, Annunciator Response for the Fire Alarm Computer, Table 3 Operator Actions for a Confirmed Fire in a Safety Related Area directs closing PRZR PORV Block Valve, 1HV-8000A.

Which one of the following completes the following statement?

Per 92005-C, Fire Response Procedure, the affected HVAC system in the area of the fire ___(1)___

and

in accordance with Tech Spec 3.4.11 (Pressurizer PORVs) Bases ONLY, after the PRZR PORV Block valve is closed, the PRZR PORV is ___(2)___ .

A.✓ (1) will be secured

(2) OPERABLE

B. (1) will be secured

(2) inoperable

C. (1) placed in the Smoke Purge Mode

(2) OPERABLE

D. (1) placed in the Smoke Purge Mode

(2) inoperable

086G2.2.37 Fire Protection System (FPS)

Ability to determine operability and/or availability of safety related equipment:
(CFR 41.7 / 43.5 / 45.12):

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where a fire in the Control

determine per Tech Spec Bases of 3.4.11 (Pressurizer PORVS) if the PORV is OPERABLE. The candidate also has to determine what actions are required for the running HVAC systems in the affected area.

This question is SRO per Figure 1 of the NRC Clarification Guidance for SRO-only questions (Tech Specs) due to knowledge of TS bases that is required to analyze TS required actions and terminology.

DISTRACTOR ANALYSIS:

A. Correct. 1st half is correct, the HVAC will be secured. The 2nd half is correct, with the block valve closed, the PORV is still capable of being manually operated and the operator can open the block valve if necessary.

B. Incorrect. 1st half is correct, the HVAC will be secured.The 2nd half is incorrect, with the block valve closed, the PORV is still capable of being manually operated and the operator can open the block valve if necessary.

C. Incorrect. 1st half is incorrect. It is plausible the candidate may think it is necessary to purge smoke during a fire, however, this will fan the flames. Smoke purge will only be placed in service once the fire is extinguished.

D. Incorrect. 1st half is incorrect. It is plausible the candidate may think it is necessary to purge smoke during a fire, however, this will fan the flames.The 2nd half is correct, with the block valve closed, the PORV is still capable of being manually operated and the operator can open the block valve if necessary.

REFERENCES:

17103A-C, Annunciator Response for Fire Alarm Computer, Tables 1 and 3
92005-C, Fire Response Procedure
Tech Spec 3.4.11 and Bases, Pressurizer PORVs

VEGP learning objectives:

LO-LP-63500-25 (SRO only) State the standard and expectations for operability.

LO-LP-39208-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.4 are exceeded.

b. The required actions for all section 3.4 LCOs.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

BASES

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR, or loss of heat sink, and to achieve safety grade cold shutdown. The PORVs are considered OPERABLE in either the manual or automatic mode. The PORVs (PV-455A and PV-456A) are powered from 125 V MCCs 1/2AD1M and 1/2BD1M, respectively. If either or both of these MCCs become inoperable, the affected PORV(s) are to be considered inoperable.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific criteria, exists when conditions dictate closure of the block valve to limit leakage.

An OPERABLE block valve may be either open and energized, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

The PORVs are required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event, an inadvertent safety injection, and to achieve safety grade cold shutdown. In addition, the block valves are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV

(continued)

TABLE 1 - UNIT 1 AND COMMON

DETECTION AND SUPPRESSION SYSTEMS REFERENCE INFO FOR FIRE ALARM RESPONSE

FIRE ZONE	FIRE ZONE DESCRIPTION	LZIP NO	LZIP LOC BLDG-LVL-RM	LDC NO	LDC LOC	LSIP NO	LSIP LOC BLDG-LVL-RM	SUPPR SYS NO	SUPPR SYS TYPE	PREPLAN NO.	AVAIL S/D TRN	CR OPER ACTIONS (SEE TABLE 3) Group 1	Group 2
78 A	CNTL BLDG - LEVEL B R-B52	1S25	CB-B-70							92778A-1	B	1a	15
78 B	CNTL BLDG - LEVEL B R-B54	1S25	CB-B-70			1F28	CB-B-42	099	MAN DELUGE	92778B-1	B	1a	15
79 A	CNTL BLDG - LEVEL B R-B47	1S25	CB-B-70							92779A-1	A	1b	15
79 B	CNTL BLDG - LEVEL B R-B49	1S25	CB-B-70			1F28	CB-B-42	107	MAN DELUGE	92779B-1	A	NONE	NONE
80	CNTL BLDG - LEVEL B B47M, B49M, B50	1S23	CB-B-70			1F27	CB-B-75	064	PREACTION	92780-1	B	1a, 1c	4, 15, 27
81 B	CNTL BLDG - LEVEL B, A, 1, 2, R-B24, A67, 168, 252	1S23	CB-B-70							92781B-1	B	1e, 2b, 6a	4, 5a, 5d, 7, 11a, 11b, 12a, 15, 16
82	CNTL BLDG - LEVEL B R-B67	1S22	CB-B-70							92782-1	A or B	NONE	NONE
83	CNTL BLDG - LEVEL B R-B45	1S23	CB-B-70			1F29	CB-A-34	059	PREACTION	92783-1	A or B	2a	4, 12b
84	CNTL BLDG - LEVEL A R-A70	1S29	CB-A-50			1F30	CB-A-58	069	PREACTION	92784-1	A or B	NONE	NONE
85	CNTL BLDG - LEVEL A R-A58	1S29	CB-A-50			1F31	CB-A-44	068	PREACTION	92785-1	B	1a, 1c, 2a, 2b, 6a	35, 5a, 5d, 7, 13a, 14, 17a
86	CNTL BLDG - LEVEL A R-A59	1S28	CB-A-58			1F33	CB-A-58	071	PREACTION	92786-1	B	1a, 1c, 2a, 2b, 6a	35, 5a, 5d, 7, 13a, 14, 17a
87	CNTL BLDG - LEVEL A R-A54	1S27	CB-A-58							92787-1	A		5c, 15
88	CNTL BLDG - LEVEL A R-A53	1S27	CB-A-58			1F32	CB-A-51	070	PREACTION	92788-1	A	1a, 1b, 1d, 2a, 6b	5b, 5c,, 8, 9, 11a, 11b, 12b, 13b, 14, 15, 16, 17b
89	CNTL BLDG - LEVEL A R-A65	1S28	CB-A-58			1F33	CB-A-58	071	PREACTION	92789-1	A	NONE	NONE
90	CNTL BLDG - LEVEL A R-A64	1S28	CB-A-58							92790-1	A or B	NONE	NONE

INITIALS

CONTINUOUS USE

TABLE 3

OPERATOR ACTIONS FOR A CONFIRMED FIRE IN A SAFETY RELATED AREA

NOTES

- On Table 1 the column marked AVAIL S/D TRN indicates the train of safe shutdown equipment that is free of fire damage, given that the specified steps below are performed. ☐
- This Table is written using Unit 1 component designations with Unit 2 component designations in parenthesis. ☐
- A fully involved room fire can produce ceiling temperatures that may exceed 700°F. Cable damage is expected to occur when exposed to direct flame impingement, or when exposed to heat at approximately 625°F. Operator actions in Table 3 should be based on an assessment of the fire damage or potential damage to electrical cables in the fire zone. ☐

CRITERIA FOR IMPLEMENTING TABLE 3 ACTIONS

Perform those Operator Action identified from Table 1 to prevent spurious equipment actuations or inactuations, ONLY IF full room fire involvement is present, OR IF the fire is damaging cables OR has the potential to grow into full room involvement.

1a. Pressurizer PORV PV-0455A may open.

(1) **On QMCB, place Pressurizer PORV Block Valve HS-8000A in CLOSE.** ________

1b. Pressurizer PORV PV-0456A may open.

(1) On QMCB, **place** Pressurizer PORV Block Valve HS-8000B in CLOSE. ________

1c. Pressurizer PORV PV-0455A may open and it may not be possible to close PORV Block Valve HV-8000A.

(1) **Dispatch** operator to locally OPEN 125V DC MCC breaker to close PV-0455A. ________

Unit-1 Breaker	Location	**Unit-2** Breaker	Location
1AD1M-04	CB-B52	2AD1M-04	CB-B29

3.8.2 Ensure that the channel for the Base Unit radio in the Control Room that is used for contact with the Fire Team Captain has been changed to "Fire Team Channel". ☐

3.8.3 IF Control Room evacuation is required, selection of command post SHOULD take into account the location of the communication equipment. Communication between the Fire Team Captain AND the Remote Shutdown panel SHOULD be established using one or more of the following options:

- Telephone
- Radio
- Gaitronics
- Sound powered headphones

3.8.4 If the fire is in an area that could pose a threat to radiological safety, notify Health Physics (HP) personnel to respond to the fire area. ☐

a. During a fire-fighting situation in a Radiation Control Area (RCA), HP should respond and perform a radiological assessment of the area.

3.8.5 Secure HVAC in the area of the fire. Reference the pre-plan for the fire zone for the procedure for the affected HVAC system, if necessary. ☐

3.8.6 If the fire situation warrants, the Plant Emergency Plan shall be initiated per NMP-EP-110, "Emergency Classification and Implementing Instructions". Information will be verbally transmitted to the Emergency Response Facilities as necessary. ☐

3.8.7 If the Plant Emergency Plan is not initiated, the SM/SS shall make the necessary notification and reports. The SM/SS shall direct other plant personnel as needed for control of the emergency and recovery operation. ☐

3.8.8 As soon as practical after the verification that a fire exists, notify the Fire Protection Program Engineer, Fire Protection System Engineers and/or the Fire Training Instructor. ☐

William H. (Bill) Lewis	3199/162	706-868-1868
Joey Shannon	3177/218	205-307-9392
James Singleton	4145/276	706-651-9635

92. G2.1.05 001/3//COND OPS/MEM - 2.9/3.9/NEW/HL-18 NRC/SRO/TNT

Which ONE of the following completes the following statement?

Per NMP-AD-016, "Fatigue Management Program," if someone is performing duties of a Fire Brigade Leader, this is considered __(1)__,

and

if deviations occur from the 10 CFR 26 work hour limits, individuals are considered "reset" from the deviation if they have had at least __(2)__ off since last at work.

A.✓ (1) Covered Work

(2) 10 hours

B. (1) Covered Work

(2) 8 hours

C. (1) Incidental Duties

(2) 10 hours

D. (1) Incidental Duties

(2) 8 hours

G2.1.05 Conduct of Operations

**Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
(CFR 41.10 / 43.5 / 45.12):**

K/A MATCH ANALYSIS:

The candidate is straight forward asked whether per NMP-AD-016 Fatigue Management Program the Fire Brigade Leader position is considered covered work or incidental duties. In addition, the candidate has to determine how many hours is the minimum required to "reset" from the 10CFR26 work limits.

DISTRACTOR ANALYSIS:

A. Correct. The first half is correct, Fire Brigade Leader is Covered Work. Per NMP-AD-016, 10 hours is the required time to "reset".

B. Incorrect. The first half is correct, Fire Brigade Leader is Covered Work. 8 hours was picked as a plausible distractor due to this was the old Tech Spec and Conduct of Operations limits for breaks between shift excluding turnover time and is a well known turnover time between some rotating shifts prior to part 26.

C. Incorrect. The first half is incorrect, Fire Brigade Leader is Covered Work. Per NMP-AD-016, 10 hours is the required time to "reset".

D. Incorrect. The first half is incorrect, Fire Brigade Leader is Covered Work. 8 hours was picked as a plausible distractor due to this was the old Tech Spec and Conduct of Operations limits for breaks between shift excluding turnover time and is a well known turnover time between some rotating shifts prior to part 26.

REFERENCES:

NMP-AD-016, Fatigue Management Program

VEGP learning objectives:

LO-LP-63503-05 State the requirements of shift manning for the following conditions: (SRO only)

e. time allowance for reduction of minimum shift crew and purpose of time allowance

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

4.14 **Eight (8)-Hour Shift Schedule** - A schedule that averages not more than nine (9) hours per workday over the entire shift cycle.

4.15 **eSOMS PQ&S®** - The Shift Operations Management System Personnel Qualification and Scheduling program, which is used to develop, schedule and track work hours and shift cycles.

4.16 **Fatigue** - The degradation in an individual's cognitive and motor functions as a result of inadequate rest.

4.17 **Incidental Duties** - Work activities performed offsite, required by the plant including but not limited to, technical assistance provided by phone calls or other work required by supervision. Incidental Duties are counted as work hours if the required activity exceeds a nominal, cumulative 30 minutes in a single break period. If the time does not exceed a nominal, cumulative 30 minutes, Incidental Duties are not considered work hours. The key word in this definition is 'required'. Self study for training or reviewing email on personal time are examples of activities that are not required and are therefore not considered to be Incidental Duties regardless of duration. A mandatory conference call such as participation in troubleshooting is an example of duties required by the plant and is counted as work hours if the cumulative time exceeds 30 minutes.

4.18 **Maintenance** - For the purposes of 10 CFR 26.4(a)(4), "maintenance" is defined as modification, surveillance, post maintenance testing and corrective and preventive maintenance that a risk informed evaluation process has shown to be significant to public health and safety. Refer to NMP-AD-016-002, "Scoping of Work Hour Limits" for detailed instruction on determining what constitutes maintenance.

4.19 **Nap or Restorative Sleep** - A brief opportunity with accommodations for restorative, uninterrupted sleep of at least one half hour in a designated area. This opportunity must be in a comfortable area conducive to sleep.

4.20 **Non-Covered Work** – any activity that is not considered covered work as described in section 4.9. Refer to NMP-AD-016-002, "Scoping of Work Hour Limits" for further guidance of covered and non-covered work.

4.21 **Off-site** - Any area not considered on-site.

4.22 **On-site** - Within the owner-controlled area.

4.23 **Outage** - The duration during which the main generator is disconnected from the electrical grid (breaker open to breaker close), not to exceed 60 days. A non-outage day for a shift is a day when the unit is not in an outage when the shift starts. An outage day for a shift is a day when the unit is in an outage when the shift starts.

4.24 **Protected Area** – As defined in 10 CFR 26, the area encompassed by physical barriers to which access is controlled.

4.6 **Condition Adverse to Safety-** Unforeseen conditions which, in the informed opinion of the Operations Shift Manager or Security Shift Supervisor, could jeopardize the safety of the public, station, personnel, or environment. Waivers are considered appropriate if required to prevent or mitigate conditions adverse to safety.

4.7 **Contractor/Vendor** - Any company or any individual not employed by SNC who is providing work or services to SNC either by contract, purchase order, oral agreement, or other arrangement.

NOTE:	The term 'Covered' as used in this procedure is different than the term 'Covered' as it applies to membership in a labor union.

4.8 **Covered Individual** - An individual subject to work hour controls defined in 10 CFR 26, Subpart I. Any individual who is granted unescorted access to a nuclear power plant protected areas and performs or directs covered work is a covered individual. Refer to NMP-AD-016-002, "Scoping of Work Hour Limits" for detailed guidance on determining personnel covered by work hour rules.

4.9 **Covered Work** - Any of the following activities:

4.9.1 Operating or on-site directing of the operation of structures, systems and components (SSCs) that a risk-informed evaluation process has shown to be significant to public health and safety.

4.9.2 Performing maintenance on or on-site directing of the maintenance of SSCs that a risk-informed evaluation process has shown to be significant to public health and safety.

4.9.3 Performing health physics or chemistry duties required as a member of the on-site emergency response organization's minimum shift complement.

4.9.4 Performing duties of a fire brigade leader.

4.9.5 Performing security duties as an armed security force officer, alarm station operator, response team leader, or watchperson, hereinafter referred to as security personnel.

4.10 **Cumulative fatigue** - The increase in fatigue over consecutive sleep-wake periods resulting from inadequate rest.

4.11 **Day Off** - A calendar day during which the individual does not start a work shift. Required duties that take > nominal 30 minutes cumulative time are work hours and preclude the day being counted as a day off.

4.12 **Deviation** – a departure from the requirements of 10 CFR Part 26, Subpart I.

4.13 **Directing** - The exercise of control over a work activity by an individual who is directly involved in the execution of the work activity and makes technical decisions for that activity without subsequent technical review or is ultimately responsible for the correct performance of that work activity. See NMP-AD-016-002, "Scoping of Work Hour Limits," for guidance on determination of directing activities.

6.1.6 On-the-Job Training (OJT), Task Performance Evaluation (TPE), Job Performance Measures (JPM), or any other type of training activity that is performed in the plant and manipulates Risk Significant SSCs must be evaluated in accordance with NMP-AD-016, Fatigue Management Program, and Part 26 Subpart I, Work Hour Rule (WHR), to determine if these activities result in the conduct of "Covered Work" as defined in Section 6.1. Training activities that manipulate Risk Significant SSCs must be considered synonymous as actual work for the purposes of managing WHR limits. If the activity is deemed "Covered Work", the trainee is a "Covered Worker" and subject to the work hour controls. Voluntary self-study, non-mandated after-hours study, off-site self-study or any other study time not specifically required as part of a documented remediation or performance improvement plan do not fall under WHR limitations and can be excluded from WHR calculations.

6.2 Work Scheduling

6.2.1 Work hours for covered individuals shall be scheduled with the objective of preventing impairment from fatigue due to duration, frequency, or sequencing of successive shifts.

6.2.2 Consider the following factors when establishing schedules:

- Work hour limits defined in 10 CFR 26, Subpart I
- Consistent start/stop times for work periods
- Impact of shift rotation
- Training requirements
- Vacation scheduling

6.2.3 Transitions into and out of covered groups, between covered groups and starting and ending outages have special rules that apply. Consult NMP-AD-016-003, "Scheduling and Calculating Work Hours" for detailed instructions for transitions.

6.2.4 Deviations from 10 CFR 26 work hour limits may occur as the result of administrative errors or unforeseen circumstances. A Condition Report shall be initiated, in accordance with NMP-GM-002, "Corrective Action Program," for each individual when this occurs. An individual is considered "reset" from deviation, whether under a waiver or otherwise, when they are less than the minimum hour requirements16/24, 26/48, 72/168, have had at least 10 hours off since last at work, and actions are in place to ensure they will meet the minimum days off requirement for their current shift cycle or outage window.

6.2.5 The software application, Shift Operations Management System (eSOMS) Personnel Qualification and Scheduling (PQ&S) shall be used to manage the shift schedules, work hours, tracking, and reports to ensure compliance with 10 CFR 26 Subpart I work hour rules. Refer to NMP-AD-016-GL01 for eSOMS PQ&S® for related guidance on work hour scheduling and tracking.

6.3 Calculating Work Hours

6.3.1 Refer to NMP-AD-016-003, "Scheduling and Calculating Work Hours," for further guidance.

93. G2.2.15 001/3//EQUIP CONTROL/C/A - 3.9/4.3/BANK-LOIT/HL-18 NRC/SRO/TNT

During a system tagout, 1LV-112D, RWST to CCP A & B Suction, was manually closed (locally at the valve) and its breaker was turned OFF. (Both the handwheel and the breaker were included on the tagout).

The Tagout was subsequently cleared with the following conditions existing:

- 1LV-112D is closed.

- No work was performed on the valve.

- The breaker for the valve has been closed and the GREEN valve position light is illuminated on the Main Control Board.

Which one of the following identifies the status of 1LV-112D and the MINIMUM post maintenance testing requirements in accordance with 10000-C, "Operations Administrative Controls"?

A. 1LV-112D is OPERABLE

Local observation of the valve stroking is required for return to service.

B. 1LV-112D is OPERABLE

Remote observation of the valve stroking is required for return to service.

C. 1LV-112D is NOT OPERABLE

The valve must be stroke timed in the open direction to demonstrate operability prior to return to service.

D✓ 1LV-112D is NOT OPERABLE

The valve must be manually unseated and stroked using the motor operator before returning to remote control and return to service.

G2.2.15 Equipment Control

Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.:
(CFR 41.10 / 43.3 / 45.13):

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a Safety Related MOV has been manually closed during a tagout but no work was performed on the valve. The candidate has to determine if the valve is OPERABLE and the proper action(s) to return the valve to service.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Per 10000-C, section 3.3.6.c the valve must be manually unseated and then stroked using the motor operator prior to return to service and declaring operable. Choice A is plausible the candidate can think the valve is operable since no work was performed on the valve and no additional action will be required for return to service.

B. Incorrect. Per 10000-C, section 3.3.6.c the valve must be manually unseated and then stroked using the motor operator prior to return to service and declaring operable. It is plausible an observation of smooth stroking is all that is required as our procedures in many instances require this local observation of valve stroking.

C. Incorrect. Per 10000-C, section 3.3.6.c the valve must be manually unseated and then stroked using the motor operator prior to return to service and declaring operable. Many of our valves required periodic stroke timing for operability, it is plausible the candidate may think this is the requirement to declare operable.

D. Correct. Per 10000-C, section 3.3.6.c the valve must be manually unseated and then stroked using the motor operator prior to return to service and declaring operable.

REFERENCES:

10000-C, Operations Administrative Control, section 3.3.6 for Manual Operation of Motor Operated Valves (MOVs)

VEGP learning objectives:

LO-LP-63500-25 (SRO only) State the standard and expectations for operability.

LO-LP-63500-31 State the standard and expectations for valve operation.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing $\leq$ 1 hour TS/TRM Action?

Yes RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)

Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO only

b. Effects of MOV Manual Operation on Plant Condition:

(1) Putting a motor operated globe valve in manual may actually compromise a clearance point. Normally the valve operator gear train essentially locks the valve in position. If the globe valve has pressure under its seat and is de-clutched, the pressure force may open the valve since the gear set no longer holds the drive sleeve in position.

(2) If manual operation is required, for reasons other than administrative controlled activities such as procedure performance or Tagouts, the associated hand switch shall be caution tagged to indicate that the valve has been manually operated.

(3) Safety related MOVs, which receive an actuation signal or are required to be repositioned to fulfill a safety-related function shall be considered inoperable.

c. Restoration Following Manual Operation:

An MOV, (including a Safety Related MOV) which has been manually actuated using the hand wheel operator, should normally be manually unseated and then stroked using the motor operator prior to returning it to remote service and declaring it operable. However, if the MOV meets the following exceptions, the valve can be returned to Remote service and declared operable without performing this stroking requirement.

A waiver to an electrical operability check may be made provided the MOV is NOT:

(1) A safety related MOV which receives an actuation signal or is required to be repositioned to fulfill a safety related function, or

(2) An MOV having a Limitorque type SMB00 actuator. See Attachment 1 for list of valves.

d. The SM authorizes any MOV stroking waiver by an entry into the Unit Control Log.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with the power lockout switches in the lockout position. Valve Number / Valve Function / Valve Position HV-8835 — SI Pump Cold Leg Inj. — OPEN HV-8840 — RHR Pump Hot Leg Inj. — CLOSED HV-8813 — SI Pump Mini Flow Isol. — OPEN HV-8806 — SI Pump Suction from RWST — OPEN HV-8802A, B — SI Pump Hot Leg Inj. — CLOSED HV-8809A, B — RHR Pump Cold Leg Inj. — OPEN	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify ECCS piping is full of water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable trains 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 MODE 3 within 6 hours and MODE 4 within 12 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 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in the correct position by placing the power lockout switches in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

mispositioned are in the correct position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies that the measured performance is within an acceptable tolerance of the original pump baseline performance. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

In addition to the acceptance criteria of the Inservice Testing Program, performance of this SR also verifies that pump performance is greater than or equal to the performance assumed in the safety analysis.

(continued)

94. G2.2.20 001/3//EQUIP CONTROL/C/A - 2.6/3.8/NEW/HL-18 NRC/SRO/TNT

Given the following:

- A local Heat Tracing Panel has some indicating lights that are not lit.
- This is a known problem and light bulbs have been replaced several times in the past.

It has been decided that a Troubleshooting Plan will be established per NMP-AD-002, "Problem Solving and Troubleshooting Guidelines."

As part of the plan, Electrical Maintenance will lift leads to measure voltage across some relay contacts in the Heat Tracing Panel.

Which ONE of the following completes the following statement?

This type of Troubleshooting Monitoring is called __(1)__ and the __(2)__ is responsible for maintaining SYSTEM status of the panel during the Troubleshooting activities.

A. (1) Intrusive

(2) Maintenance Manager

B✓ (1) Intrusive

(2) Operations Manager

C. (1) Non-Intrusive

(2) Maintenance Manager

D. (1) Non-Intrusive

(2) Operations Manager

G2.2.20 Equipment Control

Knowledge of the process for managing troubleshooting activities: (CFR 41.10 / 43.5 / 45.13):

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where Trouble Shooting is to be performed in the Heat Tracing Panel and involves lifting leads to measure voltage across some relay contacts. The candidate must determine if this is intrusive or non-intrusive trouble shooting and also has to determine the person responsible for maintaining the system status of the panel during Trouble Shooting activities.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. 1st half is correct, this is intrusive troubleshooting. The Operations Manager is the person responsible for tracking systems status.

B. Correct. This is intrusive troubleshooting. The Operations Manager is the person responsible for tracking systems status.

C. Incorrect. The first half is wrong, this is intrusive troubleshooting. The Operations Manager is the person responsible for tracking systems status.

D. Incorrect. The first half is wrong, this is intrusive troubleshooting. The Operations Manager is the person responsible for tracking systems status.

REFERENCES:

NMP-AD-002, "Problem Solving and Troubleshooting Guidelines", section 3.0 and 4.2

VEGP learning objectives:

LO-LP-63354-03 Describe the Shift Manager's responsibility concerning maintenance activities

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

1.0 Purpose

The purpose of this procedure is to provide a process for performance of troubleshooting when required for plant problem resolution. These problems may include equipment failures, abnormal operating conditions, negative performance trends or recurring events.

2.0 Applicability

2.1 This procedure is applicable to troubleshooting activities at any of the SNC sites.

2.2 Entry into the formal troubleshooting process is not intended for simple problems where the cause appears straightforward or known. In these cases investigation will be controlled by the Work Order process.

2.3 Formal troubleshooting activities shall be performed in accordance with this procedure unless waived by plant management or management within the affected department. If waived, the justification shall be documented in the appropriate location (Condition Report, Work Order, etc).

2.4 This procedure does not apply to special tests.

2.5 All troubleshooting shall use high impedance M&TE and/or isolation transformers on the signal and AC power source unless low impedance is specifically called for in equipment procedure.

3.0 Definitions

3.1 **Troubleshooting** – A systematic approach to data collection, failure analysis, or a measurement plan that results in high confidence that the complete cause of system/equipment degradation has been determined. There may be potential personnel safety risk.

3.2 **High Risk Troubleshooting** - Potential impacts are assessed as high risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.3 **Medium Risk Troubleshooting** - Potential impacts are assessed as medium risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.4 **Low Risk Troubleshooting** - Potential impacts are assessed as low risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.5 **Non-Intrusive Monitoring** – The act of monitoring a component or system by not affecting normal operation of the component or system. Examples would be using "Voltage Test Jacks," monitoring voltages across relay contacts, power supplies, etc.

3.6 **Intrusive Monitoring** – The act of temporarily altering the system to allow monitoring a component or system. This applies to electrical or mechanical testing methods.

3.7 **Stop-Decision Points** – Administrative and Physical Hold Points within the Troubleshooting Plan to limit and control activities.

4.0 Responsibilities

4.1 **Troubleshooting Leader**- Individual assigned to develop the troubleshooting plan, coordinate work and team discussions, act as a single point of contact and/or obtain changes to the plan, as assigned by the responsible department manager. Perform Just In Time Risk Assessment as described in NMP-DP-001, Operational Risk Awareness.

4.2 **Operations Department**

The Operations Department, under the direction of the Operations Manager, is responsible for ensuring the troubleshooting activities are supported by:

- Approve the Troubleshooting plan where risk has been assessed as Medium or High
- Providing personnel in support of the Troubleshooting Team
- Maintain approved system status during troubleshooting activities (i.e. Out Of Service)

4.3 **Work Planning**

- The group developing the Troubleshooting plan will determine the level of risk associated with the Troubleshooting plan by using Procedure NMP-DP-001, Operational Risk Awareness.

- The plan should consider elimination of worst case, long lead time components early in the process, as potential causes.

- Troubleshooting plan steps that alter the configuration of the plant will be implemented and controlled by a planned work order or use of referenced instructions from an approved procedure. This requirement may be waived by the operations shift manager in which case configuration changes will be controlled using detailed instructions in the troubleshooting plan and will be approved by operations before implementation.

- Troubleshooting plan steps that do not affect plant configuration control, for example system walkdown, data collection and trending, field observation, and other similar fact finding steps may be implemented by the troubleshooting plan steps.

4.4 **Maintenance Department**

The Maintenance Department, under the direction of the Maintenance Manager, is responsible for:

- Maintenance Manager or his designee will approve High or medium risk Troubleshooting activities where personal safety or economic safety are assessed as Medium or High risk
- Providing personnel in support of the Troubleshooting Team
- Ensuring that Troubleshooting Plan is performed and documented in accordance with approved site procedures and safe work practices
- Determining the need for additional support for troubleshooting activities

95. G2.3.04 001/3//RAD CONTROL/MEM - 3.2/3.7/BANK - FARLEY 2012/HL-18 NRC/SRO/TNT

Which ONE of the following completes the statements below in accordance with 91301-C, "Emergency Exposure Guidelines"?

The emergency exposure limit for a life-saving activity during a declared emergency is ___(1)___ REM TEDE.

The minimum level of authority Emergency exposure limits can be authorized by is the ___(2)___.

	(1)	(2)
A.	10	Health Physics (HP) Supervisor
B.	10	Emergency Director (ED)
C.	25	Health Physics (HP) Supervisor
D.✓	25	Emergency Director (ED)

G2.3.04 **Radiation Control**

Knowledge of radiation exposure limits under normal or emergency conditions:
(CFR 41.12 / 43.4 / 45.10)

K/A MATCH ANALYSIS:

This question asks the candidate straight forward the emergency exposure limits for life saving during an emergency and who authorizes these exposure limits.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. 10 REM TEDE is plausible because this is the limit for Protecting Valuable Property, but for saving a life, 25 REM can be required.
(2) Incorrect, Plausible because the HP Supervisor can give approval to exceed plant administrative dose limits, but not 10CFR20 limits.

B. Incorrect. 25 REM TEDE is the limit for life-saving, ED is who authorizes.

C. Incorrect. 25 REM TEDE is the limit for life-saving, ED is who authorizes.

D. Correct. 25 REM TEDE is the limit for life-saving, ED is who authorizes.

REFERENCES:

91301-C, Emergency Exposure Guidelines, Table 1

VEGP learning objectives:

LO-LP-40101-35 State the emergency TEDE limits for the following (SRO only):

a. All activities
b. Protecting valuable property
c. Life saving actions or protection of large population.

This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

TABLE 1

EMERGENCY EXPOSURE GUIDELINES (1985305256) (1985305827)

NOTES

- Dose limits listed in this table apply to doses incurred over the duration of the emergency.
- Dose to workers performing emergency services may be treated as an once-in-a-lifetime exposure and should not be added to occupational exposure accumulated under non-emergency conditions.
- Workers performing services during emergencies shall limit dose to the lens of the eyes to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value.

Dose Limit (REM) Total Effective Dose Equivalent	Activity	Condition
5	All	
10	Protecting Valuable Property	Lower Dose not practicable
25	Lifesaving or protection of large population	Lower Dose not practicable
>25	Lifesaving or protection of large population	Only on a voluntary basis to persons fully aware of the risks involved

96. G2.3.07 001/3//RAD CONTROL/MEM - 3.5/3.6/NEW/HL-18 NRC/SRO/TNT

Given the following:

- A Fuel Handling Coordinator (FHC) is entering the Spent Fuel Pool area.
- The FHC is reviewing his RWP prior to beginning work and notices an ALARA briefing is required.
- The dose rate is 900 mrem/hr due to damaged fuel assemblies.
- The FHC will also exceed 2000 mrem Annual TEDE limits while in the area.

Which one of the following completes the following statement?

Based on the area dose rate, the FHC will be required to receive an ALARA briefing prior to ___(1)___ entry

and

Per NMP-HP-001, "Radiation Protection Standard Practices," the ___(2)___ is the MINIMUM authority level required to exceed the limit.

A. (1) each

(2) HP Manager

B. (1) each

(2) Plant General Manager

C✓ (1) ONLY the first

(2) HP Manager

D. (1) ONLY the first

(2) Plant General Manager

G2.3.07 Radiation Control

Ability to comply with radiation work permit requirements during normal or abnormal conditions:
(CFR 41.12 / 45.10)

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a scenario where a Fuel Handling Coordinator (FHC) is required to enter the SFP area with high dose rates in the area. The FHC is also on the verge of exceeding his Annual TEDE limits of 2000 mrem. The candidate has to determine the minimum level of authority that may approve exceeding the annual TEDE limits.

The question is SRO per NRC Clarification Guidelines for SRO-only Questions, Figure 2, knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

DISTRACTOR ANALYSIS:

A. Incorrect. With dose rate at 900 mrem, this will be a YELLOW RWP and requires a briefing only prior to the first entry. The minimum authority level to approve the TEDE extension is the HP Manager from the choices presented.

B. Incorrect. With dose rate at 900 mrem, this will be a YELLOW RWP and requires a briefing only prior to the first entry. The minimum authority level to approve the TEDE extension is the HP Manager from the choices presented.

C. Correct. With a YELLOW RWP, only the first entry requires a briefing. The minimum authority level to approve the TEDE extension is the HP Manager from the choices presented.

D. Incorrect. With dose rate at 900 mrem, this will be a YELLOW RWP and requires a briefing only prior to the first entry. The minimum authority level to approve the TEDE extension is the HP Manager from the choices presented.

REFERENCES:

NMP-HP-001, Radiation Protection Standard Practices
43007, Issuance Use and Control of Radioactive Work Permits

VEGP learning objectives:

LO-LP-63930-06 State the entry requirements applicable to each of the following:

b. Radiation Control Area (RCA)
c. Radiation Area
d. High Radiation Area
e. Locked High Radiation Area

Section D for Analysis and interpretation of radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

- Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

6.1.55 20.2202 Notification of Incidents
Paragraph (a)(2) and (b)(2), for reporting the release of radioactive material inside or outside of a restricted area is caveated as not applicable to, "locations where personnel are not normally stationed during routine operations, such as hot cells or process enclosures." For purposes of this section, consistent with the answer to question 56 of the first NRC 10CFR20 Q & A document, locations where personnel are not normally stationed will be interpreted as areas, rooms and enclosures which are not normally occupied nor periodically patrolled during normal plant operations and maintenance.

6.1.56 20.2203 Reports of Exposures, Radiation Levels and Concentrations of Radioactive Material Exceeding the Limits
No fleet practices identified.

6.1.57 20.2204 Reports of Planned Special Exposures
No fleet practices identified.

6.1.58 20.2206 Reports of Individual Monitoring
The intent of Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," will be met in complying with this paragraph with NRC Form 5.

6.1.59 20.2301 Applications for Exemptions
No fleet practices identified.

6.1.60 20.2302 Additional Requirements
No fleet practices identified.

6.1.61 20.2401 Violations
No fleet practices identified.

6.2 Other Consensus Positions

6.2.1 Whole Body Count Performance Frequency
Monitored workers will be given an entrance and exit whole body count (WBC) or whole body scan (WBS). The exit WBC from another SNC site can be used in lieu of an entrance WBC if the SNC site was the last site the worker entered an RCA and/or monitored. Upon request from a worker, WBCs will be provided to the worker on a voluntary and reasonable basis.

6.2.2 Dose Limits for Workers Who Provide Outage Support at a SNC Plant Other Than Their Home Plant.
Workers should be limited to 500 mrem per visit, unless express consent is given by the home plants management to exceed that limit.

6.2.3 Administrative Annual TEDE Dose Limits and the Approval Authority Necessary to Exceed Limits

6.2.3.1 2000 mrem in a year requires HP Support Supervisor, Plant Health Physicist, or HP Manager approval.

6.2.3.2 4000 mrem in a year requires AGM or Plant General Manager approval.

6.2.3.3 4500 mrem in a year requires Project Vice President approval.

6.2.4 Discrepant Dosimeter Investigation Criteria

An assessment of worker's dose should be initiated for discrepant dosimetry results when the following criteria are met: the primary or secondary dosimeter dose exceeds 100 mrem; and the secondary dosimeter reading differs by more than 25% from the primary dosimeter.

6.2.5 Dose Monitoring Threshold

All individuals entering an RCA will be monitored for radiation exposure. A single dosimeter suffices for Visitors or Radiation Workers whose annual dose from sources external to the body is not expected to exceed 100 mrem at a particular station. An Optically Stimulated Luminescent Dosimeter (OSLD) and a self-reading dosimeter, such as an Electronic Dosimeter (ED), will be provided to all other individuals.

6.2.6 Training Requirements for Visitors or Temporary Radiation Workers Who Enter RCAs

All SNC plants will administer training to all who must enter the RCAs in the following manner:

6.2.6.1 If the individual is expected to receive < 100 mrem in a year, the individual will be escorted by a GET qualified worker and will be provided instructions.

6.2.6.2 If the individual is expected to receive ≥100 and <500 mrem in a year, the individual will receive a visitor handout (containing all of the instruction elements required by 10CFR19.12) and will acknowledge receipt of the instructions or handout by signing a form. The individual will have a GET-trained radiological escort. If special circumstances dictate that entries into contaminated areas are required, the individual will receive dress-out training (if he has no history of such training at our plants). Similarly, if the individual requires entry to high radiation areas, special training or instructions may be required.

6.2.6.3 If the individual is expected to receive >=500 mrem in a year, the individual is required to complete GET which includes testing. If the individual has previous GET training at a nuclear facility within the past two years or as allowed by the Training Department, then exemption GET and the exemption GET test can be administered.

6.2.7 Air Flow for Fume Hoods Containing Radioactive Materials/Fluids

Sample station fume hoods containing radioactive materials/fluids will meet a minimum air flow acceptance criteria of 100 LFPM, and will be labeled to ensure that 100 LFPM is maintained or exceeded.

6.2.8 Respirator Training and Fit Test Frequencies

Sites will apply a program whereby classroom or Computer-based training with an examination will be conducted annually. For individuals required to utilize respirator

RISKED BASED RWP FORMAT

Color Code	***Radiological Significance***	***Types of RWPs***	***Type of Briefing Required***	***General Radiological Conditions***
Green	Low	**All *General* RWPs** And ***Specific* RWPs with low radiological risk**	**No ALARA briefing required.**	**Dose Rate:** <100 mrem/hr **Contamination Levels:** <200,000 dpm/100 cm^2 ***Airborne Levels: <0.3 DAC*** * Workers should always refer to the most recent survey information for the area(s) being worked in.
Yellow	Moderate	***Specific* RWPs that are tied to unique *Work Groups* -** ***Specific* RWPs covering work in areas with intermediate levels of radiological risk.**	• **Initial ALARA briefing required prior to *first* entry.** • **Additional ALARA briefing required when specified rad conditions are exceeded.** ***A pre-job ALARA briefing will be required if:*** • **Radiological conditions that are addressed in the Worker Instructions section may be exceeded, or** • **If the RWP default settings for the accumulated dose or dose rate alarms may be exceeded, or** • **Breach of a contaminated system**	**Dose Rate:** <1000 mrem/hr **Contamination Levels:** <500,000 dpm/100 cm^2 **Airborne Levels:** <0.3 DAC * Workers should always refer to the most recent survey information for the area(s) being worked in.
Red	High	***Specific* RWPs** covering work in areas with high levels of radiological risk.	***ALARA Briefing required prior to each entry.*** • **Radiological conditions on the RWP will be based on actual, projected or historical survey information.** • **Latest rad conditions and specific instructions will be covered in the pre-job ALARA briefing**	**Dose Rate:** >1000 mrem/hr **Contamination Levels:** >500,000 dpm/100 cm^2 **Airborne Levels:** >0.3 DAC * Workers should always refer to the most recent survey information for the area(s) being worked in.

TABLE 1

97. G2.4.29 001/3//EMERG PROC/MEM - 3.1/4.4/MOD-HL17 NRC/HL-18 NRC/SRO/TNT

Per ~~procedure~~ NMP-EP-110, "Emergency Classification Determination and Initial Action," which ONE of the following identifies a duty that **CAN** be delegated by the Emergency Director?

A. The decision to request federal assistance.

B✓ Coordinating and directing emergency operations.

C. The decision to recommend protective actions to offsite authorities.

D. Authorizing use of potassium iodide (KI) tablets during a declared emergency.

G2.4.29 Emergency Procedures / Plan

Knowledge of the emergency plan:
(CFR 41.10 / 43.5 / 45.11)

K/A MATCH ANALYSIS:

This question basically asks about Duties of the Emergency Director per NMP-EP-110, Emergency Classification Determination and Initial Action and which one CAN be delegated.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Per NMP-110, step 5.1.1, this duty is non-delegable.

B. Correct. Per NMP-110, step 5.1.2, this duty is delegable.

C. Incorrect. Per NMP-110, step 5.1.1, this duty is non-delegable.

D. Incorrect. Per NMP-110, step 5.1.1, this duty is non-delegable.

REFERENCES:

NMP-EP-110, Emergency Classification Determination and Initial Action, Section 5.1
HL-17 NRC Exam

VEGP learning objectives:

LO-LP-40101-08 State from memory ED duties that cannot be delegated (SRO only).
This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

4.3 **Site Area Emergency** - Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or Hostile Actions that result in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

4.4 **General Emergency** - Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or Hostile Action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

5.0 Responsibilities

5.1 Emergency Director (ED)

5.1.1 The ED has the following non-delegable responsibilities:

- The decision to declare, escalate, or terminate emergency classifications.
- The decision to notify offsite emergency response agencies.
- The decision to recommend protective actions to offsite authorities.
- The decision to request federal assistance.
- Authorization for plant personnel to exceed 10CFR20 radiation exposure limits.
- Authorization for use of potassium iodide (KI) tablets during a declared emergency.
- The decision to dismiss nonessential personnel from the site at an ALERT or higher emergency classification.

5.1.2 The ED has the following delegable responsibilities:

- Maintaining communications with offsite authorities regarding all aspects of emergency response.
- Providing overall direction for management of procurement of site-needed materials, equipment, and supplies, documentation, accountability, and security function.
- Directing the notification and activation of the emergency organization; including emergency response facility activation.
- Coordinating and directing emergency operations.
- If requested by offsite agencies, the ED shall dispatch SNC representatives to offsite government centers.
- Modifying Emergency Plan Implementing Procedures, Security Plan, Security Plan Implementing Procedures and adjusting Emergency Response Organization staffing.

- Coordinating NRC activities to reduce the duplication of effort and reduce the impact on the plant staff during the emergency situation.
- Directing the assignment of an individual as Decision Maker if Severe Accident Management Guidelines (SAMG) are implemented.

5.1.3 The Shift Manager (SM) is responsible for initial classification of events. The SM shall assume the responsibilities of the ED until relieved by another qualified ED.

5.1.4 If the SM is unavailable, then the unaffected unit's Shift Supervisor (SS) will assume the responsibilities of the ED until relieved.

5.1.5 IF the SM is unavailable and the event involves both units, the Unit 1 SS OR another qualified ED will assume the responsibilities of the ED until relieved.

5.1.6 Transfer of ED responsibilities is completed in accordance with Checklist 3.

5.1.7 After turnover of ED responsibilities, the SM then continues to be responsible for recognizing changes in plant conditions and advising the ED concerning classification of events.

5.1.8 Any one of the following qualified persons may assume the position of ED after receipt of turnover information from the off going ED.

- Plant Manager
- Site Support Manager
- Operations Manager
- Maintenance Manager
- Any qualified Emergency Director

5.1.9 The Technical Support Center (TSC) Manager and Emergency Operations Facility (EOF) Manager are responsible for providing recommendations on emergency classifications to the ED.

6.0 <u>Procedure</u>

6.1 Precautions / Limitations

6.1.1 This procedure establishes minimum requirements for emergency classification. The ED may use judgment as the final criterion for determining the classification of off-normal events that are not included in this procedure.

6.1.2 The value of any emergency actions, which may require movement of plant personnel, must be judged against the danger to personnel or nuclear safety.

6.1.3 Classification should not be delayed in anticipation of either events being terminated or the threat to safety ending.

6.1.4 Personnel and plant safety must be addressed as the highest priority, if necessary, prior to an emergency classification.

1. G2.4.37 001/3/N/A/ED DUTIES/F 3.0/4.1/M-HL-15 AUDIT/HL-17/SRO/EMT/GCW

Per Procedure 91102-C, "Duties Of The Emergency Director", which one of the following identifies a duty that CAN be delegated by the Emergency Director?

A.✓ Filling the position of Decision Maker, if Severe Accident Management Guidelines (SAMGs) are implemented.

B. Recommending protective actions to offsite authorities and content of notification messages, after the initial notifications have been made and the TSC is fully staffed.

C. Authorizing personnel radiation exposures in excess of 10CFR20 limits, if necessary.

D. Deciding to request assistance from federal support groups.

98. G2.4.40 001/3//EMERG PROC/MEM - 2.7/4.5/BANK-SEQ 2008/HL-18 NRC/SRO/TNT

Given the following:

- An event on Unit 1 requires an ALERT emergency declaration.

- The Shift Manager is incapacitated due to a medical problem.

Which ONE of the following identifies both the individual who will initially serve as the alternate for the Emergency Director position in accordance with 91101-C, "Emergency Response Organization,"

and

the requirement for conducting Assembly and Accountability in accordance with 91401-C, "Assembly and Accountability"?

	Alternate for ED	Assembly and Accountability
A.✓	Shift Supervisor (SS)	required
B.	Shift Supervisor (SS)	NOT required
C.	Shift Technical Advisor (STA)	required
D.	Shift Technical Advisor (STA)	NOT required

G2.4.40 Emergency Procedures / Plan

Knowledge of SRO responsibilities in emergency plan implementations: (CFR: 41.10 / 43.5 / 45.11)

K/A MATCH ANALYSIS:

The question presents a plausible scenario where the Shift Manager is incapacitated when an emergency declaration is required. The question asks who the alternate for the Emergency Director is and whether Assembly and Accountability is required.

ANSWER / DISTRACTOR ANALYSIS:

A. Correct. Per 91101-C, Emergency Response Organization, the Shift Supervisor is the alternate for the ED position. Per 91401-C, Assembly and Accountability, it is required to perform assembly and accountability at an Alert level or higher.

B. Incorrect. Per 91101-C, Emergency Response Organization, the Shift Supervisor is the alternate for the ED position. Per 91401-C, Assembly and Accountability, it is required to perform assembly and accountability at an Alert level or higher. It is plausible the candidate may think a Site Area or higher is required.

C. Incorrect. Per 91101-C, Emergency Response Organization, the Shift Supervisor is the alternate for the ED position, it is plausible the candidate may think the STA can hold the Emergency Director position. Per 91401-C, Assembly and Accountability, it is required to perform assembly and accountability at an Alert level or higher.

D. Incorrect. Per 91101-C, Emergency Response Organization, the Shift Supervisor is the alternate for the ED position, it is plausible the candidate may think the STA can hold the Emergency Director position. Per 91401-C, Assembly and Accountability, it is required to perform assembly and accountability at an Alert level or higher.

REFERENCES:

91101-C, Emergency Response Organization (Table 1)
91401-C, Assembly and Accountability

VEGP learning objectives:

LO-LP-40101-01 Name the key individual responsible for the implementation of the EPIPs.
LO-LP-40101-06 State who fills the ED position initially.
LO-LP-40101-07 State who the designees are for the ED position. (SRO only)
LO-LP-40101-27 State the circumstances requiring assembly of VEGP non-essential personnel. (SRO only).

This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

TABLE 1

Sheet 1 of 3

ON-SHIFT EMERGENCY RESPONSE PERSONNEL AND RESPONSIBILITIES (1985305825)

EMERGENCY POSITION	PRIMARY	ALTERNATE	REPORTS TO	FUNCTION
Emergency Director	Shift Manager (SM)	Shift Supervisor	Plant Manager	Overall direction and control of the VEGP Emergency Response Organization prior to activation of the VEGP Emergency Response Facilities. After being relieved of his duties by a senior qualified member of plant management, the (SM) returns to direct operational duties of the on-shift operations personnel.

EMERGENCY POSITION	DESIGNEES	REPORTS TO	FUNCTION
CR Communicator (ENS)	On-Shift Operations Personnel	Emergency Director	Provide initial notification to the NRC and other offsite authorities as directed. Returns to normal duties when relieved of notification responsibilities by the TSC ENS Communicator.
CR Communicator (ENN)	Shift Administrative Assistant; On-Shift Operations Personnel	Emergency Director	Provide initial notification to the offsite authorities and designated plant staff.
Shift Technical Advisor (or SRO qualified as STA)	Shift Technical Advisor (or SRO qualified STA)	Emergency Director	Plant system engineering, repair, and corrective actions.
Status Loop	On-Shift Operations Personnel	Emergency Director	Maintain communications with Emergency Response Facilities.

1.0 PURPOSE

The purpose of this procedure is to provide instructions for performing assembly and accountability of all Protected Area (PA) personnel during emergency conditions.

2.0 RESPONSIBILITIES

2.1 The Emergency Director (ED) shall have the following responsibilities:

> **NOTE**
>
> A security related emergency may delay the ordering of assembly and accountability in order to protect plant personnel from the security threat. The decision to delay the order for assembly and accountability will be made by the Emergency Director. □

2.1.1 Ordering assembly and accountability of all PA personnel for an emergency classified as an Alert or higher.

2.1.2 Directing the Technical Support Center (TSC) Manager (if TSC is activated) to dispatch a Search and Rescue Team if assembly and accountability reveals a missing person.

2.1.3 Implementation of the Control Room Assembly and Accountability Checklist of this procedure.

2.2 The Operations Support Center (OSC) Manager (or the Health Physics Foreman if OSC is not activated) shall be responsible for forming, briefing and dispatching Search and Rescue Teams.

2.2.1 The OSC Manager shall be responsible for monitoring team activities and informing the TSC Manager of significant events.

2.3 The Search and Rescue Team Leader shall be responsible for following instructions and maintaining communications with the OSC Manager or HP Foreman if the OSC is not activated.

2.4 The Nuclear Security Captain (NSC) shall have the following responsibilities:

2.4.1 Directing and coordinating all functions necessary to perform accountability of personnel within the PA.

2.4.2 Assuring that accountability of all security personnel is conducted.

99. WE03EA2.1 001/1/2/LOCA -CD/DEPRESS/C/A-3.4/4.2/LOIT BANK/HL-18 NRC/SRO/TNT

Given the following conditions:

- The crew is performing 19010-C, "Loss of Reactor or Secondary Coolant."
- RCS pressure is 450 psig and stable.
- RHR flow is reading 0 gpm on both trains, suctions aligned to the RWST.
- RWST level is 36%.

The crew is at step # 22 "Check if RCS cooldown and depressurization is required."

Which ONE of the following procedural actions is the Shift Supervisor required to perform to mitigate the given plant conditions?

A. Remain in 19010-C, reset SI and stop both RHR pumps.

B. Go to 19111-C, "Loss of Emergency Coolant Recirculation."

C.✓ Go to 19012-C, "Post-LOCA Cooldown and Depressurization."

D. Return to step 17 of 19010-C and check for Cold Leg Recirculation capability.

WE03EA2.1 LOCA Cooldown and Depressurization

Ability to determine and interpret the following as they apply to the LOCA Cooldown and Depressurization:
(CFR 43.5 / 45.13)

Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where the crew is checking if an RCS cooldown and depressurization is required. From given plant conditions, the candidate has to choose the correct procedure to mitigate the plant conditions.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. This is plausible as step 13 has the candidate check if RHR pumps should be stopped. RCS pressure is currently > 300 psig and stable which are the requirements to stop the pumps as long as suction is aligned to the RWST (which it is). It is plausible the candidates may think this is a continuous action since the following step tells the candidate if RCS pressure lowers, to restart the RHR pumps and is a continuous action. Therefore, with pressure at 450 psig and stable, he may think stopping the pumps is appropriate.

B. Incorrect. It is plausible with RHR pressure at 450 psig and 0 RHR flow the candidate may think there is a recirculation problem if he doesn't realize the RHR discharge pressure is above the shutoff head of the RHR pumps. A transition to 19111-C would be appropriate if there was no recirculation capability.

C. Correct. Conditions are met to transition to 19012-C, ES-1.2 Post LOCA Cooldown and Depressurization.

D. Incorrect. This is plausible if the candidate think continuing with 19010-C is appropriate. This will recycle the candidate back to step 17 with the presented plant conditions to check for Cold Leg Recirculation capability.

REFERENCES:

19010-C, E-1.0 Loss of Reactor or Secondary Coolant, step 22, 13, 14, 17, and 23.

VEGP learning objectives:

LO-LP-37112-01 Using EOP 19012 as a guide, briefly describe how each step is accomplished.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes RO question

No

Does the question require one or more of the following?

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures

Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes SRO-only

EMERGENCY OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure provides actions to recover from a loss of reactor or secondary coolant. (Applicable in Modes 1, 2, and 3)

ENTRY CONDITIONS

- 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION
- 19005-C, ES-0.0 REDIAGNOSIS
- 19011-C, ES-1.1 SI TERMINATION
- 19020-C, E-2 FAULTED STEAM GENERATOR ISOLATION
- 19102-C, ECA-0.2 LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED
- 19112-C, ECA-1.2 LOCA OUTSIDE CONTAINMENT
- 19121-C, ECA-2.1 UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS
- 19221-C, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING
- 19222-C, FR-C.2 RESPONSE TO DEGRADED CORE COOLING
- 19231-C, FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK
- 19262-C, FR-I.2 RESPONSE TO LOW PRESSURIZER LEVEL

MAJOR ACTIONS

- ♦ Monitor Plant Equipment for Optimal Mode of Operations.
- ♦ Check for Subsequent Failure.
- ♦ Determine Optimal Method of Long-Term Plant Recovery.

CONTINUOUS ACTIONS

Step		Actions
__	3	– Maintain Seal Injection flow to all RCPs 8 to 13 gpm.
__	8	– Maintain SG NR levels between 10% [32% ADVERSE] and 65%. Monitor SG levels for uncontrolled level rise.
__	10	– Monitor PRZR pressure for proper PRZR PORV and Block Valve operation. Monitor RCS WR CL temperature less than 220°F to arm COPS.
__	11	– Monitor for ECCS flow reduction.
__	12	– Monitor CNMT conditions to secure CNMT Spray.
__	13	– Monitor 4160V AC Emergency Busses to restart ESF equipment.
CAUTION		
__	14	– Monitor RCS pressure less than 300 psig to restart RHR Pumps.
__	21	– Conserve Ultimate Heat Sink inventory.
__	24	– Monitor RHR Pumps suction condition for CNMT sump blockage.
__	25	– Monitor RCS WR Hot Leg temperatures less than 380°F to isolate SI Accumulators.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Initiate the following:	
__• Continuous Actions and Foldout Page.	
__• Critical Safety Function Status Trees per 19200-C, F-O CRITICAL SAFETY FUNCTION STATUS TREE.	
__2. Initiate NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION AND INITIAL ACTION.	
__*3. **Maintain Seal Injection flow to all RCPs - 8 TO 13 GPM.**	
4. Check if RCPs should be stopped:	
a. ECCS Pumps - AT LEAST ONE RUNNING:	__a. Go to Step 5.
__• CCP or SI Pump	
__b. RCS pressure - LESS THAN 1375 PSIG.	__b. Go to Step 5.
__c. Stop all RCPs.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__5. Check ACCW Pumps - AT LEAST ONE RUNNING.	__5. Try to start one ACCW Pump. __IF an ACCW Pump can NOT be started within 10 minutes of loss of ACCW, THEN stop all RCPs. IF an ACCW Pump can NOT be started within 30 minutes of loss of ACCW, THEN close ACCW containment isolation valves: __• ACCW SPLY HDR ORC ISO VLV HV-1979 __• ACCW SPLY HDR IRC ISO VLV HV-1978 __• ACCW RTN HDR IRC ISO VLV HV-1974 __• ACCW RTN HDR ORC ISO VLV HV-1975
__6. Place Containment Hydrogen Monitors in service by initiating 13130, POST -ACCIDENT HYDROGEN CONTROL.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. Check SGs secondary pressure boundaries:	
a. Identify faulted SG(s): __ ANY SG PRESSURE LOWERING IN AN UNCONTROLLED MANNER. -OR- __ ANY SG COMPLETELY DEPRESSURIZED.	__a. Go to Step 8.
b. Faulted SG(s) - ISOLATED: • Steamlines __• MSIVs __• BSIVs __• TDAFW supplies __• SG ARVs • Feedlines __• MFIVs __• BFIVs __• MFRVs __• BFRVs __• AFW valves __• SG blowdown valves __• SG sample valves	__b. <u>IF</u> faulted SG(s) are <u>NOT</u> isolated, <u>THEN</u> go to 19020-C, E-2 FAULTED STEAM GENERATOR ISOLATION.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***8. Check intact SG levels:**	
__a. NR level - AT LEAST ONE GREATER THAN 10% [32% ADVERSE].	__a. IF all SGs NR levels less than 10% [32% ADVERSE], THEN maintain total feed flow greater than 570 gpm.
__b. Maintain NR levels between 10% [32% ADVERSE] and 65%.	
__c. NR level - ANY RISING IN AN UNCONTROLLED MANNER.	__c. Go to Step 9.
__d. Go to 19030-C, E-3 STEAM GENERATOR TUBE RUPTURE.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. Check SG Tubes intact:	
__a. Direct Chemistry to take periodic activity samples of all SGs one at a time.	
b. Secondary radiation - NORMAL: • MAIN STM LINE MONITORS: __• RE-13120 (SG 1) __• RE-13121 (SG 2) __• RE-13122 (SG 3) __• RE-13119 (SG 4) • CNDSR AIR EJCTR/STM RAD MONITORS: __• RE-12839C __• RE-12839D (if on scale) __• RE-12839E (if on scale) • STM GEN LIQ PROCESS RAD: __• RE-0019 (Sample) __• RE-0021 (Blowdown __• SG sample radiation.	__b. Go to 19030-C, E-3 STEAM GENERATOR TUBE RUPTURE.
__c. Check SG levels - ANY RISING IN AN UNCONTROLLED MANNER.	__c. Go to Step 10.
__d. Go to 19030-C, E-3 STEAM GENERATOR TUBE RUPTURE.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***10. Check PRZR PORVs and Block Valves:**	
__a. Power to PRZR PORV Block Valves - AVAILABLE.	__a. Restore power to Block Valves.
__b. PRZR PORVs - CLOSED.	__b. IF PRZR pressure less than 2315 psig, THEN verify closed affected PRZR PORV(s). __ IF any PRZR PORV can NOT be closed, THEN close its Block Valve.
__c. PRZR PORV Block Valves - AT LEAST ONE OPEN.	__c. IF NOT closed to isolate an excessively leaking or open PRZR PORV, AND WHEN PRZR pressure is greater than 2185 psig, THEN verify open at least one PRZR PORV Block Valve.
__d. Any RCS WR CL temperature - LESS THAN 220°F.	__d. WHEN any RCS CL temperature lowers to less than 220°F, THEN arm COPS. __Go to Step 11.
__e. Arm COPS.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*11. **Check if ECCS flow should be reduced:**	
__a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE].	__a. Go to Step 12.
b. Secondary Heat Sink:	__b. Go to Step 12.
__ Total feed flow to intact SG(s) - GREATER THAN 570 GPM. -OR- __ NR level in at least one intact SG - GREATER THAN 10% [32% ADVERSE].	
__c. RCS pressure - STABLE <u>OR</u> RISING.	__c. Go to Step 12.
__d. PRZR level - GREATER THAN 9% [37% ADVERSE].	d. Try to stabilize RCS pressure: __• Use Normal PRZR Spray if Instrument Air to Containment available. __• Do <u>NOT</u> use PRZR PORVs to stabilize RCS pressure. __Go to Step 12.
__e. Go to 19011-C, ES-1.1 SI TERMINATION.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*12. **Check if Containment Spray should be stopped:**	
__a. CS Pumps - RUNNING.	__a. Go to Step 13.
b. Containment pressure - LESS THAN 15 PSIG.	__b. WHEN Containment pressure is less than 15 psig, THEN go to Step 12.c. __Go to Step 13.
c. Any Containment radiation levels - INDICATE HIGH DUE TO PRIMARY LOCA: __ RE-002 __ RE-003 __ RE-005 __ RE-006	c. Perform the following: __1) Reset Containment Spray signal. __2) Stop Containment Spray Pumps. 3) Close CNMT SPRAY ISO VLV: __• HV-9001A __• HV-9001B __Go to Step 13.
d. Operate CS Pumps: __• Minimum of 2 hours. __• At least 1.5 hours in recirculation mode.	__d. WHEN CS Pumps have operated for at least 2 hours AND in the recirculation mode for at least 1.5 hours, THEN perform Step 12.c RNO.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

If offsite power is lost after SI reset, action is required to restart the following ESF equipment if plant conditions require their operation:

- RHR Pumps
- SI Pumps
- Post-LOCA Cavity Purge Units
- Containment Coolers in low speed (Started in high speed on a UV signal).
- ESF Chilled Water Pumps (If CRI is reset).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. Check if RHR Pumps should be stopped:	
a. RHR Pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST.	a. Go to Step 15.
b. RCS pressure:	b.
1) Greater than 300 psig.	1) Go to Step 16.
2) Stable or rising.	2) Go to Step 15.
c. Reset SI.	c. IF SI will NOT reset, THEN initiate ATTACHMENT B.
d. Stop RHR Pumps.	d.
*14. **IF RCS pressure lowers in an uncontrolled manner to less than 300 psig, THEN restart RHR Pumps.**	14.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22. Check if RCS cooldown and depressurization is required:	22.
a. RCS pressure - GREATER THAN 300 PSIG.	a. IF RHR Pump flow is greater than 500 gpm, THEN go to Step 23.
b. Go to 19012-C, ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION.	b.
23. Check if transfer to Cold Leg recirculation is required:	23.
a. RWST level - LESS THAN 29%.	a. Return to Step 17.
b. Go to 19013-C, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. Check Cold Leg recirculation capability:	17. Restore equipment to operable status if possible.
a. Power available to:	IF cold leg recirculation capability can NOT be verified from at least one flow path, THEN go to 19111-C, ECA-1.1 LOSS OF EMERGENCY COOLANT RECIRCULATION.
Train A components:	
• HV-8811A - CNMT SUMP TO RHR PMP-A SUCTION	
• RHR Pump A - OPERABLE	
• HV-8809A - RHR PMP-A TO COLD LEG 1&2 ISO VLV	
• RHR Heat Exchanger A - OPERABLE	
-OR-	
Train B components:	
• HV-8811B - CNMT SUMP TO RHR PMP-B SUCTION	
• RHR Pump B - OPERABLE	
• HV-8809B - RHR PMP-B TO COLD LEG 3&4 ISO VLV	
• RHR Heat Exchanger B - OPERABLE	

STEP: Check If RCS Cooldown And Depressurization Is Required

PURPOSE: To determine the method of long-term plant recovery

BASIS:

The operator should stay in E-1 only for loss of reactor coolant accidents for which the RCS pressure is less than the low-head SI pump shutoff head and flow from the low-head SI pumps has been verified. The low-head SI pump flow should be verified even though the RCS pressure is less than the shutoff head pressure of the low-head SI pumps plus allowances for normal channel accuracy and (with an adverse containment) post accident transmitter errors. Since the post accident transmitter errors are added on to determine the pressure requirement, the actual plant pressure may be significantly less.

For any break in the RCS for which the RCS pressure remains above the shutoff head pressure of the low-head SI pumps, the operator should transfer to guideline, ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. If the RCS pressure is less than the low-head SI pump shutoff head but low-head SI pump flow into the RCS cannot be verified, the operator should also transfer to ES-1.2. From this point on ES-1.2 would be used for plant recovery.

ACTIONS:

- o Determine if RCS pressure is greater than (B.07) psig [(B.08) psig for adverse containment]
- o Determine if low-head SI pump flow is greater than (S.03) gpm
- o Transfer to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, step 1

INSTRUMENTATION:

- o RCS pressure indication
- o Low-head SI pump flow indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

100. WE09EA2.2 001/1/2/NAT CIRC - LICENSE/C/A - 3.4/3.8/BANK-HL-17 NRC/HL-18 NRC/SRO/TNT

Cooldown per 19002-C, "Natural Circulation Cooldown," is in progress.

The Shift Supervisor is at the step in 19002-C to check that a steam void in Reactor Vessel does NOT exist.

The following data is noted:

Time	RVLIS Upper Range	PRZR Level
10:00	100	25
10:15	100	25
10:30	98	28
10:45	84	58

Which ONE of the following completes the following statement?

Based on the given conditions, the NEXT action required by 19002-C is to ___(1)___

and

the cooldown rate allowed for the procedure to be implemented is ___(2)___ .

A✓ (1) repressurize RCS within limits of Tech Spec LCO 3.4.3 to collapse potential voids in system and continue with 19002-C

(2) < 50°F per hour

B. (1) repressurize RCS within limits of Tech Spec LCO 3.4.3 to collapse potential voids in system and continue with 19002-C

(2) < 100°F per hour

C. (1) Go to 19003-C, "Natural Circulation Cooldown With Void In Vessel (With RVLIS)"

(2) < 50°F per hour

D. (1) Go to 19003-C, "Natural Circulation Cooldown With Void In Vessel (With RVLIS)"

(2) < 100°F per hour

WE09EA2.2 Natural Circulation Operations

Ability to determine and interpret the following as they apply to the Natural Circulation Operations:
(CFR: 43.5 / 45.13)

Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a natural circulation cooldown is in progress. The candidate must determine from given conditions that indicate a void is occurring, the required procedure actions and the cooldown rate limit for the appropriate procedure selected.

SRO 10CFR55.43 (b)(5)

DISTRACTOR ANALYSIS

A. Correct. Part 1 is correct, per step 23 RNO, the crew will repressurize the RCS within limits of Technical Specification LCO 3.4.3 (PTLR) to collapse potential voids in system and continue cooldown.

Part 2 is correct. The cooldown rate limit in 19002-C is < 50°F.

B. Incorrect. Part 1 is correct, per step 23 RNO, the crew will repressurize the RCS within limits of Technical Specification LCO 3.4.3 (PTLR) to collapse potential voids in system and continue cooldown.

Part 2 is incorrect. The cooldown rate limit in 19003-C is < 100°F.

C. Incorrect. Part 1 is incorrect. A transition to 19003-C is NOT required, per step 23 RNO, the crew will try to repressurize the RCS to collapse the voids.

Part 2 is correct. The cooldown rate limit in 19003-C is < 50°F.

D. Incorrect. Part 1 is incorrect. A transition to 19003-C is NOT required, per step 23 RNO, the crew will try to repressurize the RCS to collapse the voids.

Part 2 is incorrect. The cooldown rate limit in 19003-C is < 100°F.

REFERENCES

WOG Background Document, ES-0.2, Natural Circulation Cooldown
19002-C, "ES-0.2 Natural Circulation Cooldown", step # 10 and 23 and RNO.
19003-C, "ES-0.3 Natural Circulation Cooldown With Voids in Vessel (With RVLIS)", step # 7
HL-17 NRC Exam (This is a question used from last 2 exams under a different KA #).

VEGP learning objectives:

LO-LP-37012-15 State the limitations on subcooling and cooldown rate associated with natural circulation cooldown. Include the bases for any variations.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Maintain RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR): __ Use 60°F/HR curve and RCS Cold Leg temperature.	
23. Check that steam void in Reactor Vessel does NOT exist: __ • PRZR level - NO UNEXPECTED LARGE VARIATIONS. __ • RVLIS Upper Range - GREATER THAN 94%.	__23. Repressurize RCS within limits of Technical Specification LCO 3.4.3 (PTLR) to collapse potential voids in system and continue cooldown. __ IF RCS depressurization must continue, THEN go to 19003-C, ES-0.3 NATURAL CIRCULATION COOL DOWN WITH VOID IN VESSEL (WITH RVLIS).
***24. Check if ECCS should be locked out:**	
__a. Check RCS WR pressure - LESS THAN 950 PSIG.	__a. WHEN RCS WR pressure is less than 950 psig, THEN go to Step 25. __ Go to Step 29.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***7. Continue RCS cooldown and initiate depressurization:**	
__a. Maintain cooldown rate in RCS Cold Leg - LESS THAN 100°F/HR.	
__b. Maintain RCS subcooling - GREATER THAN 44°F.	
c. Maintain RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR):	
__ Use 100°F/hr curve and RCS Cold Leg temperature.	
__d. Check letdown - IN SERVICE.	__d. Depressurize RCS using one PRZR PORV. __ Go to Step 8.
__e. Depressurize RCS using Auxiliary Spray.	
***8. Control PRZR level:**	
__a. Level - GREATER THAN 19%.	__a. Control charging and letdown as necessary to raise PRZR level to greater than 19%.

Step 8 continued on next page

1. WE10EG2.4.47 001/1/2/NAT CIRC - TRENDS/H 4.2 / 4.2/NEW/H-17 NRC/SRO/EMT/GCW

Cooldown per 19002-C, "ES-0.2 Natural Circulation Cooldown", is in progress.

The SS is at the step in 19002-C to check that a steam void in Reactor Vessel does NOT exist.

The following data is noted:

Time	RVLIS Upper Range	PRZR Level
10:00	100	25
10:15	100	25
10:30	98	28
10:45	84	58

Which one of the following correctly completes the following statement?

Based on the given conditions, the NEXT action required by 19002-C is to ___(1)___

and

the MAXIMUM cooldown rate allowed for the procedure to be implemented is ___(2)___ .

A.✓ (1) repressurize RCS within limits of TS LCO 3.4.3 to collapse potential voids in system and continue with 19002-C

(2) 50°F per hour

B. (1) repressurize RCS within limits of TS LCO 3.4.3 to collapse potential voids in system and continue with 19002-C

(2) 100°F per hour

C. (1) GO TO 19003-C, "ES-0.3 Natural Circulation Cooldown With Void In Vessel (With RVLIS)"

(2) 50°F per hour

D. (1) GO TO 19003-C, "ES-0.3 Natural Circulation Cooldown With Void In Vessel (With RVLIS)"

(2) 100°F per hour

STEP: Verify Steam Void In Reactor Vessel Does Not Exist

PURPOSE: To check for void formation in the reactor vessel.

BASIS:

If abnormal RCS conditions such as large variations in pressurizer level during normal charging or spraying operations occur, a steam void may be present in the reactor vessel upper head. To collapse the void, the RCS is repressurized within the plant specific limitations outlined by the figures in the Appendix to this section. These figures, which illustrate the acceptable operating region from the pressure-temperature relationship were described in Step 13.

Though the RCS cooldown can continue to RHR System entry conditions, extra care should be taken during any subsequent RCS depressurization. In order to reach RHR System entry conditions for pressure, the upper head has to be cooled sufficiently so that RCS depressurization will not cause the PRZR level to behave abnormally.

RVLIS can also be used to detect steam voiding in the upper head region. A full RVLIS upper range indication implies at least saturated conditions exist in the upper head region. If pressure is decreased and RVLIS is monitored, any void formation would be detected.

With no CRDM fans running a waiting period, during which the RCS should not be depressurized, is required to prevent upper head steam voiding.

The waiting period (indicated in the Appendix) allows the upper head to cool off to a temperature less than saturation for 400 psig before continuing with the depressurization and is dependent upon the plant's type of upper support plate (USP). These cool-off periods vary due to the differences in the USP thickness and the upper head fluid volume. Analysis for the cool-off, which takes into account the upper head fluid heat as well as the metal heat, has shown the following:

a. For inverted top hat USP plants (12-inch thick USP and 846.6 ft^3 upper head volume), it takes 27 hours for the upper head to cool off to the appropriate temperature described above.

b. For top hat USP plants (5-inch thick USP and 507.8 ft^3 upper head volume), it takes 8 hours for the upper head to cool off to this temperature.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__b. As RCS cooldown is initiated, hold HS-0500A and HS-0500B in the BYPASS INTERLOCK position until RCS temperature is less than 550°F.	
*10. **Initiate RCS cooldown to cold shutdown:**	
__a. Check RCS boron concentration greater than required boron concentration for xenon free cold shutdown.	__a. Return to Step 6.
__b. Maintain cooldown rate in RCS Cold Legs - LESS THAN 50°F/Hr.	
__c. Dump steam to Condenser using Steam Dumps.	__c. Use SG ARVs.
__d. Maintain SG NR levels - AT APPROXIMATELY 65%.	
__e. Check RCS cooldown at 15 minute intervals.	
f. Maintain RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR):	
__ Use 60°F/HR curve and RCS Cold Leg temperature.	
__g. Perform other appropriate actions required to take the unit to cold shutdown by initiating 12006-C, RCS COOLDOWN TO COLD SHUTDOWN.	